August 11, 2005

MEMORANDUM TO: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Farouk Eltawila, Director ***/RA/***
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: PROPOSED CLOSURE OF GENERIC SAFETY ISSUE 80, "PIPE BREAK EFFECTS ON CONTROL ROD DRIVE HYDRAULIC LINES IN THE DRYWELLS OF BWR MARK I AND II CONTAINMENTS"

The Office of Nuclear Regulatory Research (RES) has completed the technical assessment of Generic Safety Issue (GSI) 80, "Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments," in accordance with Management Directive 6.4, "Generic Issues Program," and is proposing that the issue be closed.

The issue was originally identified in 1978 by the Advisory Committee on Reactor Safeguards (ACRS) during the operating license reviews of some BWRs. The ACRS posed questions concerning the likelihood and effects of a loss-of-coolant accident (LOCA) which could cause interactions with the control rod drive (CRD) hydraulic lines in such a way as to prevent rod insertion, creating the potential for recriticality when the core is reflooded. The staff investigated this potential problem and concluded in 1984 that the existing Standard Review Plan (SRP) criteria were adequate to assure integrity of the CRD hydraulic lines. These criteria assumed conservative failure stresses and break locations in coolant piping, and require examination of the effects of pipe whip and jet impingement on essential safety components (including the CRD hydraulic lines) for approximately 100 breaks. Thus, the issue was given a LOW-priority ranking. However, during site visits associated with GSI-156.6.1, "Pipe Break Effects on Systems and Components," some new piping configurations were discovered that were not considered in the original evaluation of GSI-80. In light of the concerns of GSI-156.6.1, the Office of Nuclear Reactor Regulation (NRR) recommended in March 1998 that the priority of GSI-80 be reassessed. The initial screening of GSI-80 was completed in 2003, and technical assessment of the issue was pursued.

In the technical assessment of GSI-80 (attached), an analysis of significant high energy piping breaks in the areas of the insertion and withdrawal CRD piping was completed with the use of the ANSYS code. Results indicated that the impacting pipe would have insufficient energy for the CRD pipe to be crimped totally closed following a high energy pipe break. Actual pipe-to-pipe impact testing also showed that, as the postulated energy of the impacting piping increases, this piping would break open before being crimped closed (zero flow area).

Scram motion in a BWR CRD is effected by admitting the pressure in the scram accumulator to the area below the drive piston, and venting the area above the piston to the scram discharge volume, which is at atmospheric pressure. The CRDs are equipped with a ball check valve which will admit reactor water below the drive piston if the inlet line pressure falls below reactor pressure. Thus, neither crimping nor breaking the insert line will prevent a scram when the reactor is at power. In contrast, crimping the withdraw line shut would inhibit a scram. However, breaking this line, thereby venting it to atmospheric pressure, will cause the drive to scram. Since the piping is expected to fail open before it is crimped closed, the control rods will scram using reactor pressure. RES is planning on briefing the ACRS on the technical assessment of GSI-80 in September 2005.

Attachment: Technical Assessment of GSI-80, “Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments”

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TECHNICAL ASSESSMENT OF GSI-80
"PIPE BREAK EFFECTS ON CONTROL ROD DRIVE HYDRAULIC LINES IN THE DRYWELLS OF BWR MARK I AND II CONTAINMENTS"

1. Introduction

This report provides the details of a technical assessment performed for Generic Safety Issue (GSI) 80, "Pipe Breaks Effects on Control Rod Drive Hydraulic Lines in the Drywell of [Boiling-Water Reactor] BWR Mark I and II Containments," in accordance with Management Directive (MD) 6.4, "Generic Issues Program."

1.1 Background

The Advisory Committee on Reactor Safeguards (ACRS) originally identified GSI-80 in 1978, in response to concerns regarding the likelihood and effects of a loss-of-coolant accident (LOCA), which could cause interactions with the control rod drive (CRD) hydraulic pipes in a manner that could prevent rod insertion (e.g., crimping of the withdrawal CRD piping), and create the potential for recriticality when the core is reflooded. This scenario could lead to a situation in which the emergency core cooling system (ECCS) may not be sufficient to remove this extra energy, which could result in coolant boiloff, containment failure, and core melt. The most likely condition that would prevent control rod insertion is crimping of insertion/withdrawal CRD piping bundles as a result of a high-energy recirculation system pipe break inside containment.

The following is a brief chronology of the various analyses performed for GSI-80:

(1) During a meeting with the ACRS on January 6, 1983, the staff of the U.S. Nuclear Regulatory Commission (NRC) reported the results of its investigation of GSI-80. In particular, the staff had concluded that the existing standard review plan (SRP) criteria were adequate to ensure the integrity of the CRD hydraulic lines. However, the ACRS remained concerned about BWR Mark I (MI) and Mark II (MII) containments, which are smaller and more congested than the BWR Mark III (MIII) containments upon which the staff's analysis was based.

(2) Following additional analysis in January 1984, the NRC staff assigned GSI-80 a LOW-priority ranking. Then, in 1995, the staff placed GSI-80 in the "DROP" category based on the results of a cost-benefit analysis.

(3) During site visits associated with GSI-156.6.1, "Pipe Break Effects on Systems and Components," the NRC staff discovered some new piping configurations that were not considered in the original evaluation of GSI-80. Thus, in March 1998, the NRC's Office of Nuclear Reactor Regulation (NRR) indicated that the agency should reassess the priority of GSI-80, given the concerns regarding GSI-156.6.1.

(4) In November 1999, the NRC issued a report (Ref.1), prepared by the Idaho National Engineering and Environmental Laboratory (INEEL), which was based on the results of site walkdowns for GSI-156.6.1. That report identified two events (Events 5 and 10) that may damage the CRD piping in BWR plants.

BWR Event 5, described in the INEEL report (Ref. 1), involves the rupture of reactor coolant system (RCS) piping inside containment, resulting in the failure of a number of CRD piping bundles by crimping the insertion/withdrawal piping. This results in a large

LOCA with failure to scram the reactor. The mean core damage frequency (CDF) was calculated to be 5 x 10^{-6} event/reactor year (RY). The INEEL report offered no quantitative bases for the origin of the high failure probabilities.

BWR Event 10, also described in the INEEL report (Ref. 1), involves a rupture of the residual heat removal (RHR) piping inside containment, resulting in the failure of a number of CRD piping bundles by crimping the insertion/withdrawal piping. This results in a large LOCA with failure to scram the reactor. The mean CDF was calculated to be 2.5 x 10^{-6} event/RY. Again, the INEEL report offered no quantitative bases for the origin of the high failure probabilities for this event.

(5) The NRC screening evaluation for GSI-80 (Ref. 2), dated February 14, 2003, provided qualitative information. However, it did not credit the existence of pipe whip restraints on high-energy piping systems that are located near the CRD piping bundles. The NRC screening evaluation also did not comment on licensees' pipe break analyses, which showed potential locations for pipe failures. The basis for pursuing a technical assessment of GSI-80 was the calculated values of large early release frequencies (LERFs), which slightly exceed the threshold (10^{-6} event/RY) specified in the Handbook to MD 6.4. The following is a synopsis of the screening evaluation of GSI-80.

Safety Significance: Recriticality during the course of an accident has no direct effect on the health and safety of the public. However, failure to insert a significant number of control rods could pose two separate safety problems. First, when the core is reflooded by cold emergency core cooling water, the reactor may undergo a cold water reactivity transient if the core is not subcritical. The cold water can insert considerable positive reactivity, which means that portions of the core where control rods failed to insert can return to a significant power level and experience damage. Secondly, the recirculation phase of emergency core cooling is sized to carry away decay heat. If fission heat is not shut off, the ECCS may not be sufficient to remove this extra energy, resulting in coolant boil-off, core-melt, and potential containment failure.

Evaluation: A BWR control rod is scrammed by applying pressure from an accumulator or from the reactor vessel to the volume below the CRD piston and venting the volume above the piston to the scram discharge volume which is near atmospheric pressure. If the insert line is either blocked or broken, a ball check valve built into the CRD will admit reactor water to the volume under the piston. Thus, the insert line is necessary for scram only when the reactor pressure is low, e.g., during reactor startup.

Breaking the withdraw line will open the volume above the piston to atmospheric pressure and thus cause (not prevent) a scram. The only way to prevent a scram by mechanical damage to the CRD lines is to crimp the withdraw line shut. Breaking or crimping an insert line will prevent a scram only at low reactor pressure at which time the high energy coolant lines, which are to provide the crimping force, are also at low pressure and the reactor is also at very low power. CRD hydraulic lines originate at the CRD flanges. They are routed up from these flanges, curve 90°, and travel horizontally between the CRD housings. In most cases, the lines are divided into two banks which exit the area under the vessel in two penetrations of the reactor support pedestal placed 180E apart. After traversing the drywell area, the lines exit the containment via two containment penetrations and are then routed to the two banks of hydraulic control units.

It should be noted that the outcome of the accident under consideration is relatively insensitive to scram timing, so long as the rods are successfully inserted. A small LOCA will not cause a reactor scram until either the water level drops to the scram setpoint or the drywell pressure rises to its setpoint. A large LOCA will depressurize the reactor and stop the fission chain reaction by high voiding of the moderator and the rods need not be inserted until the blowdown is complete. Thus, the interest was in complete rather than partial obstruction of the CRD lines, since partial obstruction would only delay, not prevent, the scram.

No credit was taken for the possibility that non-inserted rods might be widely dispersed and thus may not lead to recriticality. This was not as conservative as it first appeared. The CRD lines are not necessarily routed in such a manner as to disperse the drives they control, and blockage of adjacent lines may well inhibit scram in adjacent CRDs. (Two adjacent control rods can achieve criticality if withdrawn under cold conditions in a BWR.) Finally, insert and withdrawal lines were considered equally, since a large LOCA could depressurize the reactor before a rod with a crimped insert line is completely inserted. (This was in fact quite conservative.) The standby liquid control system (SBLC) is normally capable of borating the moderator to 600 ppm of natural boron (referenced to cold water density) plus a 25% safety margin. This concentration would render the core up to 5% subcritical with all control rods fully removed at cold, xenon-free conditions at the most reactive point in core life. However, following a large LOCA, the SBLC effectiveness is reduced by the diluting effect of the suppression pool, which normally contains about 7½ vessel inventories. Thus, the SBLC can realistically borate only to about 88 ppm. Based on calculations done for ATWS, this would reduce power to roughly 75% of rated (with no rod insertion) but would not shut the reactor down.

Several effects help bring power down. First, existing xenon, augmented by xenon increase, holds power down for roughly 24 hours after the accident. Second, the recirculation pumps are no longer providing forced flow through the core, which tends to bring power down by allowing more voiding. Finally, unless the pipe break area is small enough to limit leakage to less than ECCS injection flow, the water level will drop into the core, which will greatly reduce moderator density in the upper portion of the core. Nevertheless, the core must eventually be brought to cold shutdown by means of the SBLC. Over the long term, this would not be difficult, since more sodium pentaborate mixture could be added to the SBLC so long as the secondary containment remained accessible. It was assumed that the SBLC would be ultimately used to render the core sub-critical over a span of several days.

In the area under the reactor vessel, there is only one high-energy line, a two-inch lower vessel head drain which is one input to the RWCU system. This line is not considered a significant hazard to the CRD lines for several reasons:

(1) The CRD lines are routed below a set of I-beams. (The CRD housing support is attached to hanger rods which descend from these beams). Thus, the CRD lines are well shielded from the drain line which is above the I-beams.

(2) Breakage of this drain line would be a small LOCA. Normally, the reactor would continue to run, with the only problems being loss of some RWCU flow and a steam-feed flow mismatch. The reactor would not scram until the drywell pressure rose to the scram setpoint. This does not isolate the reactor and main feedwater would continue. Although some rods might fail to insert, and the resulting fission heat would

have to be accommodated, the core would not uncover, and there would be no fuel melting.

(3) Even if main feedwater were lost, HPCI has the capacity to handle a 2-inch break (double-ended) with enough extra flow to supply about 40 bundles operating at average power. Again, the core would not uncover.

(4) If HPCI is insufficient, ADS can vent about 38% of rated steam flow. Thus, unless more than 38% of the rods fail to insert, ADS should be able to depressurize the vessel to the point where the high-capacity low pressure ECCS would keep the core flooded.

In any of these small-break scenarios, there would be no fuel melting because the core would not uncover, and there would be no reflood-induced reactivity transient. Depending on the number of control rods that fail to insert, steam production might exceed the turbine bypass capacity, or the MSIVs might close. In such a case, the heat sink provided by the RHR system would likely be insufficient to accommodate the extra heat, and the containment would eventually overpressurize and fail. This would not result directly in a major release of radioactivity, because there would be no severe fuel damage. In theory, the ECCS systems would eventually overheat, experience loss of NPSH, or deplete the suppression pool, and the core would eventually uncover. This situation would be alleviated by the fact that, as the suppression pool depletes, the standby liquid control system would become more effective because the concentration of sodium pentaborate in the coolant would increase as coolant boiled off, and fission heat would diminish. Alternatively, the standby coolant supply system could be used to augment the coolant supply.

In the area between the reactor support pedestal and the drywell wall, the situation is different. Here, the CRD lines pass near the reactor coolant piping and headers. The recirculation piping exits the vessel from two nozzles located near the bottom of the annulus and travels down through the general area where the CRD lines are located to the recirculation pumps which are at a still lower elevation. Flow from the pumps travels through two pipes up to two semi-circular manifolds, which again are in the general area of the CRD lines. Each manifold then supplies driving flow to the jet pumps through a series of risers, one riser for every two jet pumps. The CRD hydraulic lines cross this area under the manifolds. The usual practice is to route each bank in an array of six horizontal rows of hydraulic lines.

Thus, the major threat to the CRD lines comes from the breakage of a large pipe connected to the reactor vessel and containing fluid at reactor pressure. Such piping includes the steam lines, feedwater lines, recirculation piping, RHR piping, and core spray piping.

The steam, feedwater, and core spray lines are located considerably higher in the drywell. This piping is not considered a significant hazard because of its distance from the CRD lines and the rather narrow annular gap through which any missiles or jets would have to pass. Thus, this analysis focused on the recirculation and RHR piping.

1.2 Technical Assessment Objective

The objective of this technical assessment of GSI-80 is to perform an in-depth analysis of the high-energy pipe break interactions documented in the NRC's preliminary evaluations of General Electric (GE) BWR MI and MII power plants (Refs. 1 and 2). Toward that end, this assessment reviews the probability of events that contribute to the CDF for BWR Events 5 and 10 (identified in Ref. 1). In addition, this assessment includes a rigorous analysis of the scenario that predicted complete closure of the CRD pipes (zero flow), to determine the realistic probability of RCS or RHR piping crimping the CRD piping. Finally, this assessment uses the revised CDF resulting from the rigorous analysis performed for this assessment, to determine an appropriate priority rating for GSI-80, in accordance with MD 6.4.

1.3 BWR Mark I and Mark II Control Rod Drive Arrangements

The general arrangements of CRD pipes as they exit the reactor pressure vessel (RPV) pedestal and penetrate the containment in the BWR MI-GE2 design differs from that of other BWR MI designs (e.g., GE3 or GE4). The MI-GE2 design, as shown in Figure 1, has three sets of CRD piping bundles, all of which exit one side of the RPV pedestal 31 degrees apart from the 270-degree location, and then proceed horizontally outward to the containment. By contrast, the MI (e.g., GE3 or GE4) and MII (e.g., GE4 or GE5) BWR designs each have four sets of CRD piping bundles with similar CRD bundle arrangements. In these BWR designs, two sets of CRD piping bundles exit the RPV pedestal at approximately ±22 degrees from the 90-degree location, and another two sets of CRD piping bundles exit the RPV pedestal at approximately ±22 degrees from the 270-degree location. The four CRD bundles proceed upward and then horizontally before going through the containment (see Figure 2).

2. Event Quantification Process

The INEEL report (Ref. 1) ranked the pipe break events according to their impact on CDF, containment failure, and offsite consequences. Three ranking categories used were:

(1) High: potential to increase CDF or offsite consequences by more than 100 percent, or containment failure probability of nearly 1.0

(2) Medium: potential to increase CDF or offsite consequences by 1 to 100 percent, or containment failure probability in the range of 0.01 to nearly 1.0

(3) Low: potential to increase offsite consequences by less than 1 percent, or containment failure probability of less than 0.01.

The rankings were qualitative; however, INEEL used individual plant examination (IPE) studies for guidance in the qualitative ranking process. BWR Events 5 and 10, which involve failure of CRD bundles as a result of pipe whip caused by RCS or RHR piping ruptures, were ranked as having medium potential impact on CDF and offsite consequences to MI and MII BWR plants.

The INEEL report (Ref. 1) used an event quantification process involving four factors, which were multiplied together to obtain a CDF value resulting from the pipe rupture event:

(1) IE: pipe rupture initiating event (IE) frequency

(2) PIPETYPE: fraction of piping considered in IE that is from the system in question

(3) TYPEFRAC: fraction of system piping that can cause another system failure from pipe whip or jet impingement

(4) RUPTPROB: probability of pipe whip or jet impingement causing another system failure

Multiplying these four factors together yields a point estimate of the CDF for each postulated event:

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

Each event in the INEEL report was modeled as log-normal (which is typical in most probabilistic risk assessments), and each was characterized by a mean value (frequency or probability) and an error factor (95^{th} percentile/median). The subsequent sections of this assessment provide a more detailed description of the four factors referenced in the INEEL report.

It should be noted that the CDF as defined above is based on LOCA frequency with an incomplete scram. This is conservative because if only one or two CRD rods fail to scram, the core may not return to criticality after reflood. However, if several rods fail to scram, there will be localized melting but not necessarily a full core melt.

3. Pipe Rupture Initiating Event Frequency

Predictions vary considerably with regard to the IE frequency for a large-diameter break LOCA. Table 1, "Failure frequencies (events/year) for the large-diameter BWR plant pipe," lists several sources for failure frequencies. The INEEL report (Ref. 1) used the most conservative value (10^{-4} event/RY) for pipe break frequency based on NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated December 1990.

4. Fraction of Piping Considered in Initiating Event (PIPETYPE)

The INEEL report (Ref. 1) describes "PIPETYPE" as the fraction of piping considered in IE that is from the system in question. This value is arrived at by estimating the number of linear feet of system piping that is inside containment and the total number of linear feet of piping that is inside containment. To determine a more realistic value of PIPETYPE, this assessment focuses on the number and fraction of potential break points in large-diameter piping inside containment. Piping of with diameter of 3 inches or larger is considered "large diameter" for this assessment. Pipe breaks involving large-diameter piping would have a higher probability of disabling the CRD insertion/withdrawal piping.

4.1 Number of Postulated Pipe Breaks Inside Containment

The number of postulated pipe breaks within each high-energy piping system can be determined from the detailed pipe break analysis inside containment, as documented in the final safety analysis reports (FSARs) of representative BWR MI and MII nuclear plants. However, this information is not available from the FSARs of the BWR MI plants that the NRC staff evaluated for the effects of pipe break inside containment before the agency issued Regulatory Guide (RG) 1.46, "Protection Against Pipe Whip Inside Containment," in May 1973. These BWR MI plants are referred to as Systematic Evaluation Program Phase III (SEP-III) plants.

There is a potential lack of uniformity among reviews of pipe break effects for SEP-III plants because of a lack of documented acceptance criteria. Therefore, to determine PIPETYPE, this technical assessment used the following pipe break analyses for non SEP-III plants, which are considered to be representative of all BWR MI and MII plants. This is acceptable because the RCS/RHR piping layout inside containment is identical for the various plants of the same GE version (GE2, GE3, GE4, and GE5):

- **BWR Mark I-GE2**: The Oyster Creek Nuclear Power Plant performed an analysis of pipe breaks inside containment, and included the results in Appendix 3.6B, "Analysis of Pipe Breaks Inside Containment," to the Oyster Creek FSAR.
- **BWR Mark I-GE3**: There are no post-SEP-III BWR MI-GE3 design plants; therefore, a detailed pipe stress analysis for is not available for high-energy pipe breaks in such plants.
- **BWR Mark I-GE4**: Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping," of the FSAR for Hope Creek Nuclear Generating Station provides an analysis of pipe breaks in BWR MI-GE4 design plants.
- **BWR Mark II-GE4**: Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping," of the FSAR for Susquehanna Steam Electric Station provides an analysis of pipe breaks in BWR MII-GE4 design plants.
- **BWR Mark II-GE5**: Nine Mile Point Unit 2 provided information in correspondence to the NRC, dated July 18, 1986, which included a detailed analysis of the dynamic effects associated with the postulated rupture of piping. This analysis provided an extensive update of information previously contained in FSAR Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping" (now deleted).

4.2 PIPETYPE

As shown in Table 2, the "PIPETYPE" for MI and MII plants is calculated considering the number of large-diameter pipe break points in the RHR or RCS piping as a fraction of the total number of large-diameter pipe break points inside the containment.

5. <u>Fraction of System Piping That Can Impact CRD Piping (TYPEFRAC)</u>

The INEEL report (Ref. 1) described "PIPETYPE" as the fraction of system piping that can cause another system failure from pipe whip or jet impingement. For this assessment, TYPEFRAC is the fraction of RCS or RHR piping length located inside the containment that can impact the CRD piping by pipe whip or jet impingement. The magnitude of TYPEFRAC was determined by reviewing the representative BWR plant drawings and pipe break evaluations, as documented in FSAR Section 3.6. Table 2 lists the TYPEFRAC values for a representative selection of BWR MI and MII plants.

6. <u>Probability of Pipe Whip or Jet Impingement Causing Another System Failure (RUPTPROB)</u>

This section addresses the probability of CRD system failure attributable to a break in RCS/RHR piping for BWR MI and MII plants. This determination is based on review of plant drawings, detailed structural analysis (if required) to determine the deformation in the RCS/RHR piping, and the behavior of the CRD piping after being impacted by RCS/RHR piping.

6.1 Mark I-GE2 Containment

A review of plant drawings indicates that a pipe break in the RHR piping in a GE2 containment cannot impact CRD piping by pipe whip or jet impingement. The TYPEFRAC for the RHR piping in an MI-GE2 containment is zero (see Table 2). Therefore, for MI-GE2 plants, the RUPTPROB is determined only for the RCS piping pipe whip or jet impingement on the CRD piping.

A break in the RCS Pump 15 discharge pipe at the downstream elbow in MI-GE2 plants has the potential to impact the CRD piping bundles (see Figure 3). Pipe breaks elsewhere in the RCS piping will not impact the CRD piping bundles.

A break in the 28-inch diameter RCS discharge piping at the RCS Pump 15 discharge elbow will propel the RCS pipe toward the center CRD piping bundle and the reactor pedestal (BWR MI-GE2). The vertical section of the RCS pipe will have to travel approximately 9 inches before it can strike the floor I-beam (approximately 11 inches wide x 18 inches high) located approximately 5 feet above the pipe break location. If the vertical section of the RCS pipe manages to go through the I-beam, it will have to travel an additional distance (approximately 15 inches), before it can strike the center CRD piping bundle located 18 feet above the break location.

The vertical section of the RCS pipe will fail before it deflects 25 inches at 18 feet above the break location and impacts the CRD bundle. (A plastic hinge will form in the vertical section of the RCS pipe below the CRD bundle as a result of the pipe whip force.) Nonetheless, this technical assessment included a finite element structural analysis, using the ANSYS computer code, to determine the behavior of the CRD piping in the unlikely event it is impacted by the RCS pipe. This analysis conservatively assumed that the RCS piping fails the I-beam on impact and then hits the CRD piping. Figure 4 illustrates the details of the finite element model, which focused on a single 1-inch diameter CRD pipe placed near the tip of the elbow of the 28-inch diameter RCS pipe, with a small blowdown force of 1,000 pounds (1.0 Kip) applied to the RCS pipe. The 1-inch diameter CRD pipe is very flexible (compared to the 28-inch diameter pipe) and will bend without any significant crushing or crimping, as shown in Figures 5 and 6, before it ruptures or forms a guillotine break.

This CRD piping deformation as shown in Figures 5 and 6 is consistent with the behavior observed in impact tests on 1-inch diameter pipe, as described in NUREG/CR-3231, "Pipe-to-Pipe Impact Program," dated May 1987 (Ref. 3). Bending (rather than local crushing) was the prevalent mode of deformation for small diameter piping. A single plastic hinge does not develop in the pipe. Instead, the pipe is plastically deformed a short distance from the tip all the way to the end. The NUREG/CR-3231 also provides the details of the impact tests on pipes with different wall thickness and diameter. The test results indicate that the local crushing of the pipes greater than 4-inch diameter will not reduce the pipe diameter by more than 55 percent, and cross sectional area by more than 25% before break/failure.

According to Section 3.6.III.2.c(4) of the NRC's Standard Review Plan (SRP), the RCS jet pipe blowdown force resulting from the pipe break can be represented by a steady-state function; $T=CP_oA$, where T is the blowdown force, P_o is the system pressure before the pipe break, A is the pipe break area, and C is the blowdown force coefficient. According to SRP Section 3.6.2, the value of C should not be less than 2.0 for subcooled water. However, to account for variation in piping layout and configuration, this assessment considered values of C ranging from 1.3 to 2.2 for RCS piping. The RCS system pressure is 1,050 psi, and the pipe break area for a pipe with an outside diameter of 28 inches and a wall thickness of 1.50 inches, is 491 square inches. Therefore, the postulated RCS blowdown force can vary between 638 and 1,133 Kips.
A separate calculation determined that a CRD bundle of 70 1-inch diameter pipes will fail/break at about 42 Kips, or 6% of the lower bound value of the blowdown force (638 Kips), during an RCS break if the initial gap between the RCS pipe and CRD bundle is 6 inches. For gaps larger than 6 inches, substantially less blowdown force will be needed to fail/break the CRD piping bundle.

Given these results, it is clear that the CRD pipes will not crimp or crush at the point of impact, in the unlikely event that the RCS piping fails the I-beam and then impacts the CRD pipe bundle following a break at the RCS pump downstream elbow. However, the CRD pipes may bend before they rupture or form a guillotine break. The CRD rods would then scram on reactor pressure and would not result in a large LOCA that could lead to core damage. Therefore, the probability of RCS pipe whip or jet impingement causing CRD system failure (RUPTPROB) can conservatively be taken as 0.10. The maximum point estimate CDF for this event in a BWR MI-GE2 reactor, as determined in Table 3, is 5.3 x 10^{-7} event/RY.

6.2 Mark I-GE3/4 Containment

A review of the plant drawings and pipe break analysis data for MI-GE3/GE4 plants indicates that breaks in the RHR/RCS pipes at the following locations can impact the CRD pipe bundle:

- RHR pipe break downstream of the isolation valve near RCS discharge line
- RCS pump discharge pipe break downstream of the isolation valve

The following subsections discuss the RUPTPROB for these two events.

6.2.1 RHR Pipe Break Downstream of the Isolation Valve Near the RCS Discharge Line

This event involves a rupture of the RHR return pipe (with a nominal diameter of 24 inches) downstream of the isolation valve, near the RCS discharge line and the elbow just upstream of the RHR check valve (see Figure 7). The check valve is the border between “high-energy” and “medium-energy” piping systems. A break at this location would propel the RHR pipe outward toward the containment (BWR MII-GE4), and would not be expected to cause any damage to the CRD piping bundles located on both sides of the pipe at a 22E angle from the center line of pipe. A circumferential break in the pipe would not move the RHR pipe toward the CRD piping; pipe whip restraints both upstream and downstream of the pipe break would prevent damage to the CRD piping bundles. Thus, the postulated impact of the ruptured RHR pipe and the CRD bundles is improbable, and the probability of RHR pipe whip or jet impingement causing CRD system failure (RUPTPROB) can conservatively be taken as 0.10. The maximum point estimate CDF for this event in a BWR MI-GE3/GE4 reactor, as determined in Table 4, is 2.5 x 10^{-7} event/RY.

6.2.2 RCS Pump Discharge Pipe Break Downstream of the Isolation Valve

This event involves a rupture of the RCS pump discharge pipe (with a nominal diameter of 28 inches) downstream of the isolation valve. If a pipe whip restraint is installed near the break, it would keep the lower end of the RCS pipe from impacting the CRD piping bundles. However, if the pipe restraint is not installed on the lower end of the piping run, the RCS pipe would be free to rotate from the downstream elbow. In that instance, it would first impact a floor beam located approximately 5 feet above the break point. If that beam failed on impact, the RCS pipe would then propel toward the CRD bundle, which is located approximately 12.5 feet above the break point, as shown in Figure 8.

This assessment included a finite element analysis, using the ANSYS computer code, to determine the deflection of the RCS pipe resulting from the pipe whip load. To ensure conservative results, the analysis assumed that the RCS pipe penetrates the floor beam without loss of energy, and the pipe whip restraint is not installed on the RCS pipe. In addition, the analysis used an upper bound value of the blowdown force (1,133 Kips) for the RCS pipe break. The results of the analysis, as shown in Figure 8, indicate that even with the conservative assumptions, the RCS pipe will not impact the CRD bundle. A jet force would be experienced from the upstream and downstream sides of the RCS break, but would not impact the CRD pipe bundle since the bundle is located outside the jet zone of influence. In addition, as described in Section 6.1, even if the RCS pipe is postulated to impact the CRD bundle, it would not crimp or crush the CRD pipes at the point of impact. The CRD pipes may bend before they rupture or form a guillotine break. The CRD rods would scram on reactor pressure and would not result in a large LOCA that could lead to core damage. Therefore, the probability of RCS pipe whip or jet impingement causing CRD system failure (RUPTPROB) can conservatively be taken as 0.10. The maximum point estimate CDF for this event in a BWR MI-GE3/GE4 reactor, as determined in Table 5, is 7.64×10^{-7} event/RY.

6.3 Mark II-GE4 Containment

A review of plant drawings and pipe break analysis data indicates that MII-GE4 plants are subject to the same RHR/RCS pipe breaks that can impact the CRD pipe bundles in MI-GE3/4 plants. Specifically, those pipe break locations are as follows:

- RHR pipe break downstream of the isolation valve near the RCS discharge line
- RCS pump discharge pipe break downstream of the isolation valve

The following subsections discuss the RUPTPROB for these two events.

6.3.1 RHR Pipe Break Downstream of the Isolation Valve Near the RCS Discharge Line

This event is identical to the one described for MI-GE3/4 plants in Section 6.2.1. However, since the PIPETYPE and TYPEFRAC are different for MII-GE4 and MI-GE4 containments (see Table 2), the maximum point estimate CDF for this event, as determined in Table 6, is 5.27×10^{-7} event/RY.

6.3.2 RCS Pump Discharge Pipe Break Downstream of the Isolation Valve

This event is similar to the one described for MI-GE3/4 in Section 6.2.2. Therefore, the results of the analyses described in Section 6.2.2 also apply to this event. However, since the PIPETYPE and TYPEFRAC are different for MII-GE4 and MI-GE4 containments (see Table 2), the maximum point estimate CDF for this event, as determined in Table 7 is 5.85×10^{-7} event/RY.

6.4 Mark II-GE5 Containment

A review of plant drawings and pipe break analysis data indicates that breaks in the RHR/RCS pipes at the following locations can impact the CRD pipe bundle:

- RHR discharge pipe break at RCS pipe connection
- RCS pump discharge pipe break downstream of the isolation valve
- RCS pump vertical discharge pipe break at the upper end

The following subsections discuss the RUPTPROB for these three events.

6.4.1 *RHR Discharge Pipe Break at the RCS Pipe Connection*

This event involves a rupture of the residual heat removal (RHR) system, where it joins the RCS pump discharge pipe, adjacent to the CRD piping bundles. The break is postulated to occur at the downstream side of the RHR elbow (with a nominal diameter of 12 inches), which would produce a jet force that would accelerate the RHR pipe outward toward the containment in a BWR MII-GE5 design. This event is similar to those described for GE3 and GE4 plants in Sections 6.2.1 and 6.2.2, with the exception that the RHR pipe diameter in this instance is 12 inches instead of 24 inches. In addition, a pipe whip restraint exists immediately upstream of an isolation valve at approximately 7 feet below the break location. This pipe whip restraint would prevent the pipe from moving in the horizontal plane.

After exiting the reactor pedestal, the one CRD piping bundle runs parallel and on either side of the RHR piping and proceeds upward at an angle of approximately 74E. A horizontal gap of 18 inches separates the 12-inch diameter pipe and the CRD bundles. Thus, to impact the CRD piping bundles, the RHR pipe would have to move 18 inches at a 90E angle (perpendicular) to the direction of the jet force coming from the RCS, which is improbable. In addition, a jet force would be experienced from the downstream side of the isolation valve (as a result of reactor blowdown), which would not be expected to damage the CRD piping bundles because they are located outside the jet zone of influence. Therefore, the probability of RHR pipe whip or jet impingement causing CRD system failure (RUPTPROB) can conservatively be taken as 0.10. The maximum CDF for this event in a BWR MII-GE5 reactor, as determined in Table 8, is 5.93 x 10^{-7} event/RY.

6.4.2 RCS Pump Discharge Pipe Break Downstream of the Isolation Valve

This event involves a rupture of the RCS pump discharge pipe (with a nominal diameter of 24 inches) downstream of the second isolation valve, as shown in Figure 9. This event is only identified in the updated final safety analysis report (UFSAR) for Unit 2 of the Nine Mile Point Nuclear Station. The other three MII-GE5 plants (Columbia Generating Station and Units 1 and 2 of LaSalle County Station) do not postulate a break in the discharge pipe downstream of the second isolation valve, even though GE designed the piping layout, and pipe whip restraint for all four plants. (All four plants have the same number and locations of pipe whip restraints.)

A pipe whip restraint immediately following the second isolation valve on the vertical run of pipe prevents the pipe from moving radially outward away from the containment and may keep the lower end of the RCS pipe from impacting the CRD piping bundles. If this restraint failed, the vertical leg of the RCS pipe would have to deflect approximately 24 inches before it would impact the concrete wall of the reactor biological shield and (possibly) the CRD bundle.

In the unlikely event that the vertical leg of the RCS pipe impacts the CRD bundle, the CRD pipes would bend (without any significant crushing or crimping), as described in Section 2.4.1 and illustrated in Figures 5 and 6, before rupturing or forming a guillotine break. The CRD rods would scram on reactor pressure and would not result in a large LOCA that could lead to core damage. Therefore, the probability of RCS pipe whip or jet impingement causing CRD system failure (RUPTPROB) can conservatively be taken as 0.10. The maximum point estimate CDF for this event in a BWR MII-GE5 reactor, as determined in Table 9, is 5.3 x 10^{-7} event/RY.

6.4.3 RCS Pump Vertical Discharge Pipe Break at the Upper End

This event involves a rupture of the RCS pump discharge pipe (with a nominal diameter of 24 inches). This event is similar to the break described in Section 6.4.2, with the exception that this break would occur at the upper end of the vertical pipe run downstream of the RCS pump, just below the horizontal RCS header. The pipe whip restraint on the vertical leg of the pipe would stop pipe movement in the horizontal direction, but the vertical piping run could still move (but not much) in the vertical direction following a pipe break. The pipe whip restraints on the

RCS circular header would prevent vertical movement of the pipe, as would the RHR pipe (with a nominal diameter of 12 inches) connected to the RCS pipe near the pipe break point. In addition, the RHR pipe has a pipe restraint near the RHR/RCS interface. A jet force would be experienced from the upstream and downstream sides of the break, but would not be expected to damage the CRD piping bundles. The postulated impact of the ruptured RCS pipe and the CRD bundles is improbable.

7. Probabilistic Analysis

The four factors, IE, PIPETYPE, TYPEFRAC, and RUPTPROB, were used to form a simple event tree. Statistical distributions for the four factors were set up as follows:

Initiating Event Frequency (IE): The "classic" estimate of 10^{-4} large pipe break event/RY, which is the estimate used in both WASH-1400 and NUREG-1150, was chosen for this analysis. (The effect of using newer and lower estimates will be explored in the discussion section below.) As in the NUREG-1150 PRAs, a log-normal distribution was used, with an error factor of 10.

PIPETYPE: A normal distribution was used for this parameter, centered on the "engineering lower limit" value for each piping system (RHR or RCS), BWR product line, and containment design, with the standard deviation adjusted such that the 95th percentile matched the "engineering upper limit" value of 1.30 times the "engineering lower limit" value.

TYPEFRAC: Similarly, a normal distribution was used for this parameter, centered on the "engineering lower limit" value for each piping system (RHR or RCS), BWR product line, and containment design, with the standard deviation adjusted such that the 95th percentile matched the "engineering upper limit" value of 1.50 times the "engineering lower limit" value.

RUPTPROB: This parameter is more unusual. The most probable value, according to the engineering analysis, as calculated for both containment designs, all four BWR product lines, and for both RHR and RCS piping, is zero. Nevertheless, the engineering analysis estimated an "upper limit" of 0.1 for this parameter. To address this set of assumptions, an exponential distribution was used, with the exponential parameter λ set to 0.1. The resulting distribution has a most probable value at zero, but a mean value of 0.1.

CDF: The event tree was quantified as described above, and a Monte Carlo analysis of 10,000 samples was used to form a distribution for the CDF for each piping system, and each containment design/BWR product line configuration. The results are summarized as follows:

Containment	Product line	Sequence	Point Estimate*	Mean	Median	5th percentile	95th percentile
Mark I	GE2	RCS	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
	GE3&4	RCS	3.9E-7	4.0E-7	8.6E-8	2.6E-9	1.6E-6
		RHR	1.3E-7	1.3E-7	2.3E-8	8.6E-10	5.4E-7
		total	5.2E-7	5.3E-7	1.2E-7	4.0E-9	2.1E-6
Mark II	GE4	RCS	3.0E-8	5.1E-8	8.3E-9	1.7E-10	2.0E-7
		RHR	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
		total	3.0E-7	3.3E-7	7.2E-8	2.3E-9	1.3E-6
	GE5	RCS	2.7E-7	2.8E-7	5.9E-8	1.1E-9	1.1E-6
		RHR	3.0E-7	3.1E-7	6.6E-8	2.1E-9	1.3E-6
		total	5.8E-7	5.9E-7	1.3E-7	4.6E-9	2.4E-6
*	The point estimate values are based on the NUREG-1150 initiating event frequencies, the calculated value for PIPETYPE and TYPEFRAC, and the value for RUPTPROB						

The figures in the table above are in units of core damage events/RY. As the table parameters themselves illustrate, the number of significant figures does not imply accuracy to this degree, but instead are provided as an aid in following the calculation.

MD 6.4, Appendix C, Figure C5, gives the criteria for continuing a generic issue based on CDF. According to this figure, a generic issue with an associated mean ΔCDF below 10^{-6} event/RY can be excluded from further consideration, unless another parameter indicates otherwise. None of these CDFs rise to this threshold.

Large Early Release Frequency (LERF). On average, the pressure suppression containments used in BWRs are more likely than PWR-style large-volume containments to have early structural failures in the event of a severe core damage accident. Because of this, BWR PRAs often report LERFs that are a significant fraction of their CDFs.

However, the accident sequence of interest here is by definition a mitigated LOCA, where the core is successfully reflooded but a fraction of the control rods do not re-insert, causing a localized criticality. There may be some localized fuel melting, although a severe reactivity transient is not really likely. There will be considerable voiding in the coolant, which greatly reduces rod reactivity worth, and the reflooding (which takes on the order of 40 to 50 seconds) will not be rapid compared to, say, a control rod drop event. The rate of reactivity addition is likely to be slow enough to allow some moderator temperature and voiding feedback to help limit any the reactivity excursion.

The most likely cause of containment failure would be from overpressurization, since the various systems available for containment heat removal may not be capable of coping with both decay heat and the extra heat load resulting from continued fission energy production in the localized area. The usual early containment failures (overpressure from core-wide ATWS, fuel coolant interactions, and direct impingement of core debris on the containment boundary) are not likely to result from a localized criticality within a reflooded core. A failure of the containment to isolate would be possible, but this mechanism usually has a probability of 1% or less. Thus, the LERFs associated with this generic issue are not likely to exceed the 10^{-7} event/RY threshold for continuance.

Public Risk: To estimate public risk, a calculation was performed for a GE4 in a Mark I containment (specifically, based on the NUREG-1150 Level II and Level III analyses for the Peach Bottom plant), but using the generic site for generic issue analysis. This plant and containment configuration is the most common, and has the second highest core damage frequency, 5.3×10^{-7} event/RY. (The highest CDF, which is the GE5 in a Mark II containment, is 5.9×10^{-7} event/RY - only about 11% higher.)

The course of the event begins with the reflooding of the core, with a certain fraction of the control rods remaining withdrawn, causing a return to criticality in a localized area. Depending on the rate of reflooding, severe fuel damage may occur in the fuel assemblies in the area of the withdrawn control rods. Unless the containment fails to isolate, containment failure is unlikely at this point.

After the initial reflooding of the core, cooling will be provided by the RHR system operating in either LPCI mode (where the RHR takes suction from the suppression pool, pumps it through the RHR heat exchangers and then into the reactor, with the excess coolant eventually draining through the break and back to the suppression pool), or containment cooling mode (where the coolant is directed either back to the suppression pool or to the drywell or wetwell spray headers, after leaving the RHR heat exchangers). Depending on the amount of extra heat generation being contributed by the localized criticality, the RHR heat exchangers may not have sufficient capacity to accommodate the extra heat load, and the suppression pool temperature will increase.

The suppression pool is designed to absorb the energy of the blowdown from the pipe break and remain under 170E F, but nevertheless will already be at a significantly elevated temperature. If heatup continues due to insufficient RHR capacity, the suppression pool will eventually reach boiling. At some point, the RHR pumps will fail and, as coolant boils in the reactor and is not replenished, the core will experience core-wide severe damage. This will occur a significant time after the initial shutdown, and will reduce the radiological consequences. The containment, which will already be at an elevated pressure, may fail due to overpressure or due to a variety of other causes.

There is no plant damage state in the NUREG-1150 PRA that exactly matches this accident sequence. However, Plant Damage State (PDS) 7 is quite similar. PDS-7 corresponds to an ATWS event, initiated by a stuck-open safety/relief valve, with a failure of the SBLC. The course of the accident, including the suppression pool heatup, etc., is very similar, but PDS-7 does not include the possibility of severe fuel damage in a small portion of the core at the beginning of the event. This should not have a significant on consequences to the public, since the containment is not likely to fail until much later, and fission products from failed fuel in the vicinity of the unscrammed control rods will be contained until the remainder of the core is damaged.

The accident sequences leading to PDS-7 also differ from that of this generic issue in that, in the case of PDS-7, the suppression pool is first heated to nearly boiling, and then the ATWS event occurs with the reactor still at high pressure. (Since this generic issue's sequences are initiated by a large break LOCA, core damage at high pressure are not expected.) Thus, use of the PDS-7 sequence may overestimate the radiological consequences. (This will not change any conclusions.)

Therefore, the radiological consequences were approximated by using PDS-7, i.e., the same allocation of zirconium oxidation level, containment failure mode, etc., but no vessel breach at high vessel pressure. The risk was calculated using the standard assumptions for generic issues, i.e., person-rem to a 50-mile radius using a uniform population density of 340 people per square mile, a half-mile exclusion radius, and a central midwest meteorology. The result (for a GE4 in a Mark I containment) was a mean risk of 0.89 person-rem/RY. This is well below the threshold of 100 person-rem/RY given in Appendix C of the Handbook for Management Directive 6.4, "Generic Issues Program."

As a check on this use of the LOCA-initiated plant damage state, the calculation was repeated, but this time using the plant damage state corresponding to a LOCA-initiated core damage accident (PDS-1). The result (again, for a GE4 in a Mark I containment) was a mean risk of 0.66 person-rem/RY. Again, this is well below the threshold of 100 person-rem/RY.

8. Discussion

Other estimates of Initiating Event Frequency: This analysis used the NUREG-1150 estimate of 10^{-4} pipe break/RY. Other, more modern estimates are all lower than this value. Thus, use of any of the other estimates would not change the conclusion.

Effect of Standby Coolant Supply: Every BWR has a means of injecting water from the ultimate heat sink into the vessel, usually by providing a means of lining up the RHR pump suctions to service water. This capability, which does require manual action on the part of the plant operators, was not credited in this analysis. However, this action could avert a full core meltdown even if the suppression pool were to overheat and fail.

9. Conclusion

This technical assessment describes a detailed analysis of the high-energy pipe break interactions documented in preliminary evaluations of BWR MI and M2 power plants for GSI-80 (Refs. 1 and 2). The CDFs for the various RHR and RCS pipe break events that could potentially impact CRD piping have been determined for MI and MII plants. All of the calculated CDF values are less than the threshold (10^{-6} event/RY) specified in the Handbook to MD 6.4. Therefore, GSI-80 will be closed with no changes to the existing regulations or guidance.

10. References

1. 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, November 1999.

2. Memorandum to S. Collins from A. Thadani, "Generic Issue 80, A Screening Evaluation of GSI-80, 'Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments,' February 14, 2003.

3. NUREG/CR-3231, "Pipe-to-Pipe Impact Program," U.S. Nuclear Regulatory Commission, May 1987.

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Table 1

Failure Frequencies (Events/Calendar Year) for the Large-Diameter BWR Plant Pipe

Source	Frequency (events/calendar year)
NRC Expert Elicitation Passive System LOCA (Draft 2004) for a pipe diameter of 18 inches or more	0.000002
NRC Expert Elicitation Passive System LOCA (Draft 2004) for pipe diameters between 7 and 18 inches	0.000007
NUREG/CR-5750 (1999)	0.00002
NUREG-1150 (1990)	0.0001
WASH-1400	0.0001

Table 2

PIPETYPE and TYPEFRAC by System and Facility Design

BWR Design	Total No. of Break Points on Large Diameter Piping Inside Containment	System (nominal pipe diameter, inches)	Break Points	PIPETYPE (System Fraction)	TYPEFRAC (CRD Impact)
Mark I-GE2 Oyster Creek	117	RHR (14)	8	0.07	0
		RCS	20	0.17	0.16
Mark I-GE4 Hope Creek	80	RHR (12, 20)	6	0.08	0.16
		RCS (28)	22	0.28	0.14
Mark II-GE4 Susquehanna	73	RHR (20, 24)	11	0.15	0.18
		RCS (28)	22	0.3	0.1
Mark II-GE5 Nine Mile Point 2	128	RHR (12, 20)	21	0.16	0.19
		RCS (28)	22	0.17	0.16

Table 3

Mark I-GE2 Plant CDF Due to the RCS Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.221*	0.17	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.24**	0.16	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)

* The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.

** The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.

Table 4

Mark I-GE4/GE3 Plant CDF Due to the RHR Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.104*	0.08	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.24**	0.16	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)

* The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.

** The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.

Table 5

Mark I-GE3/GE4 Plant CDF Due to the RCS Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.364*	0.28	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.21**	0.14	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)
*	The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.		
**	The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.		

Table 6

Mark II-GE4 Plant CDF Due to the RHR Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.195*	0.15	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.27**	0.18	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)
*	The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.		
**	The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.		

Table 7

Mark II/GE4 Plant CDF Due to the RCS Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.39*	0.3	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.15**	0.1	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)
*	The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.		
**	The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.		

Table 8

Mark II/GE5 Plant CDF Due to the RHR Pipe Impact on CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.208*	0.16	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.285**	0.19	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)
*	The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.		
**	The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.		

Table 9

Mark II-GE5 Plant CDF Due to the RCS Pump Discharge Pipe Impact on the CRD Bundle

Factor	Upper Bound Value	Calculated Value	Factor Description
PIPETYPE	0.221*	0.17	Fraction of large diameter BWR RCS piping inside the containment
TYPEFRAC	0.24**	0.16	Fraction of RCS piping that can impact the CRD bundle from a pipe whip
RUPTPROB	0.1	0	Probability of RCS pipe impact crimping more than seven CRDMs (14 pipes)
*	The Upper bound value of the PIPETYPE is assumed to be 1.30 times the calculated value determined in Table 2 to account for differences in plant layout.		
**	The Upper bound value of the TYPEFRAC is assumed to be 1.50 times the calculated value determined in Table 2 to account for differences in plant layout.		

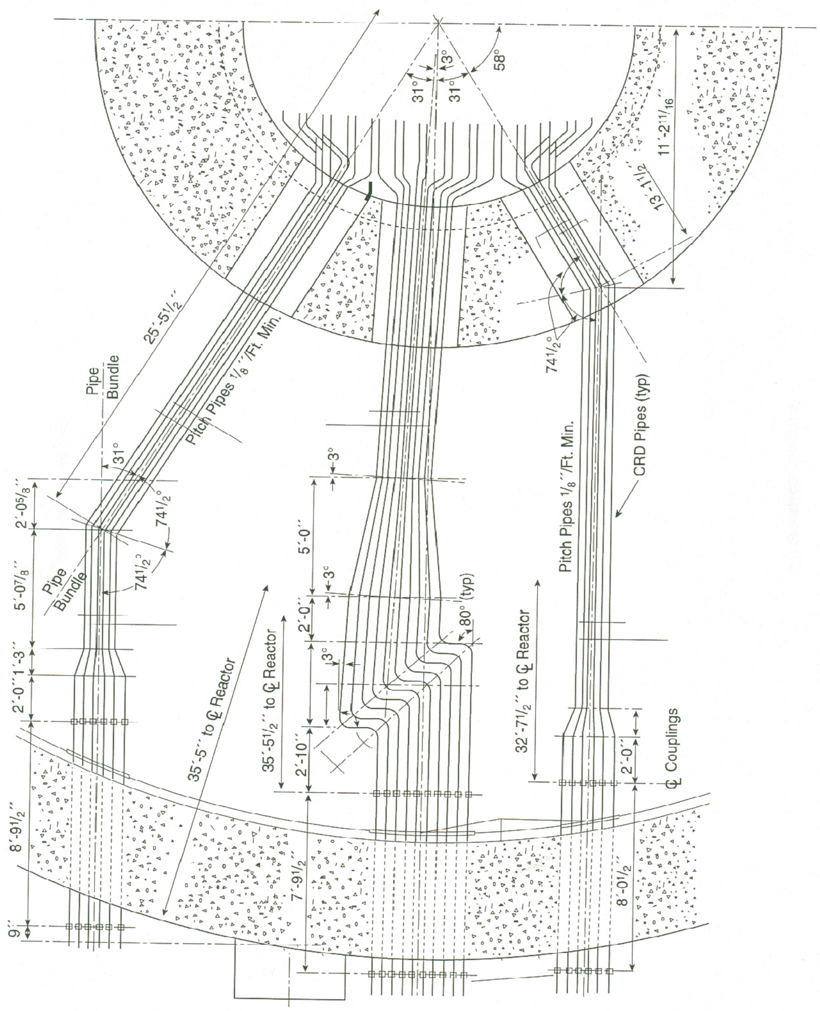

Figure 1 Plan view of the Control Rod Drive Bundles in MI-GE2 Containment

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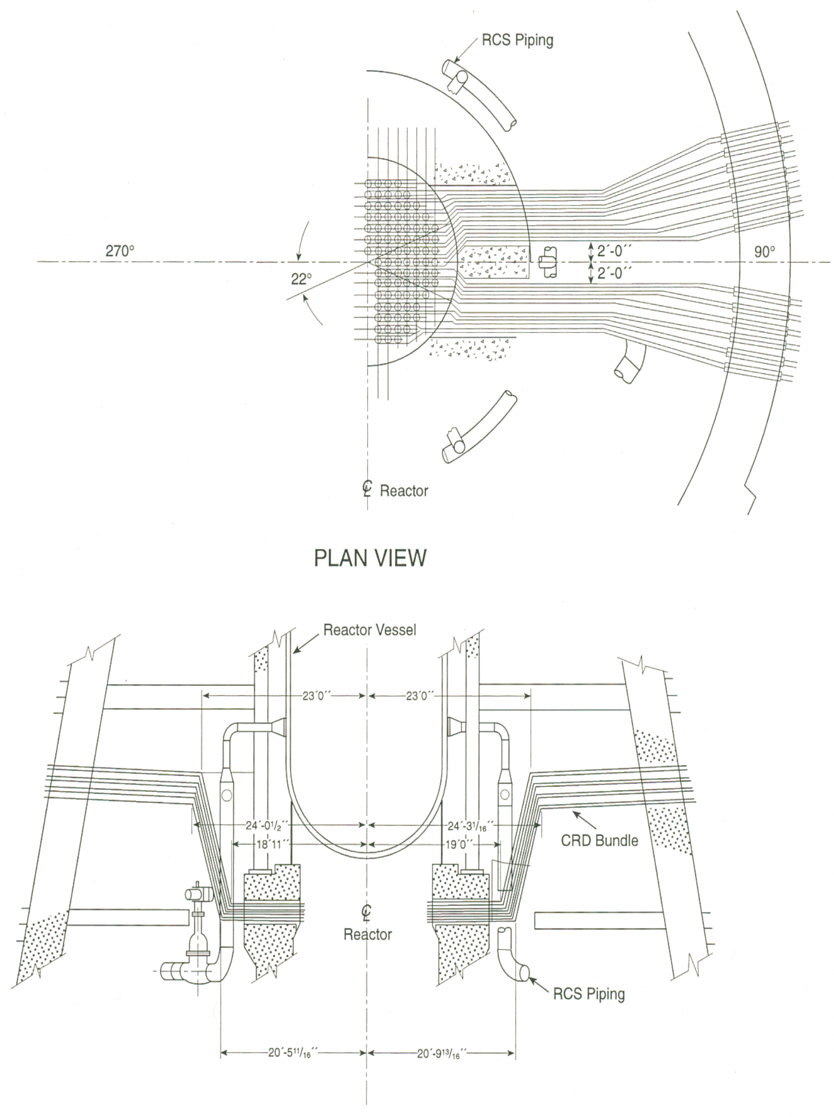

Figure 2 Control Rod Drive Bundle Layout in GE3/GE4/GE5 Mark I/II Containment

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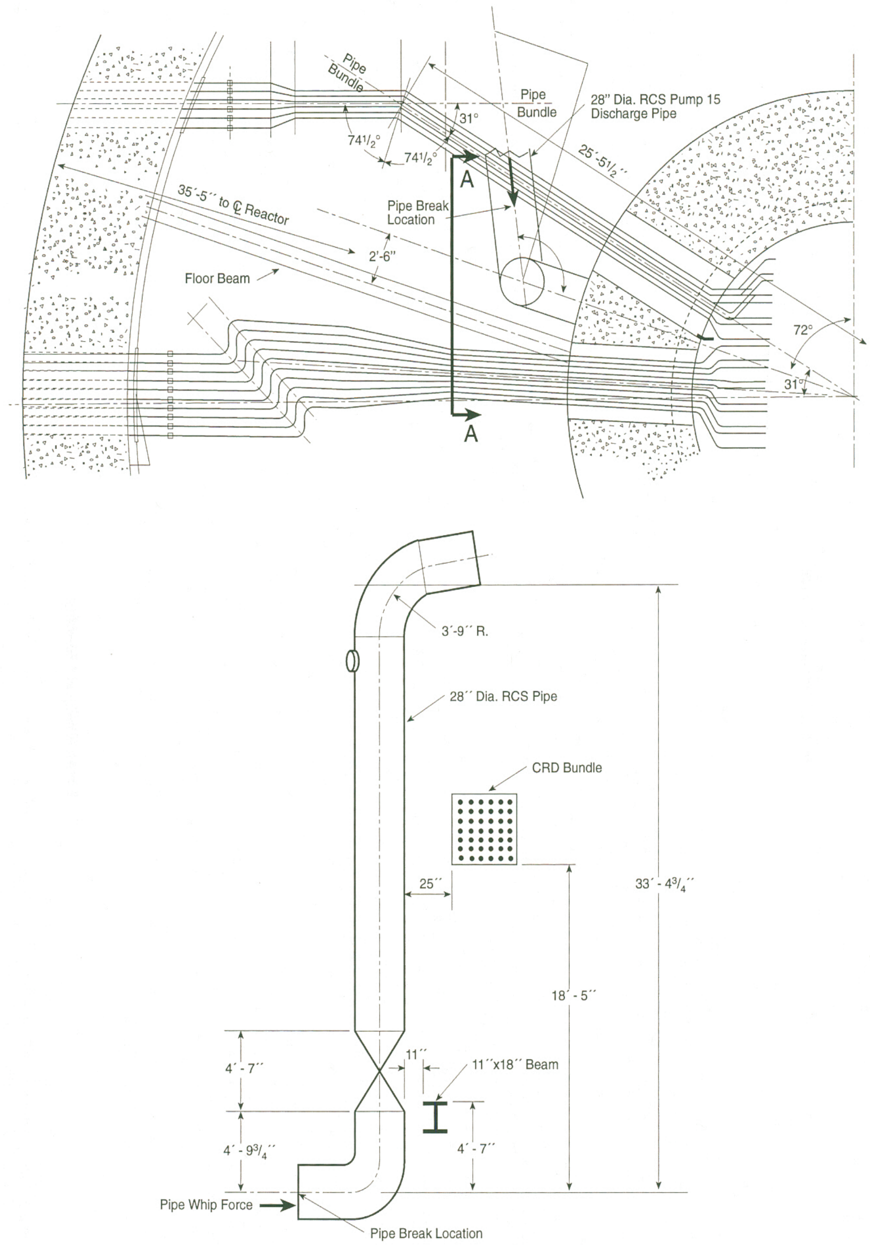

Section A-A

Figure 3 Break in RCS Pump 15 Discharge Pipe in MI-GE2 Plant

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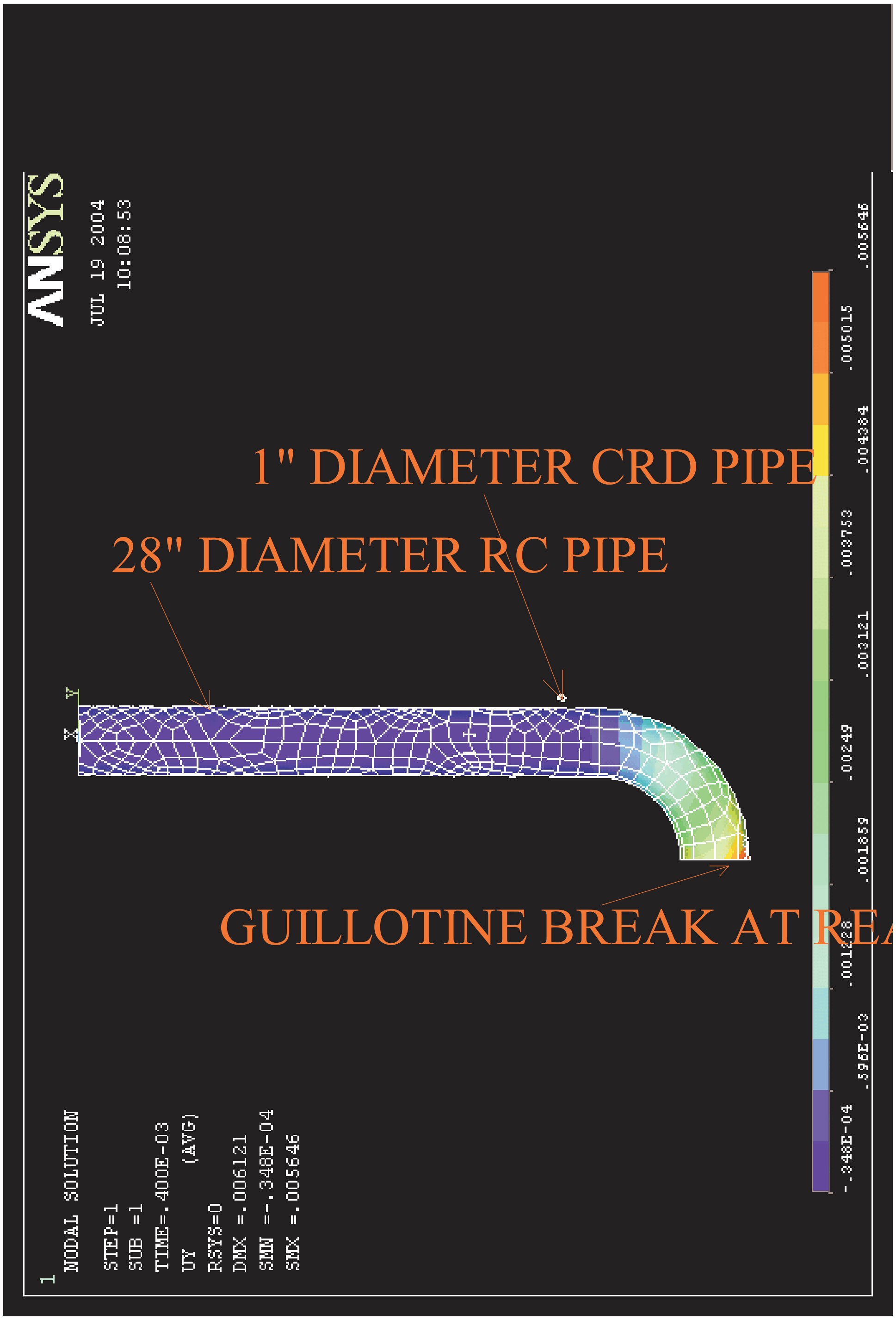


Figure 4. ANSYS Model of the RCS Pipe Impact on the CRD Pipe (Initial Condition)

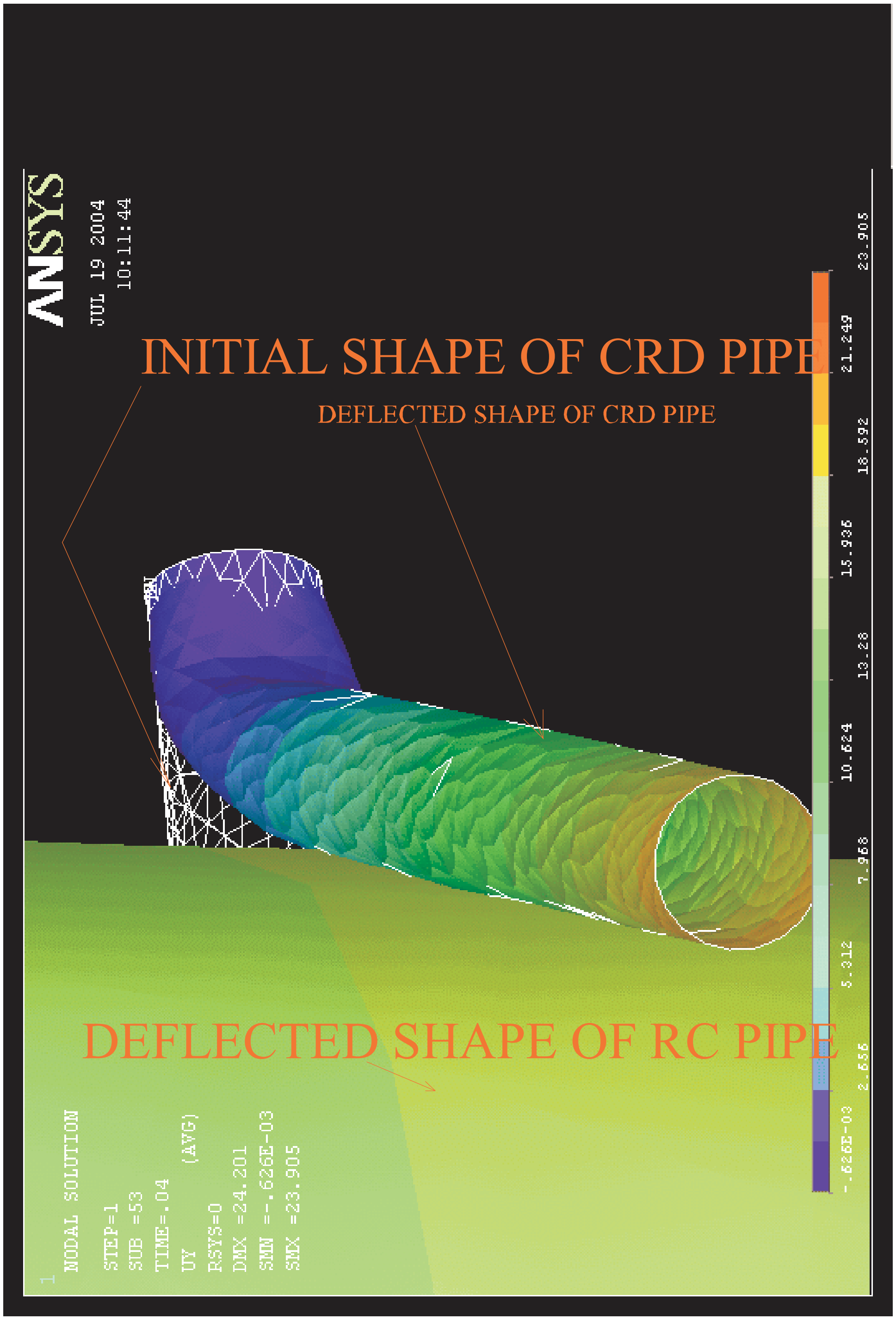


Figure 5. Deflected Shape of the CRD Pipe after the RCS Pipe Impact (View 1)

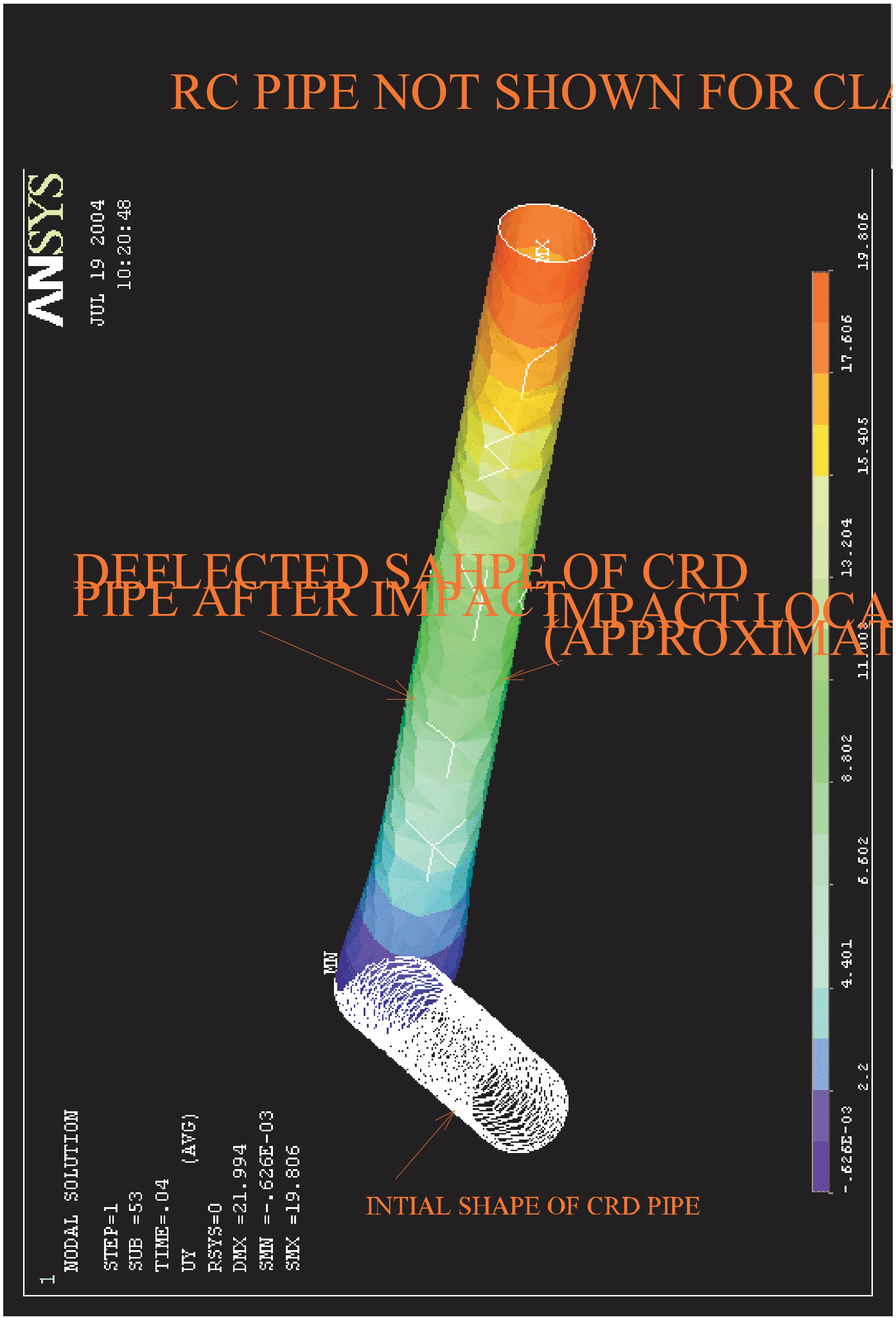
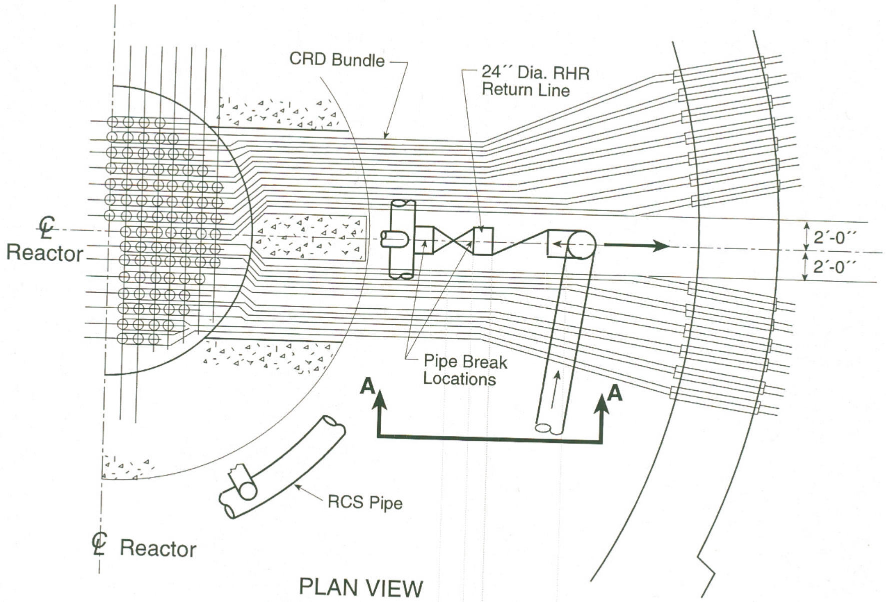
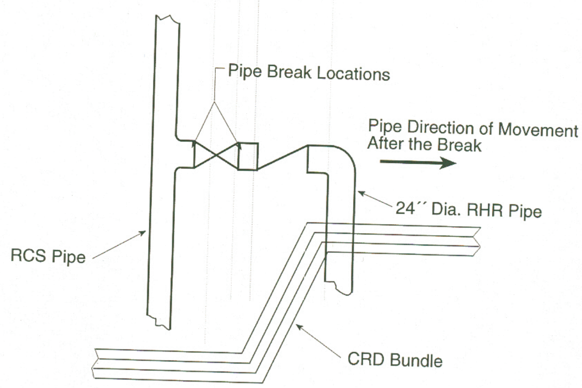


Figure 6. Deflected Shape of the CRD Pipe after the RCS Pipe Impact (View 2)





Section A-A

Figure 7 Mark I GE3/GE4 RHR Pipe Break Downstream of Isolation Valve

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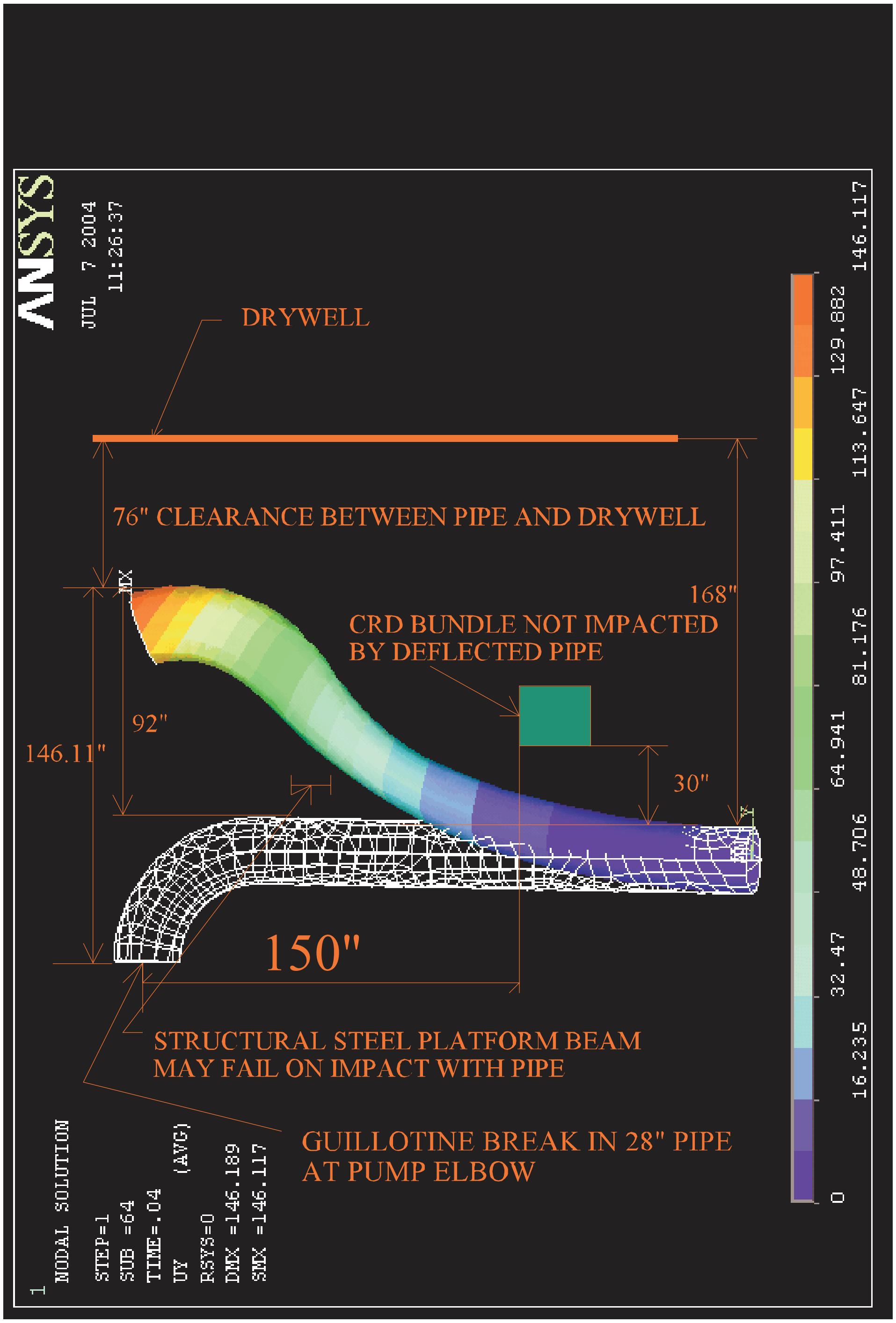
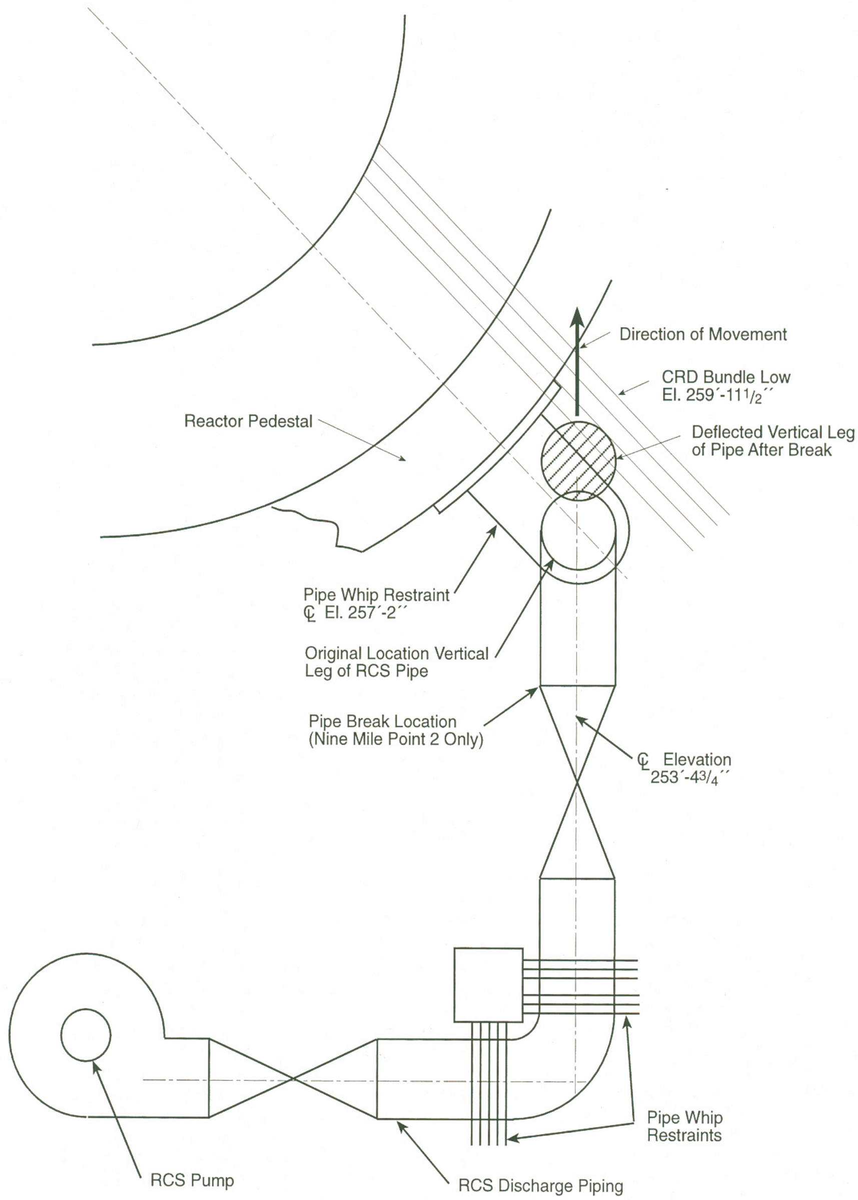


Figure 8. Deflected Shape of the RCS Pipe Due to Upper Bound Blowdown Force



RHR PUMP DISCHARGE PIPE BREAK DOWNSTREAM OF ISOLATION VALVE

IN THE MII-GE5 CONTAINMENT

FIGURE 9

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