NDE 20250612-001

June 12, 2025

U.S. Nuclear Regulatory Commission

Document Control Desk
11555 Rockville Pike
1 White Flint N; Mail Stop: 0-12-D2
Rockville, MD 20852

EPRI Docket No. 99902021

Subject: ELECTRIC POWER RESEARCH INSTITUTE – SUBMITTAL OF RESPONSES TO THE US NRC's REQUEST FOR ADDITIONAL INFORMATION (MAY 2025) ON EPRI TECHNICAL REPORT 3002028939, "RISK-INFORMED HIGH-ENERGY LINE BREAK EVALUATION REQUIREMENTS".

By letter dated July 23, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24205A146), the Electric Power Research Institute (EPRI) submitted EPRI Technical Report (TR) 3002028939, "Risk-Informed High-Energy Line Break Evaluation Requirements," dated June 2024, to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. EPRI TR 3002028939 provides an alternative risk-informed methodology for assessing and confirming that plant structures, systems, and components that are important to safety are adequate to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures.

By letter dated September 10, 2024 (ADAMS Accession No. ML24214A027), the NRC staff accepted EPRI TR 3002028939 for review. During the regulatory audit review process and discussions with EPRI staff, a set of requests for additional Information (RAIs) were shared in draft form with EPRI on May 8, 2025, and the final RAIs were shared with EPRI via e-mail on May 19, 2025 (ML25129A090).

Please find attached EPRI's formal responses to the RAIs received on May 19, 2025. In addition, please note that the RAI responses also discuss modifications to be made in the report based on discussions with NRC staff during the audit.

We recognize the time and effort made by the NRC staff in the review of EPRI TR 3002028939 and appreciate the opportunity to provide responses to the RAIs received on May 19, 2025. We are at your disposal to provide any clarifications to the RAI responses, or any other clarifications needed with respect to the review of this report

Sincerely,

Michael Ruszkowski
Director, Plant Support
Nuclear Sector, EPRI

20250612-001/ccm

c: Lois James, NRC
Rick Fougerousse, EPRI
Ashley Lindeman, EPRI
Fernando Ferrante, EPRI

ATTACHMENT 1 – RAI RESPONSES ON TR 3002028939

ELECTRIC POWER RESEARCH INSTITUTE REPORT 3002028939, "RISK-INFORMED HIGH-ENERGY LINE BREAK EVALUATION REQUIREMENTS"

REQUESTS FOR ADDITIONAL INFORMATION

1. Mechanical Engineering & Inservice Testing

Regulatory Basis: Appendix A, "General Design Criteria for Nuclear Power Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," provides the principal design criteria that establish the necessary design, fabrication, construction, testing, and performance requirements for systems, structures, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," states:

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

RAI 1

Can licensees use Electric Power Research Institute (EPRI) Technical Report (TR) 3002028939, "Risk-Informed High-Energy Line Break Evaluation Requirements [(RI-HELB)]," June 2024, to address nonconforming or degraded conditions of high- energy line break (HELB) SSCs with respect to the licensing basis? Does EPRI TR 3002028939 contain any limitations with respect to evaluating nonconforming or degraded conditions of HELB SSCs?

EPRI Response to RAI 1:

TR 3002028939 does not contain any limitations with respect to evaluating nonconforming or degraded conditions of HELB SSCs. The recommended process is that after EPRI TR 3002028939 is approved, a licensee then makes a submittal or 50.59 change. After approval of the submittal or the 50.59 change, the licensee can then assess the non-conforming condition in accordance with the deterministic approach or the risk-informed (RI) approach. If the licensee does not have an approved RI-approach, then the plant cannot "RI" the operability evaluation. If the RI-approach is approved, the licensee can use it for an HELB operability evaluation. For example, if the safety significance of a postulated pipe rupture was originally determined to be "RC6" but an issue is found and it becomes "RC5" due to the issue, then the licensee can still follow the RI-methodology.

RAI 2

Provide explanation whether or not EPRI TR 3002028939 can be used to change the licensing basis for the plants that have implemented EPRI TR-1006937, "Extension of the EPRI Risk-Informed ISI [inservice inspection] Methodology to Break Exclusion Region Programs," April 4, 2002, as well as plants that have not implemented the EPRI TR-1006937 methodology.

Specifically, the requirement for 100 percent volumetric inservice examination of all pipe welds for piping near the containment penetration area should be conducted during each inspection interval as defined in IWA-2400, American Society of Mechanical Engineers (ASME) Code, Section XI per U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position (BTP) Mechanical Engineering Branch 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside of Containment" June 19, 1987 (ADAMS Accession No. ML031150493), and updated in BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment" March 2007 (ADAMS Accession No. ML070800008).

EPRI Response to RAI 2:

The methodology in TR 3002028939 is not applicable to break exclusion region (BER) piping. This includes plants that have implemented EPRI TR-1006937, Rev. 0-A, "Extension of the EPRI Risk-Informed Inservice Inspection (RI-ISI) Methodology to Break Exclusion Region (BER) Programs," August 2002, as well as plants that have not implemented the EPRI TR-1006937 methodology.

The introduction and summary sections of TR 3002028939 will be revised to add the new wording below. This change is also addressed in the response to RCI 1.

This methodology is applicable to all Class 2, 3 and non-safety related systems with the exception of that portion of piping that lies within the break exclusion region (also known as no break zone).

RAI 3

EPRI TR 3002028939 does not contain discussion on an appropriate zone of influence (i.e. distance of jet impingement effects from break to impacted equipment) used to determine the SSCs that are potentially subject to a HELB and/or jet impingement load. Provide an explanation and description of the zone of influence. Note: EPRI TR-1006937 Section 2.3.6 states, in part: "6. Jet Impingement – SRP [Standard Review Plan] 3.6.2 [Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping] may be used to evaluate jet impingement targets and potential load impact on structures, systems, and components. In lieu of SRP 3.6.2, plant-specific criteria and analyses may be used, and conservative assumptions and engineering judgments derived from plant design and analysis may be used as follows: Electrical or active equipment within the zone of influence of the break is assumed to fail (e.g., active valve is assumed to fail in its normal position prior to break) unless otherwise qualified. The typical zone of influence is 10 to 20 pipe diameters (e.g. NUREG/CR 2913, "Two-Phase Jet. Loads," January 1983, [ADAMS Accession No. ML073510076], Reference 6)."

EPRI Response to RAI 3:

Section 2.1.3 of TR 3002028939 will be updated as follows concerning the zone of influence and other changes for consistency with TR 1006937 (changes are highlighted).

2.1.3 Consequence Evaluation

Because piping within the scope of a RI-HELB application is normally operating (that is, pressurized and at high temperature), the postulated PBF will result in an initiating event or forced plant shutdown. The evaluation of the impact of the postulated PBF should be accomplished using a plant-specific list of initiating events from the plant PRA and design basis documentation (that is, HELB documentation). For systems previously not within the scope of the current HELB program, this could also include events that might not be explicitly modeled by either process. When a PBF causes an initiating event that is not explicitly modeled in the PRA or causes an initiating event modeled in the PRA but with additional mitigating impacts, PRA quantification is required. The following provides a general procedure for the PRA quantification:

- Applicable Initiator is set to 1.0 (this provides a CCDP/CLERP result).
- All other initiating events are set to 0.0.
- Applicable Impacts are set to TRUE using basic events to simulate the impacts.
- The preceding is done for CDF and LERF quantification; the result is CCDP and CLERP.

An initiating event is likely as a result of a HELB (for example, steam or feedwater line break); additional mitigation impacts can also occur, such as loss of a system (for example, loss of charging, feedwater, and so forth) because of an indirect effect (for example, spraying/jet impingement of an electrical bus, flooding of the room, and so forth). When conducting the evaluation of HELB impact, there are eight consequence evaluation criteria to be considered; each is summarized as follows (Note that these criteria must also be used for deterministic analysis of HELB as described in the SRPs.):

- **Containment isolation valves**. Valves in the vicinity of the break are assumed to fail unless survival is justified by plant design and/or analysis.

- **Containment penetrations**. Assumed to fail if not designed or analyzed for a DEGB load. Design features can be credited to preclude DEGB loads. When failure of the penetration is assumed (for example, no design or analysis information demonstrates otherwise), the leakage around the penetration failure is assumed to be large enough to satisfy the large release portion of CLERP in the consequence evaluation, unless analysis can justify smaller releases.

- **Unrestrained whipping pipe impact on equal or larger nominal pipe size (NPS)**. No impact except on thinner wall pipe where through-wall cracks are assumed unless there is analytical and/or experimental justification. Through-wall cracks are postulated if the impacted pipe has thinner wall thickness, except where analytical and/or experimental data for the expected range of impact energies demonstrate the capability to withstand an impact without rupture (for example, SRP 3.6.2).

- **Unrestrained whipping pipe impact on smaller NPS**. Failure is assumed unless it is demonstrated capable by design or analysis. Circumferential and longitudinal breaks are postulated except where analytical and/or experimental data demonstrate capability.

- **Unrestrained whipping pipe impact on SSCs**. Plant-specific criteria and analyses and/or SRP 3.6.2 are used to evaluate potential physical impacts of pipe whip. Engineering judgments based on plant design and analyses are used along with conservative assumptions to determine impacts such as the following.

  - Conservatively apply unrestrained piping length to identify potential targets.

  - If a structural target is designed similar to another structural target that has already been analyzed for pipe whip impact with similar loads, then this can be used as a reasonable basis. Otherwise, the structural target (for example, common wall with adjacent area) is assumed to fail.

- Equipment with active functions or electrical equipment, such as a motor- or air-operated valve, is assumed to fail (valve is assumed to fail in its normal position prior to the break). Check valves can be treated like piping (as described above).

- The determination of pipe whip potential (for example, potential for developing a hinge) can be derived from plant analyses of similar configuration.

- **Jet impingement**. Plant-specific criteria and analyses and/or SRP 3.6.2 are used to evaluate potential impacts of jets. Engineering judgments based on plant design and analyses are used along with conservative assumptions to determine impacts such as the following.

  - Electrical or active equipment within the zone of influence of the break is assumed to fail (for example, an active valve is assumed to fail in its normal position prior to break) unless otherwise qualified. The typical zone of influence is 10 to 20 pipe diameters (see NUREG/CR-2913).

  - If a structural or passive component-type of target is designed similar to another target, already analyzed and found to be acceptable for similar loads, then this can be used as a reasonable basis. Otherwise, the target (for example, common wall with adjacent area) is assumed to fail.

  - Plant analyses of jet impingement can be used to derive insights into potential impacts. For example, the jet impingement impact from another analyzed pipe that has a similar zone of influence can be used.

- **Other spatial impacts**. SSCs in the area of the break are assumed to fail unless design/analyses or appropriate engineering judgments, based on plant design and spatial evaluations, justify otherwise. Equipment qualification for the DEGB environment must be considered, as well as flooding and compartment overpressure. The following provides additional guidance:

  - Physical separation can be credited with regard to the containment structure and isolation. For example, equipment inside containment can be credited with isolating a break outside containment. For large high-energy line breaks, only automatic isolation can be credited and it must be qualified per design basis.

  - Equipment Qualification (EQ) – Equipment in affected areas might have been qualified as part of an EQ program. If this equipment is to be credited in the RI-HELB evaluation, then the harsh environment identified as part of the EQ profile (temperature, pressure, humidity, jet impingement, and pipe whip) will need to envelope (or equal) the environment created by the assumed RI-HELB break. Caution should be applied because the RI-HELB break will always assume that the equipment available to isolate the break has an inherent unreliability. That is, the RI-HELB evaluation looks at both successful and unsuccessful isolation (and the resultant environments).

  - Temperature, pressure, water spray, flooding, and compartment pressure must be considered when evaluating the impacts previously described. Electrical equipment in the break area is assumed to fail unless a technical basis and/or qualification are available. Engineering judgments based on plant design can be used to evaluate whether compartment pressure can cause catastrophic failure of the room. An isolated room should be assumed to fail unless analysis can demonstrate otherwise.

- **Spatial propagation**. When postulating propagation to adjacent areas, both isolation success and failure are considered. For the failure-to-isolate case, the consequences are likely to be unanalyzed (beyond design basis), thus, spatial propagation impacts must be analyzed or core damage assumed (CCDP = probability of isolation failure). For the isolation success case, the environmental impacts might be similar to analyzed cases. Engineering judgment can be used based on plant design and analysis that is consistent with Probabilistic Risk Assessment (PRA)/Individual Plant Examination for External Events (IPEEE) studies, such as the following examples:

– Equipment in the vicinity of the propagation path (on the other side of a door or wall failure) is assumed to fail unless qualified or protected from the break (similar to design basis or SRP 3.6.2).

– For the isolation failure case, spatial propagation must be evaluated relative to impacts and equipment and is assumed to fail unless qualified or protected (similar to design basis or SRP 3.6.2). Secondary propagation paths have to be considered as propagation continues to other areas.

– For the successful isolation case, impacts beyond the immediate vicinity of the propagation path depend on distance, size of the adjacent room or area, and vent path (for example, openings to an adjacent room or upper elevations).

The extension of the RI-ISI methodology to RI-BER and RI-HELB is depicted in Figure 2-7.

RAI 4

EPRI TR 3002028939, Section 2.1.1, states, in part,

> As application of the RI-HELB methodology applies to high-energy systems, the likelihood of having significant time available for operator actions may be limited. Typically, only automatic isolation is credited for HELB events if the event does not prevent isolation from functioning. In considering very small breaks that do not generate automatic signals, detection and isolation is considered, but the spatial impacts are much less significant and there has to be time, detection, etc. If isolation is possible, the consequence assessment should be conducted for both cases: successful and unsuccessful isolation. Operator recovery actions are further covered in Section 3.3.3.2 of EPRI TR-112657.

EPRI TR 3002028939, Section 3.2.1, states, in part,

> For smaller breaks, it is anticipated that operator actions to scram the plant and turbine are expected to occur before an automatic action occurs. There are numerous indications to the operators as follows:

EPRI TR 1006937 states, in part, “Physical separation can be credited with regard to the containment structure and isolation. For example, equipment inside containment can be credited with isolating a break outside containment. For high-energy line, only automatic isolation can be credited, and it must be qualified per design basis.”

Provide explanation whether or not operating manual recovery actions per Section 3.3.3.2 of EPRI TR-112657, “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” Revision B-A, December 1999 (ADAMS Accession No. ML013470102), for all high-energy piping are part of the EPRI TR 3002028939 methodology.

EPRI Response to RAI 4:

Section 2.1.1 of TR 3002028939 will be updated as follows to address how operator manual recovery actions are considered (changes are highlighted).

2.1.1 Fundamental Principles

The possibility of isolating a break is also identified and accounted for as part of the consequence analysis. A break could be isolated by a protective check valve or a closed isolation valve, or it could be automatically isolated by an isolation valve that closes on a given signal. If not automatically isolated, a break can be isolated by an operator action, given successful diagnosis. The likelihood of isolating a break depends on the availability of isolation equipment, a means of detecting the break, the amount of time available to prevent specific consequences (for example, flooding of the room or draining of the tank), and human performance (see examples below).

As application of the RI-HELB methodology applies to high energy systems, the likelihood of having significant time available for operator actions may be limited. Typically, only automatic isolation is credited for HELB events if the event does not prevent isolation from functioning. In considering very small breaks that do not generate automatic signals, detection and isolation is considered, but the spatial impacts are much less significant and there has to be time, detection, etc. (see examples below).

If isolation is possible, the consequence assessment should be conducted for both cases: successful and unsuccessful isolation. Operator recovery actions are further covered in Section 3.3.3.2 of EPRI TR-112657.

Note that not all HELB breaks will be isolable by operator actions. The following provides examples:

- For large breaks (e.g., double-ended breaks) that are usually limiting with regard to consequence, only automatic isolation should be credited as there is not enough time for operator response (e.g., blowdown has already occurred and causes consequences).

- For small breaks that are not large enough to generate existing automatic isolation signals, operator actions can only be credited if the following are met:

  - There is an alarm and/or clear indication to which the operator will respond

  - The response is directed by procedure

  - The isolation equipment (e.g., valves) is not affected by the break

  - There is enough time to perform isolation and reduce consequences

Other – it is possible that smaller piping could become HELB lines where a double-ended break is small with minor consequences and possibly there is no automatic signal. If operator actions are credited, the above requirements must also be applied.

RAI 5

Regarding EPRI Report 3002028939, Section 2.4, RC5 (without flow-accelerated corrosion (FAC)) plant modification to reduce consequence to Low (RC6) or 10 percent inspection based on degradation mechanism, the NRC staff needs clarification for the 10 percent inspection.

What is the frequency/inspection method (ultrasonic testing (UT), radiograph testing (RT) or visual testing (VT)?) for the 10 percent inspection? What is the impact of the inspection result on the Risk Characterization? Please clarify plant modifications.

EPRI Response to RAI 5:

See below for responses to each specific question.

- Provide clarification for 10 percent inspection

  Past experiences have shown that a 10% inspection population size is a reasonable size for highly reliable pressure boundary components from a large break potential perspective and medium consequence of failure perspective. The RI-HELB methodology also contains change in risk requirements consistent with Reg Guide 1.174 acceptance criteria. If this assessment shows unacceptable increases in risk, then additional plant actions are required (e.g. larger inspection population, plant modifications, etc.).

- What is the frequency/inspection method for 10 percent inspection?

  Application of RI-ISI uses NDE techniques that are designed to be effective for specific degradation mechanisms and examination locations. This inspection for cause approach involves identification of specific damage mechanisms that are likely to be operative, the location where they may be operative, and appropriate examination methods and volumes specific to address the damage mechanism. This approach provides assurance that risk significant locations selected for examination will be examined using effective methods.

  Appendix VIII to ASME Section XI provides requirements for demonstrating the effectiveness of ultrasonic examination procedures and personnel for Section XI examinations. The scope of Appendix VIII does not include all damage mechanisms and locations relevant to RI-ISI such as FAC and MIC in augmented inspection programs—examination for these damage mechanisms is specifically addressed in the RI-ISI process. Appendix VIII has been implemented by the industry through the performance demonstration (PD) program administered by EPRI. Irrespective of Appendix VIII implementation, Licensees maintain their responsibility to ensure that appropriate examination methods are applied in every case.

- What is the impact of the inspection result on the risk characterization?

  The inspection result has no impact on the risk characterization determination, as the risk characterization is consequence and failure potential based.

- Clarify plant modifications

  An example of a plant modification is provided below.

  To develop an example application for a safety-related system, the turbine driven auxiliary feedwater (TDAFW) steam supply system was chosen from a PWR plant. In order to set the stage for an example application, the original design at this plant is used as a starting point. During the licensing process and design basis HELB evaluations a plant modification was made to mitigate a steam break in the TDAFW pump area. This example evaluates the consequences without the modification in order to establish this example. The following summarizes the prior analysis and insights:

  - *Initiating Event Evaluation – The steam lines from steam generators B and C are pressurized with steam all the way to the TDAFW pump turbine. Based on PRA* *evaluations, breaks in the auxiliary building would be a high consequence due to environmental impacts.*

- *FSAR – In order to prevent the adverse environmental consequences (i.e., temperature and pressure) of a rupture in the steam line from propagating to other areas of the auxiliary building containing available shutdown equipment, and at the same time allowing the adverse environmental effects to vent to the atmosphere, the following structural design changes were initiated: the rollup door to the equipment access shaft was removed (this allowed discharging steam to vent up the shaft to the atmosphere); and the area containing the TDAFWP was isolated from safety-related equipment by the addition of walls and watertight doors to protect nearby equipment from flooding.*

Conclusion – if the original design (e.g., without the above modification) were evaluated using the RI-HELB methodology, the result would have been a high consequence. This would have required a modification to reduce the consequence to medium or low consequence. The above modification accomplishes this requirement.

RAI 6

EPRI Report 3002028939, Section 2.4, RC5 (with FAC): Ensure that FAC program addresses the most important locations (This moves the component to RC6 or RC7 depending on whether there are other degradation mechanisms besides FAC). The NRC staff need clarification how to address the piping other than the most important locations under the FAC program.

EPRI Response to RAI 6:

FAC programs predict, detect, and monitor FAC in plant piping and other pressure retaining components based on EPRI guidelines in the Nuclear Safety Analysis Center (NSAC)-202L (EPRI report 3002000563). Inspections are selected from a variety of sources including:

1. Results of lines analyzed using Predictive Plant Model (e.g., CHECWORKS)
2. “Susceptible-Not-Modeled” lines
3. Extrapolations of prior inspection results, commonly called “trending”
4. Plant experience
5. Operating experience
6. FAC-susceptible equipment
7. Engineering judgment
8. Entrance effect locations

Based on the results of the inspections, the remaining FAC service life of each component is evaluated to determine the next inspection or repair/replace the component. Results of the plant wall thickness measurements are used to enhance the analysis of the CHECWORKS analysis of uninspected components.

In accordance with NSAC-202L Appendix A, small-bore lines may not be included in the inspection program if all of the following apply:

1. The line is not part of a safety-related system

2. A failure would not cause a reactor shutdown or measurable loss of power

3. A failure can be readily isolated or controlled

4. A failure would not likely result in personnel injury

The NRC Generic Aging Lessons Learned (GALL) Report (NUREG-1801, Vol. 2, Rev. 1, ML052110006) relies on implementation of NSAC-202L to effectively manage the effects of FAC.

RAI 7

The degradation mechanisms used in EPRI TR 3002028939 are based on EPRI TR-112657, Revision B-A, and are amenable to mitigation by inspections.

EPRI TR-112657, states, in part,

> Now when considering the possible range of impacts that changes in inspection programs could conceivably have on rupture frequencies, the current service experience that is summarized in the preface to our response to RAI F-1 (on EPRI RI-ISI Methodology on TR-106706 [18]), this range is in turn limited by the fact that pipe failures can be caused by degradation mechanisms, severe loading conditions, or some combination of these. The vast majority of severe loading condition failures such as vibration fatigue, water hammer, frozen pipes and human error are not amenable to mitigation by inspections that are geared to find damage produced by an active degradation mechanism. (page 6-3)

Section 2.5.2 of EPRI TR-112657 also acknowledges that that vibrational fatigue should be treated outside the RI-ISI program.

a. Provide an explanation of how the degradation mechanisms which are not amenable to mitigation by inspection as described in EPRI TR-112657, such as but not limited to vibration fatigue, water hammer, flow induced vibration, etc. are addressed in EPRI TR 3002028939.

EPRI Response to RAI 7a:

The complete list of degradation mechanisms that will be assessed for RI-HELB applications, including DMs which are not amenable to mitigation by inspection such as vibration fatigue, water hammer, and flow induced vibration are identified and addressed as indicated in the response to RAI 7b.

b. ASME Section III Appendix W contains degradation mechanisms which are not included in EPRI TR 3002028939. Has EPRI performed an analysis/evaluation which concludes that all applicable ASME Section III Non-Mandatory Appendix W degradation mechanisms have been included?

EPRI Response to RAI 7b:

The degradation mechanisms (DM) included in Section 2.2.1 of TR 3002028939 for RI-HELB are the same DMs that have been assessed to determine DM susceptibility in RI-ISI (traditional/streamlined) and RI-BER applications in the existing operating reactor fleet for the past 25+ years with some minor changes incorporated over that period of time to reflect industry operating experience.

Some of the additional DMs contained in Appendix W, such as thermal embrittlement and environmentally assisted fatigue are addressed via aging management programs for the period of extended operation, and the effects of flow-induced vibration (FIV) on the structural integrity of any affected HELB components resulting from a power uprate are evaluated and addressed in plant LAR submittals following the guidance in RS-001 for extended /stretch power uprates, and RIS 2002-03 for measurement uncertainty recapture uprates.

Appendix W will be added as a reference to Section 2.2.2 of TR 3002028939 and considered during the “plant-specific service history” review. The following words and table will be added:

In addition, dependent upon the scope of the RI-HELB application and the affected piping systems or components, consideration should be given to any additional degradation mechanisms contained in ASME Section III Appendix W, as appropriate.

Degradation Mechanism	**Pipe Rupture Potential**
Intergranular stress corrosion cracking	Medium
Transgranular stress corrosion cracking	Medium
Irradiation-assisted stress corrosion cracking	Medium
Strain-induced stress corrosion cracking	Medium
Hydrogen embrittlement stress corrosion cracking	Medium
Thermal aging embrittlement	High
Irradiation embrittlement	Low
Hydrogen damage embrittlement	Medium
General corrosion (wastage)	Medium (CS); Low (SS)
Pitting corrosion	Medium
Crevice corrosion	Medium
Microbiologically influenced corrosion (fouling)	Medium
Environmentally-assisted fatigue	Medium
Flow-accelerated corrosion	High
Erosion	Medium
Fretting corrosion	Low
Thermal fatigue	Medium
Vibration loads	High
Water hammer loads	High[(1)]
Unstable fluid flow	High
Creep	Low

(1) Locations susceptible to a DM with a medium rupture potential shall be upgraded to high rupture potential if the pipe segment is subject to water hammer loads.

c. Table 2-5 of EPRI TR 3002028939, titled “EPRI system for evaluation of pipe rupture potential” provides high, medium and low pipe rupture potential based on degradation mechanisms for which a piping segment is susceptible to. What would the pipe rupture potential (high, medium or low) be for a degradation mechanism not ranked in Table 2-5 of EPRI TR 3002028939 from ASME Section III Non-Mandatory Appendix W such as but not limited to flow inducted vibration or vibrational fatigue?

EPRI Response to RAI 7c:

The pipe rupture potential for degradation mechanisms contained in ASME Section III Non-Mandatory Appendix W are listed in the table provided in the response for RAI 7b.

RCI 1

EPRI TR 3002028939, Section 4, "Conformance with Risk-Informed Decision-Making Principles," discusses that the RI-HELB methodology is not applicable to the reactor coolant pressure boundary. Is EPRI TR 3002028939 only applicable to ASME Class 2 and 3 and non-safety-related (NSR) piping or is EPRI TR 3002028939 applicable to ASME Class 1 piping as well? Provide clarification.

EPRI Response to RCI 1:

TR 3002028939 is only applicable to ASME Class 2 and 3 and non-safety-related (NSR) piping. It is not applicable to the ASME Class 1 reactor coolant pressure boundary as discussed under Principle 2 in Section 4.

The introduction and summary sections of TR 3002028939 will be revised to add the new wording below. This change is also addressed in the response to RAI 2.

This methodology is applicable to all Class 2, 3 and non-safety related systems with the exception of that portion of piping that lies within the break exclusion region (also known as the no break zone).

2. Probabilistic Risk Assessment (PRA) Licensing

Regulatory Basis: Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, states that the engineering analyses (including traditional and probabilistic analyses) conducted to justify a proposed licensing basis change should (1) be appropriate for the nature and scope of the change, (2) be based on the as-built and as-operated and maintained plant, and (3) reflect operating experience at the plant.

RAI 8

As discussed in Sections 2.1.2 and 2.1.3 of EPRI Report 3002028939, the consequence of pipe rupture is categorized in terms of conditional core damage probability (CCDP) and conditional large early release probability (CLERP). These risk metrics will be determined quantitatively using the plant-specific PRA models that reflect the as-built, as-operated plant, as discussed in Section 2 of the report. The staff noted that the scope of the PRA model may include both safety-related (SR) and NSR equipment. However, in the example provided in Section 3 of this TR, SR equipment is specifically emphasized while observations of NSR equipment are minimal. Some examples include:

a. Compartment pressurization pressure calculations for the turbine building produced no areas of concern with respect to SR equipment.

b. The effects of pipe whip on structures and walls and SR components were calculated for postulated main steam and feedwater pipe breaks in the turbine building.

c. It was confirmed that there is no SR equipment anchored or in close proximity to the shield wall.

Clarify the apparent differences in addressing SR equipment and NSR equipment that could be in the scope of the PRA. Discuss whether additional guidance is needed for the consequence evaluation to properly assess the impact of failed NSR equipment that could be in the scope of the PRA.

EPRI Response to RAI 8:

The cited text is taken from that portion of TR 3002028939 that summarizes the FSAR's summary description of the deterministic HELB analyses which is focused on safety related equipment. However, the RI-HELB evaluation must consider all equipment modeled in the PRA (SR and NSR) and is discussed later in Section 3.2.1 (Design and PRA Review) and Section 3.2.3 (Walkdown).

In addition, Section 2.1.4 of TR 3002028939 will be updated to add the following words.

The impact of a pipe failure and resulting interactions with other components (that is safety related, and non-safety related) are assessed as part of the consequence evaluation.

RAI 9

As discussed in Sections 2.1.2 and 2.1.3 of EPRI Report 3002028939, the consequence of pipe rupture is categorized in terms of CCDP and CLERP. These risk metrics will be determined quantitatively using the plant-specific PRA models that reflect the as-built, as-operated plant, as discussed in Section 2 of this TR. Section 2.1.3 of this TR indicated that one of the steps to quantitatively evaluate the consequences is: "Applicable Impacts are set to TRUE using basic events to simulate the impacts." The NRC staff noted that not all equipment, or associated failure modes due to HELB, may be modeled in the plant-specific PRA. Clarify how the licensee would evaluate the consequences if the target equipment (i.e., equipment damaged due to the direct or indirect effects of a pipe break) is not modeled in the PRA or if the target equipment is not currently subjected to a HELB but the piping evaluated becomes high energy as a result of the proposed change.

EPRI Response to RAI 9:

For equipment not modeled in the PRA, the PRA analyst would need to determine why the component was not modeled. There is plant equipment that remains not modeled in the PRA as it is not needed or credited for mitigation in the model.

Piping and equipment that was previously not subject to a HELB but later becomes a target of a HELB must be evaluated as part of the RI-HELB evaluation. This would include updating the HELB scenarios and/or developing new HELB scenarios for piping that is now high energy.

The following words will be added as the last paragraph under Step 2 (Failure Mode and Effects Analysis) in Section 2.

The RI-HELB methodology may be applied under a variety of circumstances. For example, to an existing plant design, a proposed plant modification (e.g. future power uprate) or a non-conforming condition. As such, the RI-HELB evaluation should reflect the intended application. As an example, a postulated moderate energy piping failure may not adversely affect a component within a certain local proximity. If however, as a result of a proposed power uprate the piping would become high energy, then the zone of influence of its postulated failure may increase, and new targets may be identified and adversely impacted. These new targets and the *impact of their failure due to the potential new HELB need to be reflected in the RI-HELB evaluation.*

Updating requirements for the plant-specific PRA are outside the scope of this document but it would be expected that the PRA would be updated to reflect the final disposition of the RI-HELB application (e.g. post power uprate conditions).

RAI 10

RG 1.174, Revision 3, Section 2.1.1.2 identifies seven considerations to evaluate how the proposed licensing basis change impacts defense-in-depth. The NRC staff noted that Section 4 of the EPRI Report 3002028939 provided a discussion of the five key risk- informed decisionmaking principles that risk-informed licensing basis changes are expected to meet. It is not clear in this TR how those seven considerations have been considered. Discuss how those seven considerations have been addressed when developing the RI-HELB evaluation method. Justify that the licensing basis change using the RI-HELB method is consistent with the defense-in-depth philosophy.

EPRI Response to RAI 10:

The seven considerations for evaluating the impact on defense-in-depth per Revision 3 of RG 1.174 are addressed as follows:

Existing Words in Section 4 of TR 3002028939

Principle 2. The proposed licensing basis change is consistent with the DID philosophy.

Piping systems in a nuclear power plant contribute to DID in two important ways: the piping of the reactor coolant pressure boundary provides one of the sets of barriers in the barrier DID arrangement. This barrier protects the release pathway from the reactor core to containment release pathways, and part of it is responsible for protecting against potential containment bypass pathways. This RI-HELB methodology is not applicable to the reactor coolant pressure boundary, and, as such, there is no adverse impact to this DID arrangement.

Second, piping contributes to DID in its role in the protection of the core through providing critical safety functions that require piping system integrity. As can be seen in the preceding sections, the RI-HELB methodology requires that PBFs that would fail a critical safety function be categorized as high safety-significant. These include those failures that would impact key inventory sources, plant-specific outliers that contribute to core damage or containment performance, and failure of the ultimate heat sink and components that can have intersystem impact.

New Words to be added under Principle 2 in Section 4 of TR 3002028939

Application of the RI-HELB methodology is consistent with the NRC's defense in depth philosophy as outlined in Section 2.1.1.2 of RG 1.174 Revision 3. The numbered list below describes how defense in depth is preserved in the context of the RI-HELB methodology:

1. *Preserve a reasonable balance among the layers of defense.*

   The RI-HELB methodology does not significantly reduce the effectiveness of a layer of defense that exists in the plant design before the implementation of the proposed licensing basis change. For plants licensed to 10CFRPart50;

- *Existing plant designs are robust with respect to surviving HELBs and existing plant programs minimize the likelihood of an HELB occurring. The RI-HELB methodology will not increase the likelihood of an HELB or create new significant HELBs.*

- *Prevention of a severe accident (core damage) if an event occurs. The RI-HELB Methodology will not significantly impact the availability and reliability of SSCs providing the safety functions that prevent HELBs from progressing to core damage. In fact, any HELB with a high consequence rank (CCDP > 1E-04 / CLERP > 1E-05, assuming a failure probability of 1.0) are required to undergo plant modification or use existing deterministic HELB guidance (e.g., NUREG-0800, SRP). As such, only those HELBs that can be shown to be lower risk (i.e., substantial redundancy and diversity to respond to the HELB) would benefit from the RI-HELB methodology.*

- *Containment of the source term if a severe accident occurs. The RI-HELB methodology will not significantly impact the containment function or SSCs supporting that function such as containment fan coolers and sprays. In fact, any HELB with a high consequence rank (CCDP > 1E-04 / CLERP > 1E-05, assuming a failure probability of 1.0) are required to undergo plant modification or use existing deterministic HELB guidance (e.g., NUERG-0800, SRP). As such, only those HELBs that can be shown to be lower risk (i.e., substantial redundancy and diversity to respond to the HELB) would benefit from the RI-HELB methodology.*

- *Protection of the public from any releases of radioactive material. The RI-HELB will not significantly reduce the effectiveness of the emergency preparedness program, including the ability to detect and measure releases of radioactivity, notify offsite agencies and the public, and shelter or evacuate the public as necessary.*

2. *Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.*

   The RI-HELB does not substitute programmatic activities for design features to an extent that significantly reduces the reliability and availability of design features to perform their safety functions without overreliance on programmatic activities. The RI-HELB reflects the as built / as operated plant and does not identify or require additional compensatory measures.

3. *Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.*

   The RI-HELB methodology does not significantly reduce the redundancy, independence, or diversity of systems. Via implementation of the RI-HELB methodology there are no changes to the plant that would result in an increase in the expected frequency of an HELB event.

   Any HELB events with a high consequence rank (CCDP > 1E-04 / CLERP > 1E-05, assuming a failure probability of 1.0) are required to undergo plant modification or use existing deterministic HELB guidance (e.g., NUREG-0800, SRP) thereby preserving redundancy, independence and diversity. Only those HELBs that can be shown to be lower risk (i.e., substantial redundancy and diversity to respond to the HELB) would benefit from the RI-HELB methodology.

4. *Preserve adequate defense against potential CCFs.*

   The RI-HELB does not significantly reduce defenses against CCFs that could defeat the redundancy, independence, or diversity of the layers of defense; fission product barriers;

and the design, operational, or maintenance aspects of the plant as the RI-HELB methodology does not;

- *Introduce a new potential CCF cause or event for which a defense is not in place. The RI-HELB methodology can be used on existing or postulated new HELB events, but it must be shown that it does not reduce existing defense strategies or introduce a new cause, event, or coupling factor. That is, significant reductions in defense strategies and significant CCF events would result in high consequence ranks using the RI-HELB methodology (i.e., CCDP > 1E-04 / CLERP > 1E-05). As such, these events would be required to be subject to plant modification or use of existing deterministic HELB requirements.*

- *Increase the probability or frequency of a cause or event that could cause simultaneous multiple component failures as these failures would result in high consequence ranks using the RI-HELB methodology (i.e., CCDP > 1E-04 / CLERP > 1E-05). As such, these events would be required to be subject to plant modification or use of existing deterministic HELB requirements.*

- *Introduce a new coupling factor for which a defense is not in place as these would result in high consequence ranks using the RI-HELB methodology (i.e., CCDP > 1E-04 / CLERP > 1E-05). As such, these events would be required to be subject to plant modification or use of existing deterministic HELB requirements.*

- *Weaken or defeat an existing defense against a cause, event, or coupling factor as these would result in high consequence ranks using the RI-HELB methodology (i.e., CCDP > 1E-04 / CLERP > 1E-05). As such, these events would be required to be subject to plant modification or use of existing deterministic HELB requirements.*

5. *Maintain multiple fission product barriers.*

   The RI-HELB methodology does not significantly reduce the effectiveness of the multiple fission product barriers.

   The RI-HELB Methodology does not create a significant increase in the likelihood or consequence of an event that simultaneously challenges multiple barriers and does not introduce a new event that would simultaneously impact multiple barriers. These types of events would result in high consequence ranks using the RI-HELB methodology (i.e., CCDP > 1E-04 / CLERP > 1E-05). As such, these events would be required to be subject to plant modification or use of existing deterministic HELB requirements.

6. *Preserve sufficient defense against human errors.*

   The RI-HELB methodology does not significantly increase the potential for or create new human errors that might adversely impact one or more layers of defense.

   The RI-HELB methodology reflects the as built / as operated plant. Human actions credited in the RI-HELB evaluation, if any, would need to address human reliability considerations consistent with the ASME/ANS PRA Standard including performance shaping factors (such as mental and physical demands and level of training).

7. *Continue to meet the intent of the plant's design criteria.*

   The proposed licensing basis change will not affect the plant's ability to meet the intent of the design criteria referenced in the licensing basis.

> *General Design Criteria 4 (GDC-4) requires that SSCs important to safety be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures. The RI-HELB Methodology is an alternative methodology for meeting (GDC-4), that is, an acceptable means for assessing and confirming that plant structures, systems, and components (SSCs) that are important to safety are adequate to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures.*

In summary, the RI-HELB methodology maintains consistency with the defense-in-depth philosophy as there is no significant impact on a single consideration (i.e., the intent of each defense-in-depth consideration is maintained).

RAI 11

RG 1.174, Section 3, indicates that the primary goal of performance monitoring is to ensure that no unexpected adverse safety degradation occurs because of the change(s) to the licensing basis. The RG states that the licensee should propose monitoring programs that adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's engineering evaluation and integrated decisionmaking that support the change to the licensing basis. Section 4 of the EPRI Report 3002028939 states, in part, "there are no unique aspects of the RI-HELB methodology insofar as monitoring requirements are concerned." However, in Section 2.4 of this TR, for HELB response strategies, for risk categories (RCs) RC2, RC4, and RC5 plant modifications are recommended to lower the consequences to change the RC. RC1, RC2, and RC3 require a 25 percent inspection population and RC4 and RC5 require a 10 percent inspection population based on the degradation mechanism. It is not clear to staff if current inspection programs are sufficient to monitor the performance consistent with RG 1.174, Revision 3, Section 3. Confirm that the licensee will include a description of the monitoring programs and their implementation ensuring that no unexpected adverse safety degradation occurs because of the change(s) to the licensing basis such that the RI-HELB evaluation conclusions would remain valid.

EPRI Response to RAI 11:

Monitoring programs put in place as a result of the application of the RI-HELB methodology shall be consistent with Section 3 of Regulatory Guide 1.174, Revision 3 to ensure that the RI-HELB evaluation conclusions remain valid. As such, Section 2.6 of TR 3002028939 will be updated as follows:

2.6 Performance Monitoring

> Monitoring programs put in place as a result of the application of this methodology shall be consistent with Section 3 of Regulatory Guide 1.174, Revision 3.
>
> When developing the performance-monitoring programs the following options should be considered:
>
> - If FAC applies to any of the scope, the FAC program and service history should be reviewed to confirm that they are in accordance with industry state of practice (for example, use of system health reports).
>
> - ISI inspection program.

- If any of the scope is accessible during power operations, operator walkdowns should be considered (for example, if not already in scope of walkdowns, consider adding).

- It may be possible to monitor inaccessible areas with ventilation performance, radiation monitors, and so forth.

RAI 12

As discussed in Sections 2.1.2 and 2.1.3 of EPRI Report 3002028939, the consequence of pipe rupture is categorized in terms of CCDP and CLERP. Section 2.1.1 of this TR discusses the identification of important equipment that could be impacted by the spatial (indirect) effects of a HELB. Section 2.1.3 of this TR describes how impacts to equipment are assessed using the PRA when quantitatively evaluating the consequences. The NRC staff note that in human reliability analysis (HRA), available time is an important factor when evaluating the human error probability (HEP) and available time may be reduced if a high-energy line break were to occur. A HELB can also result in a harsh environment such that local operator actions (e.g., isolation) are no longer feasible. Although this TR does discuss operator actions, it does not explicitly address potential impacts to HRA and the associated human error probabilities for operator actions. Clarify how adverse impacts to operator actions would be identified and assessed, including discussion of if HEPs would be adjusted, as part of the RI-HELB method.

EPRI Response to RAI 12:

Section 2.1.1 of TR 3002028939 will be updated as follows to highlight impact on possible local operator actions (changes are highlighted).

Spatial effects are an example of indirect effects caused by PBFs. These include the effects of high temperature, flood, jet impingement, and pipe whip on equipment located in the vicinity of the break. Spatial consequences of the break are determined based on the location of the analyzed break and the relative position of important equipment. Analyzed locations of the break should be consistent with locations analyzed in other spatial analyses performed for the plant (for example, internal flood [IF] analysis). The presence of important equipment or local operator actions credited in a specific location should be identified through these analyses and should be confirmed by a walkdown.

3. Piping and Head Penetrations Questions

Regulatory Basis: Title 10 of the CFR Part 50, Appendix A, General Design Criterion

(GDC) 4 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of the pipe ruptures postulated in SRP

Section 3.6.2. The staff reviews and approves the plant-specific piping system submitted from licensees and applicants to eliminate these dynamic effects. A staff approved leak- before-break (LBB) analysis permits licensees to remove protective hardware such as pipe whip restraints and jet impingement barriers, redesign pipe connected components, their supports and their internals, and other related changes in operating plants.

RAI 13

The text in EPRI TR 3002028939, Section 2.2.1 and Table 2-4 generate uncertainty as to how to assess the different degradation mechanisms. Section 2.2 of this TR uses a combination of a slightly modified version of Section 3.4.2.3 and Table 3-16 from the 1999 EPRI TR-T112657, Revision B-A. Notably, EPRI TR 3002028939 Table 2-4 contains some changes that reflect the changes in operating experience, EPRI guidance, and regulations since 1999. The criteria and susceptible regions given in EPRI TR 3002028939 Table 2-4 thus do not match the information for primary water stress corrosion cracking (PWSCC) in the text of EPRI TR 3002028939, Section 2.2.1. There are also differences between the table and the text for other degradation mechanisms.

a. Section 2.2.1 of this TR would be significantly clearer and more usable if the text and the table were consistent.

EPRI Response to RAI 13a:

Section 2.2.1 of TR 3002028939 will be updated (see Attachment 2) to establish consistency between the text and Table 2-4.

b. What updates to Table 2-4 of this TR would be appropriate given the operating experience and changes to regulations since 1999, including updates to documents like MRP-146, "Implementation Survey Summary Report (MRP-275)," which is now on Revision 2? MRP-146 is mentioned in the Appendix but not in the body of the proposed TR.

EPRI Response to RAI 13b:

Table 2-4 of TR 3002028939 will be updated (see Attachment 2) based on RI-ISI application and operating experience as shown below.

- A note has been added for TASCS scenarios in Table 2-4 concerning the potential for "leakage flow" and "turbulent penetration", to use the results from plant-specific MRP-146 programs.

- A note has been added in Table 2-4 to address that the methodology of EPRI Report 1011945 (Enhanced Crevice Corrosion Criteria for RI-ISI Evaluations) may be used to determine susceptibility to CC.

c. Page 26 – In the section on PWSCC for pressurized water reactors (PWRs), it states that piping and attachments (i.e., thermowells) are considered susceptible to PWSCC when they are fabricated from mill- annealed Alloy 600 base material and their associated welds Alloy 82/182 that is cold worked or cold worked and welded without subsequent stress relief, are exposed to primary water, and operate at high temperatures. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition," SRP 3.6.3, "Leak-Before-Break Evaluation Procedures" Revision 1, March 2007 (ADAMS Accession No. ML063600396), does not distinguish as to whether only cold worked Alloy 600 and 82/182 welds are susceptible to PWSCC and must be evaluated. It states that if Alloy 600/82/182 material is used, then PWSCC is a concern. However, if Alloy 690 and 52/152 welds are utilized, PWSCC is not a concern. Additional information needs to be provided if Alloy 600/82/182 material is used, specifically inspections, cladding/overlays or replacement with Alloy 690/52/152. There has been much operating experience (OE) since 1999, and this information should be part of the discussion.

EPRI Response to RAI 13c:

The mechanism attribute criteria for PWSCC in Section 2.2.1 will be revised consistent with updated Table 2-4 and the susceptible material discussion and other details will be removed. Updated Section 2.2.1 now states PWSCC is evaluated in accordance with the owner's existing PWSCC inspection program and, as applicable, the requirements endorsed by the regulatory authority having jurisdiction at the plant site (for example, 10CFR50.55a(g)(6)(ii)(F) dated June

21, 2011).

d. Page 26 – In the section on Pitting (PIT), it states that materials are susceptible to PIT, including austenitic stainless steels, nickel alloys, and carbon and low alloy steels. PIT susceptibility is a strong function of oxygen level and chloride level concentration. Please provide additional information on methods to address PIT susceptibility such as EPRI water chemistry procedures are in place.

EPRI Response to RAI 13d:

The mechanism attribute criteria for Pitting in Section 2.2.1 will be revised consistent with updated Table 2-4. Verifying conformance with the EPRI water chemistry guidelines is part of the DM assessment process.

RAI 14

a. EPRI TR 3002028939, Section 2.4, HELB Response Strategies, Page 36 – What actions are taken if the FAC program is not met? What components are modified? EPRI TR 3002028939, Section 2.4 states that for high-risk regions, twenty-five percent (25%) of the inspection population is performed. Please provide details as to what type of inspections will be performed.

EPRI Response to RAI 14a:

Application of RI-ISI uses NDE techniques that are designed to be effective for specific degradation mechanisms and examination locations. This inspection for cause approach involves identification of specific damage mechanisms that are likely to be operative, the location where they may be operative, and appropriate examination methods and volumes specific to address the damage mechanism. This approach provides assurance that risk significant locations selected for examination will be examined using effective methods.

Appendix VIII to ASME Section XI provides requirements for demonstrating the effectiveness of ultrasonic examination procedures and personnel for Section XI examinations. The scope of Appendix VIII does not include all damage mechanisms and locations relevant to RI-ISI such as FAC and MIC in augmented inspection programs—examination for these damage mechanisms is specifically addressed in the RI-ISI process. Appendix VIII has been implemented by the industry through the performance demonstration (PD) program administered by EPRI. Irrespective of Appendix VIII implementation, Licensees maintains their responsibility to ensure that appropriate examination methods are applied in every case.

If the FAC program is not met, then the requirements of the RI-HELB methodology are not met, and the methodology cannot be applied. Regarding what components are modified, see the response to RAI 5 for an example of a plant modification.

b. EPRI TR 3002028939, Section 2.4, HELB Response Strategies, Page 36 – What is the definition of “Most important locations”?

EPRI Response to RAI 14b:

The process outlined in the response to RAI 6 identifies how FAC programs predict, detect, and monitor for FAC in plant piping. Inspection locations are selected based on a number of considerations and factors. Based on the inspection results, the remaining service life of each FAC component is evaluated to ensure that the most important locations from a wear rate perspective are being managed and monitored.

c. EPRI TR 3002028939, Section 4, Conformance with Risk-Informed Decision-Making Principles, Page 71 – Principle 4, what type of examinations will be utilized to demonstrate that risk increases would be small and consistent with the intent of the NRC's policy statement on Risk-Informed Decisionmaking on safety goals for the operations of nuclear power plants.

EPRI Response to RAI 14c:

The examinations described in the response to RAI 14a coupled with the risk impact assessment described in Section 2.5 of TR 3002028939 requires that if there are any increases in risk that they be small and consistent with Reg Guide 1.174 acceptance criteria.

RAI 15

a. EPRI TR 3002028939, Section 5, Summary. Page 72 – Under Section 5, Summary, please provide additional information as to why other plant designs and related programs (i.e., material modifications) are outside the scope of this application.

EPRI Response to RAI 15a:

The intent of the statement, "other plant designs and related programs (e.g., determining the scope of equipment required to be within an environmental qualification program) are outside the scope of this application", is to clarify that the adoption of the RI-HELB methodology into a plant's design and licensing basis is plant specific. Any change to a plant's design and licensing basis (e.g., MRP, BWRVIP) would be addressed outside of the scope of TR 3003028939.

b. NUREG-0800, Section 3.6.3, Revision 1, March 2007, states that an evaluation over the entire life of the plant for the plant piping system include environmental conditions. Does this apply to EPRI TR 3002028939? Please explain the reasoning.

EPRI Response to RAI 15b:

Section 2.6 of TR 3003028939 will be updated to add the following words. In addition, Figure 2-2 (RI-HELB methodology overview) will be updated requiring the adjustment of HELB response strategies if delta risk is not met.

In addition, consistent with RG 1.174, risk-informed decision-making process, the licensee is required to review changes to the plant, operational practices, applicable plant and industry operational experience over the entire life of the plant including environmental conditions, and, as appropriate, update the PRA and the RI-HELB evaluations.

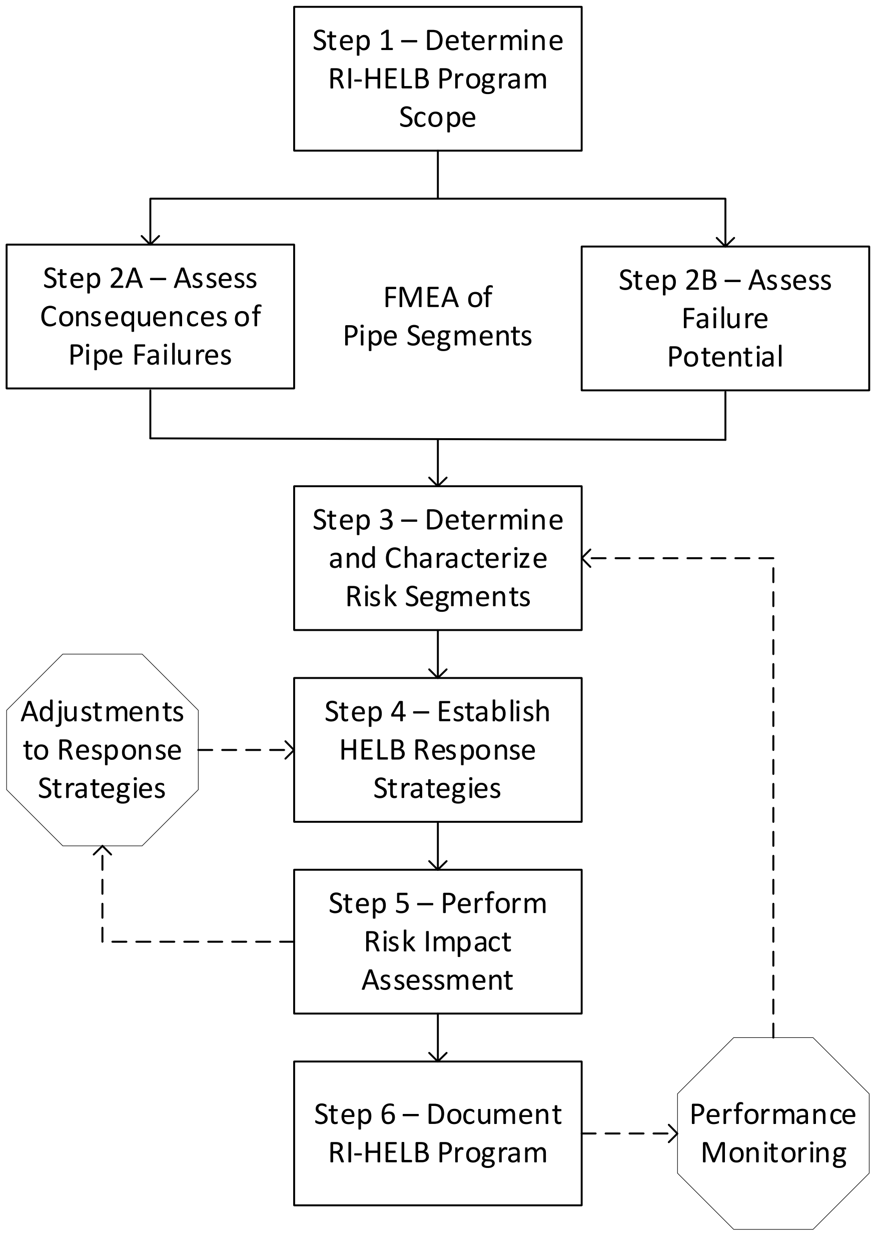

Figure 2-2. RI-HELB methodology overview

c. Balance of Plant reviews include capability, reliability and sensitivity of the reactor coolant pressure boundary leakage detection systems inside containment. Please describe the leakage detection systems and whether they meet RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” and NUREG-0800, Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection,” Revision 1, March 2007 (ADAMS Accession No. ML070610277).

EPRI Response to RAI 15c:

Leakage detection is not credited. To clarify this, Section 2.1 of TR 3003028939 will be updated as follows:

The CoF evaluation assumes a failure probability of 1.0 thus there is no credit taken for leakage detection.

4. Long-Term Operations and Modernization

Regulatory Basis: Title 10 of the CFR Section 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," describe a required program for qualifying the electric equipment. The qualification program must include and be based on the following:

(1) Temperature and pressure. The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design-based accident during or following which this equipment is required to remain functional.

(2) Humidity. Humidity during design-based accidents must be considered.

(3) Chemical effects. The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) Radiation. The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) Aging. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.

(6) Submergence (if subject to being submerged).

(7) Synergistic effects. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

(8) Margins. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

RAI 16

EPRI TR 3002028939 states, in part, that "…other related programs (e.g., determining the scope of equipment required to be within an environmental qualification program) are outside the scope of this application." However, changes to considerations and conditions for pipe breaks (e.g., location, severity, etc.) inherently could impact10 CFR 50.49 environmental qualification (EQ) zones (which establish the environmental parameters for determining which equipment needs to be qualified and to what threshold) and reduce or remove requirements for equipment qualification (either 10 CFR 50.49 or 10 CFR 50, Appendix A, GDC 4). Please provide additional explanation as to why 10 CFR 50.49 is outside the scope of this TR since NRC staff approval of EPRI TR 3002028939 could have a direct or indirect impact on the equipment that is currently required to be qualified per 10 CFR 50.49.

EPRI Response to RAI 16:

The statement, “other plant designs and related programs (e.g. determining the scope of equipment required to be within an environmental qualification program) are outside the scope of this application” was intended to clarify that the adoption of a RI-HELB methodology into a plant’s design and licensing basis is plant specific. Any change to a plant’s design and licensing basis—including any impact from adopting the RI-HELB methodology on the equipment that is currently required to be qualified per 10 CFR 50.49–would be addressed outside of the scope of EPRI TR 3003028939. The basis for this approach was intended to focus the regulatory review on the risk-informed methodology, which is limited to determining the risk significance of postulated piping failures and appropriate plant response strategies (see Sections 1 and 5 of TR 3003028939). As a result, any changes made specific to the EQ program, or other programs, would be addressed on a plant-by-plant basis as part of the existing licensing process.

The scope of equipment required to be within an environmental qualification program could change as a result of adopting a RI-HELB approach. Plant modifications to lower the risk significance of a postulated HELB has the potential to add equipment to the scope of a § 50.49 EQ program. The application of a RI-HELB approach could also eliminate the need to environmentally qualify electric equipment only credited to detect and/or mitigate a postulated high energy pipe rupture, if the postulated high energy pipe rupture is categorized by the RI-HELB plant specific evaluations as a low-risk HELB (e.g., RC6 or RC7). For electric equipment credited in multiple HELB scenarios, the most risk-significant scenario should be used in determining HELB response strategies.

RAI 17

EQ requires consideration of design-basis events and accidents, including HELB. Please explain why EQ requirements, guidance, and expectations for considering and calculating HELBs (e.g., NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” Revision 1 (ADAMS Accession No. ML031480402)) were not addressed in the TR.

EPRI Response to RAI 17:

The response to RAI 16 clarifies that specific changes made to the EQ program as a result of adopting the methodology in EPRI TR 3003028939 would be addressed on a plant-by-plant basis as part of the License Amendment Request. The adoption of a risk-informed HELB methodology into the current design and licensing basis of a plant would be addressed in the same manner as any other change that affects fundamental inputs to the EQ program.

For currently operating plants, EQ programs would continue to address the postulated HELBs following existing deterministic HELB requirements in accordance with the response strategies delineated in EPRI TR 3003028939 for high consequence (RC1, RC2, RC4) pipe rupture events, unless additional measures are taken to reduce the consequence of failure to medium or low (See Figure 2-8). The determination of the compartmental response (e.g., temperature, pressure, humidity, & flooding) would remain consistent with existing deterministic methods along with accounting for changes in break locations or changes in mass & energy releases. HELBs classified as low risk (RC6 & RC7) do not need to consider the requirements of 10 CFR 50.49– since environmentally induced common cause failures have been addressed as part of the risk classification process.

ATTACHMENT 2 – UPDATED SECTION 2.2.1 AND TABLE 2-4

DM Evaluation

As covered previously, the RI-HELB methodology requires the applicable DM(s), if any, to be identified for the in-scope piping under evaluation. The DMs to be assessed are listed as follows, summarized in Table 2-4, and described in the subsequent paragraphs:

- Thermal stratification, cycling, striping (TASCS)
- Thermal transient (TT)
- Intergranular stress corrosion cracking (IGSCC)
- Transgranular stress corrosion cracking (TGSCC)
- External chloride stress corrosion cracking (ECSCC)
- Primary water stress corrosion cracking (PWSCC)
- Microbiologically-influenced corrosion (MIC)
- Pitting (PIT)
- Crevice corrosion (CC)
- Erosion-cavitation (E-C)
- Flow-accelerated corrosion (FAC)

Thermal Fatigue

Mechanism Description

Thermal fatigue can occur as a result of alternating stresses caused by thermal cycling of a component resulting in accumulated fatigue usage and leading to crack initiation and growth.

Attribute Criteria

Austenitic and carbon steel pipe segments are evaluated for the potential for degradation from TASCS and TT, as indicated in the following:

- **TASCS**. Pipe segments where the diameter is greater than 1 inch (25.4 mm), the slope of the segment less than 45° from the horizontal, and the segment experiences hot/cold fluid mixing with a temperature difference greater than 50°F (28°C) are considered susceptible to degradation from TASCS. Examples of such fluid mixing include leakage past valves separating hot and cold fluids and swirl penetration by hot fluid into previously cold pipe segments. When these criteria are exceeded, an additional evaluation can be performed to determine whether the Richardson number is greater than 4.0.

Note: For TASCS scenarios in Table 2-4 concerning the potential for "leakage flow" and "turbulent penetration" the results from plant-specific MRP-146 programs may be used.

- **TT**. Austenitic stainless steel and carbon steel pipe segments having operating temperatures greater than 270° or 220°F (132°C or 104°C), respectively, where there is a relatively rapid cold (hot) water injection into a previously hot (cold) pipe segment with a temperature difference greater than 200°F (93°C) or 150°F (66°C), respectively, are considered susceptible to degradation from TT. When these temperature differences are exceeded, additional evaluations can be performed to determine whether the temperature difference is greater than the allowable temperature difference.

Stress Corrosion Cracking

Stress corrosion cracking (SCC) encompasses several mechanisms, as follows:

Intergranular Stress Corrosion Cracking

Mechanism Description

IGSCC results from a combination of sensitized materials (caused by a depletion of chromium in regions adjacent to the grain boundaries in weld heat-affected zones [HAZs]), high applied stress (including residual welding stress), and a corrosive environment (high level of oxygen and/or other contaminants).

Attribute Criteria

BWRs. Piping within the scope of the RI-ISI evaluation is typically compared to piping included in the existing BWR plant IGSCC inspection program per NRC Generic Letter 88-01 or EPRI BWRVIP-075. Piping in the RI-ISI evaluation scope should be identified as susceptible to IGSCC for the purpose of RI-ISI evaluation if it is inspected as part of the existing plant IGSCC inspection program.

PWRs. Welds and HAZs in austenitic stainless steel PWR piping under tensile stress and containing fluid with a high dissolved oxygen content and initiating contaminants (e.g., thiosulfate, fluoride, or chloride) are considered susceptible to degradation from IGSCC. For locations operating at greater than 200°F (93°C), the presence of initiating contaminants is not required for IGSCC to be active.

Transgranular Stress Corrosion Cracking

Mechanism Description

TGSCC is stress corrosion cracking that occurs through the grains of the material and usually in the presence of halogens. It is not necessarily associated with a particular metallurgical condition, such as grain boundary sensitization, but is affected by high local residual stresses, such as those caused by welding or local cold work.

Attribute Criteria

Austenitic stainless steel pipe segments under tensile stress, operating at greater than 150°F (66°C), and containing fluid with a high dissolved oxygen content and halides (e.g., fluoride or chloride) are considered susceptible to degradation from TGSCC.

External Chloride Stress Corrosion Cracking

Mechanism Description

ECSCC is stress corrosion cracking due to chloride intrusion onto the outside surface of a pipe segment, either through direct contact with chlorinated water or through the leaching of chloride from wet insulation.

Attribute Criteria

Austenitic stainless steel pipe segments under tensile stress and operating at greater than 150°F (66°C) are considered susceptible to degradation from ECSCC if the outside surface is either (1) an within five diameters of a probable leak path (e.g., valve stems) and covered with nonmetallic insulation not in compliance with RG 1.36, or (2) exposed to wetting from concentrated chloride-bearing environments (e.g., sea water, brackish water, or brine).

Primary Water Stress Corrosion Cracking

Mechanism Description

PWSCC can occur in nozzles, welds, and HAZs when high-temperature primary water is present in combination with a susceptible material and high tensile stress.

Attribute Criteria

PWSCC is evaluated in accordance with the owner’s existing PWSCC inspection program and, as applicable, the requirements endorsed by the regulatory authority having jurisdiction at the plant site (for example, 10CFR50.55a(g)(6)(ii)(F) dated June 21, 2011).

Localized Corrosion

Local corrosion encompasses several mechanisms, as follows:

Microbiologically Influenced Corrosion

Mechanism Description

Microbes, primarily bacteria, have been found to cause widespread damage to low-alloy and carbon steels. Similar damage has also been found at welds and HAZs for austenitic stainless steels.

Attribute Criteria

Pipe segments are considered susceptible to degradation from MIC if they operate at less than 150°F (66°C), encounter low or intermittent flow (including stagnant flow), have a fluid pH less than 10, and where the fluid either contains organic material (such as a raw water system) or is otherwise not treated with biocides.

Pitting

Mechanism Description

PIT is a form of localized attack on exposed surfaces with greater corrosion rates at some locations than at others. High local concentrations of impurity ions, such as chlorides and sulfates, tend to concentrate in oxygen-depleted pits, giving rise to a potentially concentrated aggressive solution in this zone.

Attribute Criteria

Pipe segments that encounter low flow and contain fluid with a high dissolved oxygen content and initiating contaminants (e.g., fluoride or chloride) are considered susceptible to degradation from PIT.

Crevice Corrosion

Mechanism Description

CC is the electrochemical reaction caused by an oxygenated medium within a piping system. Regions containing crevices (narrow gaps) that can result in oxygen depletion and a relatively high concentration of chloride ions or other impurities are considered susceptible to CC.

Attribute Criteria

Locations where a geometric crevice (e.g., a thermal sleeve) exist are considered susceptible to degradation from CC if they operate at less than 150°F (66°C) and contain fluid with a high dissolved oxygen content.

Note: The methodology of EPRI Report 1011945 (Enhanced Crevice Corrosion Criteria for RI-ISI Evaluations) may be used to determine susceptibility to CC.

Flow Sensitive

These mechanisms consist of Erosion-Cavitation and FAC.

Erosion-Cavitation

Mechanism Description

E-C represents degradation caused by turbulent flow conditions, which erode (wear away the metal of) the pipe wall by cavitation. Cavitation damage is the result of the formation and instantaneous collapse of small voids within fluid subjected to rapid pressure and velocity changes as it passes through a region where the flow is restricted (e.g., a valve, pump, or orifice).

Attribute Criteria

Pipe segments downstream of potential cavitation sources are considered susceptible to degradation from E-C if they encounter flow for more than 100 hours per year at a temperature less than 250°F (121°C) and a fluid velocity greater than 30 feet per second. When these criteria are exceeded, an additional evaluation can be performed to determine whether the difference between the static pressure and vapor pressure, divided by the pressure differential across the cavitation source, is less than 5.

Flow-Accelerated Corrosion

Mechanism Description

FAC is a complex phenomenon that exhibits attributes of erosion and corrosion in combination. Factors that influence whether FAC is an issue are velocity, dissolved oxygen, pH, moisture content of steam, and material chromium content.

Attribute Criteria

FAC is evaluated in accordance with the owner’s existing FAC inspection program.

Table 2-4. DM criteria and susceptible regions

DM		Criteria	Susceptible Regions
TF	TASCS	NPS >1 in., and pipe segment has a slope <45° from horizontal (includes elbow or tee into a vertical pipe), and potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or potential exists for leakage flow past a valve (that is, in-leakage, out-leakage, cross-leakage) allowing mixing of hot and cold fluids (Note: use results of plant-specific MRP-146 program), or potential exists for convection heating in dead-ended pipe sections connected to a source of hot fluid, or potential exists for two phase (steam/water) flow, or potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow (Note: use results of plant-specific MRP-146 program), and Calculated or measured ΔT >50°F, and Richardson number >4.0	Nozzles, branch pipe connections, safe ends, welds, HAZs, base metal, and regions of stress concentration

Table 2-4 (continued). DM criteria and susceptible regions

DM		Criteria	Susceptible Regions
TF (continued)	TT	operating temperature >270°F for stainless steel, or operating temperature >220°F for carbon steel, and potential for relatively rapid temperature changes including: cold fluid injection into hot pipe segment, or hot fluid injection into cold pipe segment, and $\lvert \Delta T \rvert$ >200°F for stainless steel, or $\lvert \Delta T \rvert$ >150°F for carbon steel, or $\lvert \Delta T \rvert > \Delta T$ allowable (applicable to both stainless and carbon)	
SCC	IGSCC (BWR)	Evaluated in accordance with existing plant IGSCC program according to NRC Generic Letter 88-01 or EPRI BWRVIP-075	Welds and HAZs
	IGSCC (PWR)	Austenitic stainless steel (carbon content ≥0.035%), and Operating temperature >200°F, and Tensile stress (including residual stress) is present, and Oxygen or oxidizing species are present **OR** Operating temperature <200°F, the preceding attributes apply, and Initiating contaminants (for example, thiosulfate, fluoride, or chloride) are also required to be present	
	TGSCC	Austenitic stainless steel, and Operating temperature >150°F, and Tensile stress (including residual stress) is present, and Halides (for example, fluoride or chloride) are present, and Oxygen or oxidizing species are present	Base metal, welds, and HAZs

Table 2-4 (continued). DM criteria and susceptible regions

DM		Criteria	Susceptible Regions
SCC (continued)	ECSCC	Austenitic stainless steel, and Operating temperature >150°F, and Tensile stress is present, and An outside piping surface is within five diameters of a probable leak path (for example, valve stems) and is covered with nonmetallic insulation that is not in compliance with RG 1.36, **OR** Austenitic stainless steel, and Tensile stress is present, and an outside piping surface is exposed to wetting from concentrated chloride-bearing environments (that is, sea water, brackish water, or brine)	Base metal, welds, and HAZs
	PWSCC	Evaluated in accordance with the owner's existing PWSCC inspection program and, as applicable, the requirements endorsed by the regulatory authority having jurisdiction at the plant site (e.g., 10CFR50.55a(g)(6)(ii)(F) dated June 21, 2011)	Nozzles, welds, and HAZs without stress relief
Localized Corrosion	MIC	Operating temperature <150°F, and Low or intermittent flow, and pH <10, and Presence/intrusion of organic material (for example, raw water system), or Water source is not treated with biocides	Fittings, welds, HAZs, base metal, dissimilar metal joints (for example, welds and flanges), and regions containing crevices
	PIT	Potential exists for low flow, and Oxygen or oxidizing species are present, and Initiating contaminants (for example, fluoride or chloride) are present	

Table 2-4 (continued). DM criteria and susceptible regions

DM		Criteria	Susceptible Regions
Flow Sensitive	CC	Crevice condition exists (that is, thermal sleeves), and Operating temperature >150°F, and Oxygen or oxidizing species are present	Regions containing crevices
	E-C	Cavitation source, and Operating temperature <250°F, and Flow present >100 h/yr, and Velocity > 30 ft/s, and $(P_d - P_v) / \Delta P < 5$	Fittings, welds, HAZs, and base metal
	FAC	Evaluated in accordance with existing plant FAC program	According to plant FAC program