May 1, 2001

MEMORANDUM TO: Ashok Thadani, Director
Office of Nuclear Regulatory Research

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

SUBJECT: TASK ACTION PLAN FOR GENERIC ISSUE 156.6.1, "PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT"

Attached for your information and approval is the proposed final task action plan (TAP) for the resolution of Generic Safety Issue (GSI) 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment." The proposed final plan is based on comments and suggestions that were provided at a December 19, 2000, briefing on the draft TAP and informal and formal follow-on comments from each of the RES technical divisions and technical staff in the Office of Nuclear Reactor Regulation (NRR). The revised plan reduces the estimated implementation period from about 42 months to about 18 months from initiation. It also reduces the estimated overall required staff resources from about 2.5 FTE to .65 FTE. We believe that the revised evaluation approach provides for a more effective, efficient and timely resolution to the GSI while limiting the impact on potentially affected licensees. However, it is projected that $300K in contractor support would still be required to ensure that there is adequate realism of the most risk-significant HELB interactions that will be considered in the TAP.

The BWROG, NEI and one PWR licensee plan to provide peer review comments on the prioritization study and analysis. These comments are expected by the end of FY 2001. The plan calls for utilizing these comments to the extent practicable in the technical evaluation. Additionally, a meeting, open to the public, will be scheduled in order to obtain additional stakeholder comments on the prioritization and TAP for GSI 156.6.1. These additional input will also be utilized, as appropriate, in the implementation of the TAP.

It should be noted that the proposed final TAP is very comprehensive. It includes a detailed description of the contingency action steps that would need be taken <u>only</u> if the preliminary risk significance screening steps in the early phase of the plan found that a more detailed analysis was needed. By inspection of the events that are considered in this plan, this is <u>not</u> expected, except for a few of the limiting events hypothesized for the prioritization analysis. Even so, for these few hypothesized events, we fully expect to either: (1) obtain additional voluntary information from industry to reduce the excessive conservatism, or (2) utilize contractor support involving additional effects analysis to reduce the excessive conservatism. Accordingly, it is anticipated that there will <u>not</u> be a need to conduct any site visits to collect additional plant-specific information as may be inferred from the comprehensive nature (i.e., contingency steps) of the plan.

At this time sufficient resources have not been identified to implement the plan. Some limited FTE resources have been identified to begin implementing some of the tasks in the TAP during FY 2001. However, in the absence of FTE and contractor support resources needed to fully implement the TAP, especially the most critical tasks, it is recommended that implementation of the TAP be deferred until adequate resources can be identified. If approved, the attached TAP would complete both milestones in the RES FY 2001 operating plan for GSI 156.6.1.

Please contact me or Stuart Rubin (SDR1), 415-7480, if you have any questions or require additional information.

Attachment: As stated

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Please contact me or Stuart Rubin (SDR1), 415-7480, if you have any questions or require additional information.

Attachment: As stated

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TASK ACTION PLAN
PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS INSIDE CONTAINMENT
(GENERIC SAFETY ISSUE 156.6.1)
(April 2001)

Lead Organization:	Division of Systems Analysis and Regulatory Effectiveness (DSARE)/RES
Task Manager:	Stuart Rubin, RES/DSARE
Lead Manager:	Farouk Eltawila, RES/DSARE
NRC Principal Reviewers:	Sunil Weerakkody, RES/DRAA/OERAB John Ridgely, RES/DRAA/PRAB TBD, RES/DET/MEB
Applicability:	Systematic Evaluation Program III Plants (41 LWRs involving licensing reviews completed before November 1975)
Projected Completion Date:	Approximately 18 months from initiation[1]

[1]Once resources (FTE and contractor support) are available.

1. PROBLEM DESCRIPTION

In 1967, the Atomic Energy Commission (AEC) published the General Design Criteria (GDC) for comment and interim use. The AEC staff's implementation of the GDC required consideration of postulated pipe break effects inside containment. However, due to the lack of formal documented review criteria, AEC staff review positions were continually evolving. Review uniformity was initially developed in the early 1970s with the issuance of a *Draft Safety Guide* entitled "Protection Against Pipe Whip Inside Containment." This draft document represented one of the first to formally compile the deterministic criteria that the AEC staff had been using to varying degrees for several years 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. This *Draft Safety Guide* was subsequently revised and issued in May 1973 as Regulatory Guide 1.46 (RG 1.46). The AEC implemented RG 1.46 only on a forward-fit basis.

In November 1975, the NRC staff issued *Standard Review Plan* (SRP) Sections 3.6.1 and 3.6.2 which considered the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc. on structures, systems and components (SSCs) relied on for safe reactor shutdown. After 1975, the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc. on these SSCs were considered in the staff's reviews.

In 1977, the NRC initiated the Systematic Evaluation Program (SEP). The purpose of the SEP was to review the designs of the 51 operating nuclear power plants that had been reviewed by the staff before the SRP was issued. The SEP was divided into two phases. In Phase I, the staff defined the technical areas for which regulatory requirements had changed enough over time to warrant a re-evaluation by the staff. One of these areas was Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems and Components Inside Containment."

In Phase II, the staff compared the design of 10 of the 51 older operating plants to the SRP. The 10 plants are referred to as "SEP-II" plants. Based on these reviews, the staff identified a total of 27 technical areas that required some corrective action at one or more of the 10 SEP-II plants that were reviewed. The staff concluded that these technical areas generally applied to the remaining 41 operating plants (referred to as "SEP-III" plants) that had also received their operating licenses before the SRP was issued. One of these areas was Topic III-5.A.

The staff subsequently undertook an effort to determine whether the 27 technical areas either had been or was being addressed by other regulatory programs and activities. Each of the 27 areas were placed in one of four 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 low safety significance and required no further regulatory action; (3) issues that were unresolved but existing regulatory programs had been identified that covered the scope of the technical concerns and implementation would resolve the specific issue (e.g., 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.

Twenty-two of the technical areas were categorized as either Category 3 or Category 4. The effects of high energy line breaks on structures, systems and components inside containment was one of three issues categorized as Category 4. Accordingly, Generic Safety Issue (GSI) 156, “Systematic Evaluation Program” was established to resolve these 22 issues. The effects of high energy line breaks on structures, systems and components inside containment was designated as GSI 156.6.1. In August 1999, work on an enhanced prioritization for GSI 156.6.1 was completed. The enhanced prioritization recommended a high priority for resolution of GSI 156.6.1.

2. PLAN FOR PROBLEM RESOLUTION

The plan for the resolution of GSI 156.6.1 involves the conduct of a technical evaluation and possible regulatory analysis of HELBs inside containment for the SEP-III plants. The technical evaluation as a minimum will consist of a safety goal evaluation as described in NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission.” The safety goal evaluation will utilize probabilistic risk assessment (PRA) methods to quantify the potential risk reduction associated with addressing postulated unacceptable HELBs inside containment for the SEP-III plants. The risk reduction is that which might be expected if the 41 SEP-III plant licensees re-evaluated HELBs inside containment in accordance with the deterministic (i.e., traditional engineering analysis) SRP guidance and implement corrective actions for those interactions that were found to not meet applicable deterministic acceptance criteria. The applicable deterministic acceptance criteria involve the ability to achieve safe shutdown of the plant, maintain containment integrity and maintain a coolable geometry within the reactor core. If, and only if, the safety goal evaluation results are favorable would a regulatory analysis of values and impacts be conducted

The risk reduction will be based on the projected change in the estimated core damage frequency per reactor-year (ΔCDF/RY) for the SEP-III plants. The ΔCDF/RY for any and all postulated unacceptable inside containment HELB interactions that have been identified or hypothesized and documented as a result of a re-evaluation by or for any pre-SRP plant will be utilized in the evaluation. The sources of information for these unacceptable interactions will consist of scenarios previously found and documented. They involve three categories of documented information: (1) the SEP reviews for the 10 SEP-II plants; (2) SEP-II plant or SEP-III plant licensee re-reviews of postulated unacceptable inside containment HELB scenarios documented in licensee event reports (LERs) and; (3) the documented HELB scenarios hypothesized in connection with the enhanced prioritization for GSI 156.6.1. The unacceptable interactions in each of the three categories will be screened to identify the interactions that are potentially the most risk significant with respect to ΔCDF/RY. The safety goal evaluation will be based on the unacceptable interactions in the three categories that are potentially the most risk-significant. The risk significance of the unacceptable interactions will be calculated using realistic assumptions that are based on traditional engineering insights, information and analyses. The determination of the ΔCDF/RY for both the risk screening of postulated interactions and the PRA of the most risk significant interactions will utilize current estimates of initiating event frequencies that have been peer reviewed by both NRC and industry experts. Uncertainty analyses will also be conducted and will include an assessment of the variations in ΔCDF/RY attributable to differences in the importance of HELBs inside containment among the SEP-III plants. Sensitivity analyses will also be conducted to assess the effects of pipe break probability on the estimated

ΔCDF/RY. It should be noted that the postulated unacceptable HELB interactions identified in connection with the SEP II reviews or documented in LERs will not be reevaluated with respect to the specific corrective actions that were previously taken by the licensees and reviewed by the staff.

As stated earlier, 51 operating nuclear power plants were reviewed and licensed before the SRP was issued. The 41 plants within the scope of GSI 156.6.1 (i.e., the SEP-III plants) is a subset of the 51 plants, with initial license review dates after the SEP-II plants. Within a specific LWR class and NSSS vendor type, the basic design features of the SEP-II and SEP-III plants are similar. The estimated risk significance of any inside containment unacceptable HELB interaction effects that might be identified as a result of a comprehensive re-evaluation of the SEP-III plants (using post-SRP methods and acceptance criteria) will be inferred from the estimated risk significance of the most risk significant unacceptable interactions that have been identified to date for the 51 plants using post-SRP methods and acceptance criteria and the most risk significant of those that have been hypothesized. There are two sources of known unacceptable interactions. These are the postulated unacceptable interactions that were either found, addressed and documented for the SEP-II plants as a result of the systematic evaluation program reviews or the postulated unacceptable interactions that were found and addressed by licensees of either SEP-II plants or SEP-III plants and documented in licensee event reports (LERs). The *hypothesized* unacceptable interactions are those that were selected for the enhanced prioritization of 156.6.1. The Task Action Plan (TAP) consists of estimating the risk of the most risk significant of the unacceptable interaction scenarios in each of the three groups.

Task 1 <u>Unacceptable HELB Interactions Identified by the SEP-II Plant Reviews</u>

Ten SEP-II plants were reviewed for the effects of HELBs inside containment by the systematic evaluation program: Palisades, R. E. Ginna, Oyster Creek, Dresden 2, Millstone 1, Yankee Rowe, Haddam Neck, LaCrosse, Big Rock Point and San Onofre1. Four of the ten plants found no unacceptable interactions as a result of the SEP reviews: Dresden 2, Millstone 1, Big Rock Point and San Onofre 1. Accordingly, only six plants will need additional information as input for the ΔCDF/RY analysis: Palisades, R. E. Ginna, Oyster Creek, Yankee Rowe, Haddam Neck and La Crosse. However, Yankee-Rowe, La Crosse, Haddam Neck, and San Onofre 1 are no longer licensed to operate. Thus for the tasks below, additional plant-based information, where identified and needed, will be pursued at only the (three) SEP-II plants which found unacceptable interactions and are still licensed to operate. These are:

Task 1.1 Collect and Review Documents Describing Postulated Unacceptable Interactions, Catalogue Characteristics

Obtain from INEEL copies of all reference documents that were collected on the unacceptable interactions identified for the SEP II plants for the enhanced prioritization of GSI 156.6.1. Using NUDOCS obtain copies of all documents on SEP Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems and Components Inside Containment" for each of the SEP-II plants. Review the INEEL and SEP documents to determine and document for each SEP plant the circumstances of each identified unacceptable interaction. For each such interaction, catalogue: the high energy system with the postulated break, the postulated break location, the piping internal pressure, the type of break (i.e., circumferential or longitudinal break), the type of postulated interaction (i.e., pipe whip or jetting interaction), the piping diameter and piping thickness, the piping material, the system structure or component (SSC) target, the SSC failure mode (e.g., fail high, low, as is), the effect on protection systems or mitigation systems (e.g., failure of RPS trip channels, failure of engineered safety feature automatic initiation system trip channels, loss of a containment boundary or function), the postulated single active random failure, the offsite power status, other postulated initial conditions and/or boundary conditions, operator actions (if any), the reactor system safety consequences (e.g., core damage) and, the corrective actions taken to reduce or eliminate the potential for the event.

Task 1.2 Determine the Pipe Break Probability for the Type of Piping Involved in the Postulated Break

Obtain and review the potential pipe failure rate databases that may be appropriate for determining the pipe break probability for the type of piping involved in the postulated unacceptable interactions. Select the reference pipe break probability database to be used. As appropriate, update the reference database to account for more recent failure rate data and failure causes. As appropriate, the task will also determine the pipe break probability to be used for the applicable type of piping based on the specific system application including the involved piping diameter.

Task 1.3 Screen Postulated Unacceptable Interactions for Change in Core Damage Frequency

Conduct a screening assessment of the change in core damage frequency (ΔCDF/RY) for each postulated unacceptable interaction for each of the SEP II plants. The screening assessment will utilize the risk screening assessment methods used by the Accident Sequence Precursor (ASP) Program for LERs for more detailed review as potential precursors. Using the ASP methods, screen out interactions which are not precursor candidates (i.e., ΔCDF/RY $\geq$ 1.0E-06). Unacceptable interaction events not screened out, (i.e., those which might satisfy the safety goal screening criteria for the value-impact portion of a regulatory analysis) if any, will be further evaluated using the Tasks 1.4 through 1.9.

Task 1.4 Obtain and Analyze Piping and SSC Target Spatial Layout Information

Obtain piping isometric drawings and information on piping-target physical layout arrangement including nearby piping elbows as possible hinge points for pipe whip or jetting. As needed, conduct conference calls and/or site visits for the SEP-II plants which are still licensed to operate: Palisades, R. E. Ginna, and Oyster Creek. Collect additional direct visual observation, if possible, and drawings on the physical layouts involved for the unacceptable interactions. For plants which are no longer operating, search for and obtain publically available and previously docketed piping drawing information for the piping/target spacial arrangements.

Task 1.5 Calculate Initiation Frequencies for the Postulated Unacceptable Interactions

Determine the mean initiation frequency (events/RY) of the postulated unacceptable interactions based on the mean pipe break probability for the type of piping involved (i.e., Task 1.3) and an analysis of the percent of piping length and fractional pipe whip or jetting arc placing the target at risk of failure (from Task 1.3). As needed, evaluate the piping internal reaction forces and local piping bending stresses to determine the expected elastic and/or plastic deflection modes of the piping relative to the SSC target. Calculate the mean interaction frequency per reactor year based on the appropriate mean pipe break probability data and the fraction of piping which has the potential to adversely interact based on the layout of the piping and the target the type of interaction involved, the fraction of three dimensional space which can result in an unacceptable interaction (i.e., the effective solid angle of piping/jetting that could cause adverse effects to the total solid angle involving the possible directions of pipe break effects (whip or jetting).

Task 1.6 Determine Reference Core Damage Frequencies

Determine the reference (base) plant-specific CDF/RY for breaks (anywhere) inside containment for the piping type (diameter) involved in the postulated unacceptable interaction. Utilize the applicable plant-specific HELB event tree SPAR models for use with the SAPHIRE PRA code. Use the base (reference) failure probabilities for the applicable protection and mitigation equipment important to each pipe break event tree. Repeat these plant-specific reference (base) CDF/RY assessments for each type of piping (i.e., small, medium, large diameter) that is applicable to each postulated unacceptable interaction.

Task 1.7 Determine Core Damage Frequency for Postulated Unacceptable Interactions

Using the mean initiation frequencies (from Task 1.5), determine the adjusted plant-specific estimated CDF/RY for the postulated unacceptable interactions. Utilize the applicable plant-specific PRA event trees from the SAPHIRE code. For each protection and mitigation system that fails due to either: (1) the broken pipe (e.g., low pressure safety injection system train failure due to a HELB in the injection piping for one of its trains) or (2) the HELB effects (e.g., whipping pipe or jetting-induced failure of one or more pressure sensing switches that are relied on for automatic initiation of the alternate train of low pressure safety injection), adjust its associated base (reference) failure probabilities to unity (i.e., "house event"). Repeat these plant-specific CDF/RY assessments for each postulated unacceptable HELB interaction.

Task 1.8 Determine the Change in Core Damage Frequency for Postulated Unacceptable Interaction Scenarios

For each postulated unacceptable interaction determine the change in core damage frequency (ΔCDF/RY) based on the difference between the postulated unacceptable interaction CDF/RY and the associated reference (base) plant-specific CDF/RY.

Task 1.9 Determine the Estimated Conditional Probability of Early Containment Failure

As necessary (for scenarios involving an estimated ΔCDF/RY $\geq$ 1E-6), determine the associated estimated conditional probability of early containment failure if the unacceptable interaction involves increased potential for (i.e., consequential) early containment failure due to the effect of the HELB.

Task 2 Unacceptable HELB Interactions Identified in Licensee Event Reports

Operating nuclear power plant licensees are required to report, pursuant to 10 CFR 50.73, conditions that result in the power plant, including its principal safety barriers, being seriously degraded or result in the plant being outside its design basis. Accordingly, if any of the 10 SEP-II plants or the 41 SEP-III plants found that any of the acceptance criteria for HELBs inside containment were not met, they are required to report these circumstances to the NRC. For the tasks below, LERs submitted by any of these plants since the SRP was issued which describe postulated unacceptable interaction scenarios involving HELBs inside containment will be collected and analyzed with respect to the change in core damage frequency as well as the estimated conditional probability of early containment failure.

Task 2.1 Find LERs Involving Inadequate Design-Analysis of HELBs Inside Containment

Use the Sequence Coding and Search System (SSCS) to conduct a comprehensive search of SEP-II and SEP-III plant LERs to find design basis issues involving inadequate design-analysis of HELBs inside containment reported since January 1976. Obtain a copy and review each LER.

Task 2.2 Screen Postulated Unacceptable Interactions for Change in Core Damage Frequency

Conduct a screening assessment of the change in core damage frequency (ΔCDF/RY) for the unacceptable interactions identified in the LERs involving inadequate design-analysis of the effects of HELBs inside containment. The screening assessment will utilize the results documented in NUREG/CR-4674 series which document the ASP program results for LERs from 1969 to date. Those LERs which are not identified in the NUREG as precursors to potential core damage accidents involve an estimated change in core damage frequency that is less than 1.0E-06. These events individually do not satisfy the safety goal screening criteria for the value-impact portion of a regulatory analysis. For LERs involving ASP events (and might satisfy the safety goal screening criteria for the value-impact portion of a regulatory analysis) document the estimated ΔCDF/RY. Review the documented PRA basis for the ASP analysis and obtain unpublished reference analysis information and data, that had been collected and compiled in connection with the ASP analysis. If feasible, discuss the PRA models and data used for the ASP study with the cognizant NRC contractor and NRC staff personnel. If (and only if) the ASP program results indicate an estimated of either ΔCDF/RY $\geq$ 1.0E-05 (and an estimated conditional containment failure probability of 1E-02 to 1E-01) or ΔCDF/RY $\geq$ 1.0E-06 (and an estimated conditional containment failure probability of 1E-01 to 1) proceed to Task 2.3.

Task 2.3 Catalogue the Characteristics of Postulated Unacceptable Interactions

For each LER which meets the thresholds stated in Task 2.2, to the extent possible, catalogue: the system with the postulated high energy line break, the postulated break location, the piping internal pressure, the type of break (i.e., circumferential or longitudinal break), the type of postulated interaction (i.e., pipe whip and/or jetting interaction), the piping diameter and piping thickness, the piping material, the system structure or component (SSC) target, the SSC postulated failure mode (e.g., fail high, low, as is), the adverse effect on protection systems, mitigation systems or containment systems (e.g., failure of RPS trip channels, failure of engineered safety feature automatic initiation system trip channels, loss of a containment boundary or function), the postulated single active random failure, the offsite power status, other postulated initial conditions and/or boundary conditions, operator actions (if any), the reactor system safety consequences (e.g., core damage) and, the corrective actions taken to reduce or eliminate the potential for the event.

Task 2.4 Determine the Pipe Break Probability for the Types of Piping Involved in the Postulated Breaks

Review the UFSAR and other plant-specific information sources to determine the material and diameter of the piping involved in the HELB. Obtain and review the potential pipe failure rate databases that may be appropriate for determining the pipe break probability for the type of piping involved in the postulated break. Select the reference pipe break probability database to be used. As appropriate, update the reference database to account for more recent failure rate data and failure causes. In this regard, NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995" provides updated values of HELB frequencies and are more current than those assumed in NUREG/CR-6395 and the staff’s prioritization analysis. The latter documents utilized pipe break frequencies based on NUREG-1150 and WASH-1400. As appropriate, the task will also determine the pipe break probability to be used for the applicable type of piping based on the specific system application including the piping diameter.

Task 2.5 Obtain and Analyze Piping and SSC Target Spacial Layout Information

Obtain piping isometric drawings and information on piping – target physical layout arrangements including nearby piping elbows as possible hinge points for pipe whip or jetting. As needed, conduct conference calls and/or site visits for the involved plants (that are still licensed to operate) to collect additional direct visual observation, if needed, and isometric drawings on the physical layouts involved for the unacceptable interactions. For permanently shutdown plants, search for and obtain publically available, previously docketed piping drawing information which help characterize the piping/SSC target spacial arrangements.

Task 2.6 Calculate the Initiation Frequencies of the Postulated Unacceptable Interactions

Determine the mean initiation frequency (events/RY) based on the mean pipe break probability for the type of piping involved (from Task 2.4) and an analysis of the percent of piping length and fractional pipe whip or jetting arc placing the target at risk of failure (from Task 2.5). As needed, evaluate the piping internal reaction forces and local piping bending stresses to determine the expected elastic and/or plastic deflection modes of the piping relative to the SSC target. The mean interaction frequency per reactor year should be based on the appropriate pipe break probability data and the fractional piping length which has the potential to adversely interact with the SSC based on the piping and target spacial arrangements and the type of interaction involved, the <u>fraction</u> of three dimensional space which can result in an unacceptable interaction (i.e. the effective solid angle of piping/jetting that could cause adverse effects to the total solid angle involving the possible directions of pipe break effects (whip or jetting).

Task 2.7 Determine Reference Core Damage Frequencies

Determine the reference (base) plant-specific CDF/RY for breaks (anywhere) inside containment for the piping type (diameter) involved in the postulated unacceptable interaction. Utilize the applicable plant-specific HELB event tree SPAR models for use with the SAPHIRE PRA code. Use the reference (base) failure probabilities for the applicable protection and mitigation equipment important to each pipe break event tree. Repeat these plant-specific reference (base) CDF/RY assessments for each type of piping (i.e., small, medium or large diameter) that is applicable to each postulated unacceptable interaction.

Task 2.8 Determine Core Damage Frequency for Postulated Unacceptable Interactions

Using the initiation frequency (from Task 2.6), determine the adjusted plant-specific CDF/RY for the postulated unacceptable interactions. Utilize the applicable plant-specific PRA event trees from the SAPHIRE code. For each protection and mitigation system that fails due to either: (1) the broken pipe (e.g., low pressure safety injection system train failure due to a HELB in the injection piping for one of its trains) or (2) the HELB effects (e.g., whipping pipe or jetting-induced failure of one or more pressure sensing switches that are relied on for automatic initiation of the alternate train of low pressure safety injection), adjust its associated base (reference) failure probabilities to unity. Repeat these plant-specific CDF/RY assessments for each postulated unacceptable interaction.

Task 2.9 Determine the Change in Core Damage Frequency for Each Postulated Unacceptable Interaction

For each postulated unacceptable interaction determine the change in core damage frequency (ΔCDF/RY) based on the difference between the postulated unacceptable interaction CDF/RY and the associated reference (base) plant-specific CDF/RY.

Task 2.10 Determine the Estimated Conditional Probability of Early Containment Failure

As necessary (for scenarios involving an estimated ΔCDF/RY $\geq$ 1E-6), determine the estimated conditional probability of early containment failure if the unacceptable interaction involves increased potential for (i.e., consequential) early containment failure due to the effects of the HELB.

Task 3 <u>Hypothetical Unacceptable HELB Interactions Used for Prioritizing GSI-156.6.1</u>

Seven BWR SEP-III events and three PWR SEP-III events involving postulated unacceptable HELB interactions inside containment were hypothesized for the prioritization of GSI-156.6.1. These hypothesized scenarios will also be included in the technical evaluation. A very brief description and PRA (regulatory analysis) of these events are documented in Attachment 1 to the memorandum from Ashok Thadani to Charles Rossi, "Prioritization and Transfer of Responsibility for Generic Safety Issue 156.6.1, "Pipe Break Effects on Systems and Components Inside Containment," dated July 16, 1999. The traditional engineering (i.e., deterministic) analysis and the probabilistic analysis used to estimate the core damage frequencies for these hypothetical HELB interactions are believed to involve significant conservatism. For the tasks below, the assumptions and inputs used for these hypothetical events will be assessed and the analysis modified as, appropriate, to achieve greater realism. The most risk significant of the scenarios (i.e., those which satisfy the safety goal screening criteria for proceeding with the value-impact portion of a regulatory analysis) will be re-analyzed with respect to the change in core damage frequency and the estimated conditional probability of early containment failure using more realistic inputs and assumptions for these interaction events as described in the tasks below.

Task 3.1 Collect, Review and Discuss Information Describing the Most Risk Significant Hypothetical Unacceptable Interactions, Catalogue Characteristics

Obtain from INEEL all reference information, (e.g., photographs, field notes, trip reports) for the most risk significant postulated unacceptable interactions that were hypothesized for BWRs and PWRs in connection with the enhanced prioritization of GSI 156.6.1. Review the INEEL information to determine and document the detailed characteristics for each identified hypothetical unacceptable interaction. Discuss these interactions with INEEL contract personnel involved in the prioritization study to resolve questions and issues in order to develop a complete understanding of the interactions. For each of these hypothetical interaction, to the extent possible, catalogue: the visited plant(s) for which the events are applicable, the high energy system with the break, the precise location of the postulated break, the piping internal pressure, the break type postulated (i.e., circumferential or longitudinal break), the interaction type (i.e., pipe whip or jetting), the piping diameter and thickness, the piping material, the system, the appearance structure or component (SSC) target(s), the SSC failure mode (e.g., fail high, low, as is), the effect on protection systems (e.g., failure of RPS trip channels), mitigation systems (e.g., failure of engineered safety feature automatic initiation system trip channels) or containment systems (e.g., loss of a containment boundary or function), the assumed single active random failure (if any), the offsite power status, other postulated initial conditions and/or boundary conditions, operator actions (if any), and the reactor system safety consequences (e.g., core damage).

Task 3.2 Request Review Comments from SEP-III Plant Licensees, Owners Groups, Etc.

Request comments from selected SEP-III plant licensees and industry groups on the hypothetical unacceptable interactions. Send the prioritization analysis and the draft NUREG/CR-6395 to the licensees of the BWR and PWR plants that were visited for the prioritization, each LWR owners group, NEI, INPO, etc. Request that they review and provide comment on these documents. Provide selected additional detailed reference information collected in Task 3.1 to the licensees of the sites visited (and the public) and the industry groups to assist them in understanding the most risk significant interaction scenarios that were postulated for their respective plants. The objective of these requests is to collect additional information <u>on a voluntary basis</u> which identifies either sources of elevated conservatism in the hypothetical scenarios, or alternatively validates the assumptions for the engineering analysis basis and the PRA basis of these limiting scenarios. Comments could be based on information in the literature or licensee knowledge of their individual plant designs. Information on the plant-specific equipment arrangements would be used to determine where and how the prioritization analysis might br overly conservative, incorrect or reasonable. The information would also be used to assess whether, assuming a break, the model for the pipe break effects, or the model of the plant (or operator) response to the postulated break is incorrect, or overly conservative or reasonable. As required, discuss the purpose and desired outcomes of these requests with representatives of these organizations. Review and evaluate comments provided by these groups. Determine and document how these comments impact the assumptions, models and results for the staff's prioritization PRAs for BWRs and PWRs. As appropriate, revise the assumptions and input for these events or the plans for the technical evaluation.

Task 3.3 Obtain Stakeholder Comments on the Prioritization Analysis and Technical Evaluation Plan

Plan and conduct a meeting (or conference call), open to the public, to discuss and obtain stakeholder comments on the prioritization analysis and the plans for the technical evaluation. Review and evaluate internal and external stakeholder comments. Determine and document how these comments impact the assumptions, models and results for the staff's prioritization analyses for BWRs and PWRs. As appropriate, revise the plans for the technical evaluation.

Task 3.4 Determine the Pipe Break Probability for the Involved Piping Type

Obtain and re-review the potential pipe failure databases that may be appropriate for determining the pipe break probability for the type of piping involved in the hypothetical unacceptable interactions. Select the reference pipe break probability database to be used. As appropriate, update the reference database to account for more recent failure rate data and failure causes. As appropriate, the task will also determine the pipe break probability to be used for the applicable type of piping based on the specific system application including the involved piping diameter.

Task 3.5 Obtain and Analyze Piping and SSC Target Spacial Physical Layout and SSC Target Vulnerability Information

Obtain piping isometric drawings and information on target physical layout arrangements for each of the hypothetical unacceptable interactions, including nearby piping elbows as possible hinge points for pipe whip or jetting. For the identified unacceptable interactions, as needed, conduct conference calls with the NRC resident or licensee technical staff and/or conduct site visits for the applicable SEP-III plant. Collect additional direct visual observation if possible and drawings on the physical layouts involved for the unacceptable interactions. Where questions exist, obtain additional detailed information on the precise safety functions and potential failure modes and effects for the SSC target(s).

Task 3.6 Obtain Insights from SEP-II Plant Reviews or Engineering Analyses to Resolve Hypothetical HELBs

Review the documents collected for Task 1.1 to assess whether any of the hypothetical unacceptable interactions identified in connection with the most risk significant interactions postulated for the enhanced prioritization of GSI 156.6.1 can be made more realistic (e.g., eliminated) based on information contained in licensee safety analyses and staff safety evaluations conducted in connection with the SEP reviews for the SEP-II plants or engineering analyses. Determine if interactions can be the resolved as a result of application of prior acceptable deterministic (i.e. traditional engineering) analyses of the hypothesized HELB scenarios (e.g., BWR Mark I containment liner failure). Compare the analysis basis for such candidate scenarios to the circumstances applicable to the SEP-III scenarios. Obtain additional information as may be needed for selected SEP-III plants to determine the applicability of the SEP-II results. Conduct engineering analyses as may be appropriate to validate the SEP-II results as applicable to the SEP-III plants. For example, utilize a contractor to conduct a detailed traditional mechanical engineering (e.g., finite element) analysis of a generic BWR containment liner subjected to postulated pipe whip impacts from a steam line, feedwater line, or recirculation line. Alternatively, conduct a comparative (i.e., parametric) analysis of SEP III plant containment liner designs to SEP III plant containment liner designs. Use these results to assess the applicability of the containment liner integrity analysis and results from the SEP II plant licensee studies and SEP II plant staff reviews to the SEP III BWR plants. (See attachment 1 for additional information.)

Task 3.7 Validate Hypothetical Unacceptable HELB Interactions

Determine which (if any) of the hypothetical unacceptable HELB interactions for BWRs and/or PWRs can be reasonably resolved (i.e., dropped) or reduced in probability on the basis of the additional deterministic (i.e., traditional engineering) insights and information collected from Tasks 3.1 – 3.6. These are the hypothetical unacceptable interaction scenarios which the re-assessment determines to, in fact, <u>not</u> involve an unacceptable interaction or a much lower probability when considered in view of the additional information. Additional deterministic insights and information might include: low stress and low fatigue levels in the high energy piping postulated to break; physical and/or electrical separation of redundant and/or diverse SSCs postulated to simultaneously fail as a result of the pipe whip or jetting; previously unrecognized physical protection devices that are installed; existing structural (survivability) analysis of the potentially effected SSCs; existing analysis of the effects on the functionality of the potentially affected protection or mitigation systems; existing analysis of the overall reactor system response which shows that significant core damage would not occur. Document the scenarios that can be reasonably resolved due to the additional insights and information and the engineering analysis basis for resolution.

Task 3.8 Modification of Probabilistic Risk Assessment for the Hypothetical Unacceptable HELB Interactions

For the hypothetical unacceptable HELB interaction scenarios which can <u>not</u> be reasonably resolved (i.e., dropped) on the basis of the additional deterministic insights and information, determine where and how (for individual scenarios) the insights and information can be used to either: (1) verify the prioritization PRA ΔCDF/RY analysis or reduce elements of elevated conservatism in the PRA analysis. Document the scenarios for which it is concluded that it is reasonable to revise the PRAs because of the additional information, how the PRA analysis basis should be revised and, the technical basis for the revision.

Task 3.9 Calculate the Mean Initiation Frequencies of the Hypothetical Unacceptable HELB Interactions

For validated risk-significant unacceptable HELB interaction scenarios, determine the mean initiation frequency (interaction events/RY). The frequency should be based on the mean pipe break probability for the type of piping involved (from Task 3.4) and the engineering analysis of the percent of piping length and fractional pipe whip or jetting arc placing the target in the line of failure (from Task 3.5). As needed, evaluate the piping internal reaction forces and local piping bending stresses to determine the expected elastic and/or plastic deflection modes of the piping relative to the SSC target. The interaction events/RY should be based on the appropriate pipe break probability data and the fractional piping length which has the potential to adversely interact with the SSC based on the piping and target spacial arrangements and the type of interaction involved, the fraction of three dimensional space which can result in an unacceptable interaction (i.e. the effective solid angle of piping/jetting that could cause adverse effects to the total solid angle involving the possible directions of pipe break effects).

Task 3.10 Determine the Reference Core Damage Frequency

Determine the reference (base) plant-specific CDF/RY for breaks (anywhere) inside containment for the piping type (e.g., diameter) involved in validated postulated unacceptable HELB interaction. Select the applicable plant-specific SAPHIRE code HELB event tree SPAR models for the PWR (or BWR) SEP-III plants with a range of CDF/RY such that the plant-specific CDF/RY for the type of break varies in risk importance. Use the applicable base (reference) pipe break initiation frequency and the failure probability for the involved protection and mitigation equipment credited in the plant-specific event trees.

Task 3.11 Determine the Core Damage Frequency for the Validated Unacceptable HELB Interactions

Using the initiation frequencies (from Task 3.9), determine the plant-specific CDF/RY for each validated unacceptable HELB interaction for the range of applicable plants selected for Task 3.10. Select the same plant-specific HELB event tree SPAR model for PWR (or BWR) plants with the range of CDF/RY used in Task 3.9. For each protection and mitigation system that fails due to either: (1) the broken pipe (e.g., LPSI system A train failure due to a break in the LPSI system A train piping) or, (2) the HELB effects (e.g., whipping or jetting-induced failure of one or more pressure sensing switches that are relied on for automatic initiation of the alternate train of low pressure safety injection), adjust its associated base (reference) failure probabilities to unity. Determine the adjusted plant-specific CDF/RY assessments for each validated unacceptable HELB interaction.

Task 3.12 Determine the Change in Core Damage Frequency for the Validated Unacceptable HELB Interaction

For each validated unacceptable HELB interaction determine the change in core damage frequencies (ΔCDF/RY) based on the difference between the hypothetical unacceptable interaction CDF/RY and the associated reference (base) plant-specific CDF/RY.

Task 3.13 Determine the Estimated Conditional Probability of Early Containment Failure

As necessary (for any scenario involving an estimated ΔCDF/RY $\geq$ 1E-6), determine the estimated conditional probability of early containment failure if the unacceptable interaction involves increased potential for (i.e., consequential) early containment failure due to the effects of the HELB.

Task 4 Combine and Plot the Results for the Involved Sample of SEP III Plants

Combine (add) the ΔCDF/RY results for the scenarios from Tasks 1, 2 and 3 on a plant-by-plant basis to determine the total ΔCDF/RY due to the identified validated postulated unacceptable interactions inside containment for the involved sample plants. Identify the appropriate estimated conditional probability of early containment failure for each involved plant. Plot the plant-level data in accordance with Figure 3.2, "Safety Goal Screening Criteria" of NUREG-BR–0058, Revision 3, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission". Determine the industry average ΔCDF/RY for BWRs and for PWRs.

Task 5 Conduct Uncertainty and Sensitivity Analyses and Compare with Safety Goal Screening Criteria

Conduct a sensitivity analysis of the change in plant-by-plant ΔCDF/RY and industry average ΔCDF/RY for postulated HELBs inside containment for changes in the frequency of piping failure. Conduct an uncertainty analysis to determine the upper bound of the ΔCDF/RY for the worst-case BWR SEP III plant and the worst-case PWR SEP III plant based on the variability in ΔCDF/RY evident from the involved BWR and PWR SEP-III plant samples. Identify the appropriate estimated conditional probabilities of early containment failure and show results in accordance with Figure 3.2 of NUREG-BR–0058. If the results of the safety goal screening evaluation is not favorable (i.e., no action) terminate the technical evaluation and document the results. If the safety goal evaluation is favorable, (i.e. any decision except "no action"), proceed to Task 6.

Task 6 Estimate and Evaluate Values and Impacts

As necessary, conduct an estimation and evaluation of the values and impacts in accordance with Section 4.3 of NUREG-BR–0058, Revision 3. Perform and document the results in accordance with NUREG-BR–0058, Section 4.0, "Elements of a Regulatory Analysis."

Task 7 Document and Present Results and Prepare Recommended Regulatory Closeout

Formally document the results of the technical evaluation in a draft resolution document. Obtain and address peer review comments on the resolution document from NRC staff. Prepare the draft final resolution document. Forward the draft final resolution document to the Advisory Committee on Reactor Safeguards (ACRS). Present the results of the technical evaluation to the ACRS and obtain their comments. Finalize the Technical evaluation report. Prepare an appropriate proposed regulatory document (e.g., generic letter, regulatory information letter) and forward to NRR for issuance to the potentially affected licensees.

3. BASIS FOR CONTINUED OPERATION

The generic safety issue addressed by this TAP involves pipe break effects on systems and components inside containment. For the PWRs and BWRs potentially affected by the results of the technical evaluation involved in this TAP, it has been concluded that, pending the completion of this TAP, continued operation does not constitute an undue risk to the health and safety of the public for the following reasons:

A high energy line pipe break inside containment has never occurred at an operating LWR.

The probability of a high energy line pipe break inside containment during power operation is small and the probability that the break would occur at a piping location which could fail a system, structure or component which could result in the inability to safely shutdown the plant is even smaller.

No design basis issue involving a postulated unacceptable high energy line pipe break inside containment found and reported to date had an estimated risk meeting the Accident Sequence Precursor threshold.

Since initial licensing, severe accident management procedures and training have been implemented at each of the potentially affected LWRs thereby increasing the capability of these plants to effectively cope with a postulated high energy line break of the sort involved in this GSI.

The rationale described above constitutes the basis for continued operation until the GSI is resolved.

NRC TECHNICAL ORGANIZATION INVOLVED

This section indicates the functional and organizational responsibilities for performing this TAP until final disposition of all encompassed issues.

A. Task Manager will provide overall project direction, coordination, budgeting and scheduling and value impact analysis activities.

   Manpower Estimates: 0.17 FTE over 18 months

B. Mechanical and Structural Analyst will have the responsibility for assessing containment liner failure, pipe break probability and the engineering assessment of the HELB interaction frequency between the high energy line and the target SSC for each postulated unacceptable interaction at each of the involved plants.

   Manpower Estimates: 0.15 FTE over 18 months

C. Probabilistic Risk Assessment Analyst will have the lead responsibility for the safety goal analysis including assessing the change in core damage frequency for each postulated unacceptable interaction at each of the involved plants and the conduct of the sensitivity and uncertainty analyses.

   Manpower Estimates: 0.13 FTE over 18 months

D. Reactor Systems Analyst will have the lead responsibility for assessing the systems level consequences on reactor, containment, protection and mitigation systems for the effects of postulated HELBs inside containment for each of the involved plants. (This function may be provided by the PRA analyst and/or task manager)

   Manpower Estimates: 0.04 FTE over 18 months

E. Contract Dollars for Technical Assistance

   Contractor assistance would be used for: (1) conduct of a public meeting on the prioritization study and the TAP; (2) compilation, orderly turnover and debrief on reference information collected but not published in the prioritization study and; (3) conduct of detailed traditional engineering (finite element) analyses and evaluation of the BWR Mark I containment liner integrity.

   Contract Dollars: $300K over 18 months

It should be noted that for purposes of the resource estimates, it is assumed that none of the events analyzed in connection with Task 1 or Task 2 will pass the screening test. Therefore, it is assumed that none of the follow-on sub-tasks needed to conduct a detailed PRA (to assess the ΔCDF/RY) will need to be implemented. It is further assumed that in all cases it will be found that the integrated safety significance (including uncertainties) of all of the identified postulated unacceptable interactions attributable to particular plants do not meet the safety goal screening criteria. Therefore, it is assumed that it will not be necessary to conduct a value-impact analysis. The following table provides detailed information on the estimated staff resources required to conduct the technical assessment of GSI 156.6.1.

	Man-hours				
TASK	Mechanical-Structural Design Analysis	Reactor and Containment Systems	Probabilistic Risk Assessment	GSI Task Management	Total
1.0	Unacceptable HELB Interactions Identified by the SEP-II Plant Reviews				
1.1	40	–	80	240	360
1.2	4	–	–	–	4
1.3	–	–	80	20	100
1.4	–	–	–	–	TBD[2]
1.5	–	–	–	–	TBD
1.6	–	–	–	–	TBD
1.7	–	–	–	–	TBD
1.8	–	–	–	–	TBD
1.9	–	–	–	–	TBD
Total	40+TBD	TBD	160	260	464+TBD
2.0	Unacceptable HELB Interactions Identified in Licensee Event Reports				
2.1	–	–	–	80	80
2.2	–	–	40	20	60
2.3	TBD	TBD	TBD	TBD	TBD
2.4	TBD	–	TBD	TBD	TBD
2.5	TBD	–	–	TBD	TBD
2.6	TBD	–	–	TBD	TBD
2.7	–	–	TBD	TBD	TBD
2.8	TBD	–	TBD	TBD	TBD
2.9	–	–	TBD	TBD	TBD
2.10	TBD	TBD	TBD	TBD	TBD
Total	TBD	TBD	40 + TBD	100+TBD	140+TBD
3.0	Hypothetical Unacceptable HELB Interactions Used for Prioritizing GSI-156.6.1				
3.1	20	40	20	50	130
3.2	10	–	40	80	130
3.3	40	–	40	80	160
3.4	4	–	1	1	6
3.5	60	–	20	80	160
3.6	80	–	–	40	120
3.7	10	–	10	10	30
3.8	10	–	20	10	40
3.9	20	–	20	–	40
3.10	–	–	10	–	10
3.11	8	–	10	–	18
3.12	–	–	2	–	2
3.13	2	–	4	2	8
Total	264	40	197	353	854
4	Combine and Plot the Results for the Involved Sample of SEP III Plants				
	–	–	2	2	4
5	Conduct an Uncertainty Analysis and Sensitivity Analysis and Compare with Safety Goal				
	8	–	24	8	40
Subtotal	312	40	223	363	1161
6	Estimate and Evaluate Values and Impacts				
	–	–	–	TBD	TBD
7	Document and Present Results and Prepare Recommended Regulatory Closeout				
	40	40	40	80	200
TOTAL (Hrs)	312	80	263	363	1361
TOTAL (PSY)	0.15	.04	0.13	0.17	0.65

Table: Staff Resource Requirements for the Technical Assessment Resolution of GSI 156.6.1

[2] Resources are required only if one or more events passes the screening assessment.

Task Action Plan Schedule
(months)

	Months																			
Tasks	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	Staff Week
1.1, 2.1, 3.1																				
1.2, 3.4																				
3.2[3], 3.3																				
1.3, 2.2,																				
3.5[4]																				
3.6																				
3.7, 3.8																				
3.9 -3.13																				
4, 5																				
7																				
6															TBD	TBD	TBD	TBD		

[3] Task 3.2 was initiated before the final TAP was issued

[4] The timing of this task is determined by the refueling outage schedules of the involved plants

BWR Mark I Containment Liner Failure Assessment

BWR plants within the scope of GSI 156.6.1 involve a Mark I containment design. If a determination can be made that there is a similarity in the principal parameters that would be considered in an evaluation of containment liner failure, such as liner material, thickness and toughness, liner-to-concrete wall gap, pipe size and geometry, and the energy of the whipping pipe transmitted to and absorbed by the liner, it may be possible to extend the staff's conclusions on the SEP-II plants to the SEP-III plants within the scope of GSI 156.6.1. Another acceptable approach to confirm or reject the basis for the containment liner failure postulation, on which the INEEL prioritization is based, might be to perform a detailed generic engineering (e. g., non-linear finite element) analysis of a BWR containment liner subjected to a worst case pipe whip impact from a broken steam line, feedwater line, or recirculation line. The former could be a more cost effective approach to addressing the issue.