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MFN 07-060

Docket No. 52-010

January 23, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 77 Related to ESBWR Design Certification Application – LBB Discussion and Circulating Water System – RAI Numbers 3.6-22 and 14.3-93**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

Reference:

1. MFN 06-391, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 77 Related to ESBWR Design Certification Application,* October 11, 2006

Enclosure:

1. MFN 07-060 – Response to Portion of NRC Request for Additional Information Letter No. 77 Related to ESBWR Design Certification Application – LBB Discussion and Circulating Water System – RAI Numbers 3.6-22 and 14.3-93

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRFs	0000-0061-0794 RAI 3.6-22
		0000-0061-8776 RAI 14.3-93

Enclosure 1

MFN 07-060

Response to Portion of NRC Request for

Additional Information Letter No. 77

Related to ESBWR Design Certification Application

LBB Discussion and Circulating Water System

RAI Numbers 3.6-22 and 14.3-93

NRC RAI 3.6-22

We are not aware of any plans to use Leak Before Break (LBB) evaluation techniques for the ESBWR however LBB is described or referred to in several areas of the DCD. This primarily involves ESBWR DCD Tier 2, Revision 1,Section 3.6.3 and Appendix 3E, but it also includes references to LBB in ESBWR DCD Tier 1, Revision 1 and in other areas of the ESBWR DCD Tier 2, Revision 1. Please describe your plans to use LBB or remove discussions related to LBB from the DCD if it is not to be part of the certified design.

GE Response

LBB will not be used in ESBWR design due to conservative piping stress limits. LBB will be removed from the DCD and the Appendix 3E content will be removed.

DCD Impact

DCD Tier 2 Subsection 1.2.2.15.1, Table 1.9-3, Table 1.9-20, Table 1.10-1, Table 3.2-1, Subsection 3.6, Subsection 3.6.3, Subsection 3.6.6, Table 3.9-2, Appendix 3E will be revised as noted in the attached markup.

NRC RAI 14.3-93

Provide ITAAC in DCD Tier 1 for the Circulating Water System or provide the rational for not including it. The acceptance criteria should include an as-built inspection of the system, testing to demonstrate the system is functioning properly, and testing of the alarms and controls in the main control room.

GE Response

This RAI requests Tier 1 or ITAAC changes and/or additions; therefore, it has been reviewed per GE internal Tier 1 content determination guidelines, which are based on draft SRPs 14.3 through 14.3.11 and DG-1145 (as of July 31, 2006). This response is provided consistent with those guidelines. The following includes some of regulatory bases used to develop the GE Tier 1 content determination guidelines.

Draft SRP 14.3, Appendix A, Section IV, first paragraph states, "*While the Tier 1 information must address the complete scope of the design to be certified, the amount of design information is proportional to the safety-significance of the structures and systems of the design.*" Therefore, a graded approach, based on safety functions, is used for determining the amount of detail in the Tier 1 design descriptions (DD) and ITAAC.

Draft SRP 14.3, Appendix A, Section IV, Item B.1 states, "*The design descriptions (DD) address the* ***most safety-significant*** *aspects of each of the systems of the design, and were derived from the detailed design information contained in Tier 2. The applicant should put the* ***top-level design features and performance characteristics that were the most significant to safety*** *in the Tier 1 design descriptions. The level of detail in Tier 1 governed by a graded approach to the SSCs of the design, based on the safety significance of the functions they perform.*" Therefore, not all safety-significant systems are required to be described in Tier 1.

As delineated in 10 CFR 50.69, plants are expected to have "*safety-related SSCs that perform low safety significant functions*, and thus, those SSCs should not be considered as *most safety-significant*. However, for the ESBWR, the Tier 1 change determination process conservatively assumes that all safety-related functions qualify as *safety-significant*. For a passive plant, like the ESBWR, the safety-significant nonsafety-related SSCs are determined by applying the Regulatory Treatment of Non-Safety Systems (RTNSS) criteria. (The safety-significant nonsafety-related SSCs are addressed in Tier 2, Appendix 1D.) Therefore, the "*safety-significant aspects*" of the ESBWR involve the performance of all safety-related functions and the RTNSS functions of the nonsafety-related equipment. By exclusion, all other SSC functions are not safety-significant, and therefore, are not required to be addressed in Tier 1.

Draft SRP 14.3, Appendix A, Section IV, Item B.2, sixth paragraph states "*The level of detail specified in the ITAAC should be commensurate with the safety significance of the functions and bases for that SSC.*" Therefore, the ITAAC for a system should be based on the safety significant information in the DD.

The Circulating Water System is not safety-related nor does it qualify as a RTNSS system, and thus, it is not safety-significant. Therefore, per the guidance in draft SRPs 14.3 – 14.3.11 and DG-1145, only the system name, without a DD or ITAAC, is required to be included in Tier 1. However, Tier 1 currently contains some DD information without ITAAC, and therefore, already contains more information than is required. Consequently, no additional information for the Circulating Water System is required to be contained or added in Tier 1.

DCD Impact

No DCD change will be made in response to this RAI.

[NOTES IN GREEN HIGHLIGHT ABOVE SUBSECTION HEADINGS THAT CONTAIN CHANGES]

1.2.2.15.1 Containment System [RAI 3.6-22]

The ESBWR containment, centrally located in the Reactor Building, features the same basic pressure suppression design concept previously applied in over three decades of BWR power generating reactor plants. The containment consists of a steel-lined, reinforced concrete containment structure in order to fulfill its design basis as a fission product barrier at the pressure conditions associated with a postulated pipe rupture.

Main features include the upper and lower drywell surrounding the RPV and a wetwell containing the suppression pool that serves as a heat sink during abnormal operations and accidents.

The containment is constructed as a right circular cylinder set on the reinforced concrete base mat of the reactor building. The drywell and wetwell design conditions are provided in Section 6.2.

The drywell comprises two volumes: an upper drywell volume surrounding the upper portion of the RPV and housing the steam and feedwater piping, the SRVs, GDCS pools, main steam drain piping and upper drywell coolers; and a lower drywell volume surrounding the lower portion of the RPV, housing the FMCRDs, neutron monitoring system, equipment platform, lower drywell coolers and two drywell sumps. The drywell top opening is enclosed with a steel head removable for refueling operations.

The gas space above the suppression pool serves as the LOCA blowdown reservoir for the upper and lower drywell nitrogen and non-condensable gases that pass through the twelve drywell-to-wetwell vertical vents, each with three horizontal vents located below the suppression pool surface. The suppression pool water serves as the heat sink to condense steam released into the drywell during a LOCA or steam from SRV actuations.

Access into the upper and lower drywells is provided through a double sealed personnel lock and an equipment hatch. The equipment hatch is removable only during refueling or maintenance outages. A hatch located in the Reactor Building provides access into the wetwell.

During plant startup, the Containment Inerting System, in conjunction with the containment purge system and the drywell cooling fans, is utilized to establish an inert gas environment in the containment with nitrogen to limit the oxygen concentration. This precludes combustion of any hydrogen that might be released subsequent to a LOCA. After the containment is inerted and sealed for plant power operation, small flows of nitrogen gas are added to the drywell and the wetwell as necessary to keep oxygen concentrations below 4% and to maintain a positive pressure for preventing air in-leakage. High-pressure nitrogen is also used for pneumatic controls inside the containment to preclude adding air to the inert atmosphere.

The containment structure has the capability to maintain its functional integrity at the pressures and temperatures that could follow a LOCA pipe break postulated to occur simultaneously with loss of off-site power. The containment structure is designed to accommodate the full range of loading conditions associated with normal and abnormal operations including LOCA-related design loads in and above the suppression pool (including negative differential pressure between the drywell, wetwell and the remainder of the Reactor Building), and safe shutdown earthquake (SSE) loads.

The containment structure is protected from, or designed to withstand, fluid jet forces associated with outflow from the postulated rupture of any pipe within the containment.

The containment design does not consider~~s~~ ~~and~~nor utilize~~s~~ leak-before-break (LBB) applicability ~~only in~~with regard to protection against dynamic effects associated with a postulation of rupture in high-energy piping. ~~Subsection 3.6.3 and Appendix 3E describe the implementation of the LBB approach for excluding design against the dynamic effects from postulation of breaks in high energy piping.~~ Protection against the dynamic effects from the piping systems not qualified by the exclusion from the dynamic effects caused by their failure is provided for the drywell structure. The drywell structure is provided protection against the dynamic effects of plant-generated missiles (Section 3.5).

The containment structure has design features to accommodate flooding to sufficient depth above the top of active fuel to permit safe removal of fuel assemblies from the reactor core after a postulated design basis accident (DBA).

The containment structure is configured to channel flow from postulated pipe ruptures in the drywell to the suppression pool through vents submerged in the suppression pool, which are designed to accommodate the energy of the blowdown fluid.

The containment structure and penetration isolation system, with concurrent operation of other accident mitigation systems, are designed to limit fission product leakage during and following a postulated DBA to values well below leakage calculated for allowable off-site doses.

In accordance with Appendix J to 10 CFR 50, the containment design includes provisions for testing at a reduced pressure below the peak calculated DBA LOCA pressure to confirm containment leakage is below the design limit. Special testing capabilities are provided during outages to measure local leakage, such as individual air locks, hatches, drywell head, piping, electrical and instrument penetrations. Other features are provided to measure isolation valve leakage and to measure the integrated containment leak rate. Results from the individual and integrated preoperational leak rate tests are recorded for comparison with subsequent periodic leak rate test results.

The design value for a maximum steam bypass leakage between the drywell and the wetwell through the diaphragm floor including any leakage through the wetwell-to-drywell vacuum breakers is limited. Satisfying this limit is confirmed by initial preoperational tests as well as by periodic tests conducted during refueling outages. These tests are conducted at differential pressure conditions between the drywell and wetwell that do not clear the drywell-to-wetwell horizontal vents.

A watertight barrier is provided between the open reactor and the drywell during refueling. This enables the reactor well to be flooded prior to removal of the reactor steam separator, dryer assembly and to facilitate underwater fuel handling operations. Piping, cooling air ducts and return air vent openings in the reactor well platform must be removed, vents closed and sealed watertight before filling the reactor well with water. The refueling bellows assembly is provided to accommodate the movement of the vessel caused by operating temperature variations and seismic activity.

Containment isolation is accomplished with inboard and outboard isolation valves on each piping penetration that are signaled to close on predefined plant parameters. Systems performing a post-LOCA function are capable of having their isolation valves reopened as needed.

Drywell coolers are provided to remove heat released into the drywell atmosphere during normal reactor operations

Table 1.9-3 [RAI 3.6-22, only section 3 shown in this table]

Summary of Differences from SRP Section 3

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Section/Subsection Where Discussed
3.2.1		None	
3.2.2		None	
3.3.1		None	
3.3.2		None	
3.4.1		None	
3.4.2		None	
3.5.1		None	
3.5.2		None	
3.5.3		None	
3.6.1 and 3.6.2	~~II – Postulated pipe rupture.~~	~~Large bore piping can utilize leak before break option as provided in GDC 4 October 27, 1987, "Modification of General Design Criterion 4."~~None	~~3.6 and 3.6.3~~
3.7.1 and 3.7.3	II- Two earthquakes, the SSE and the OBE shall be considered in the design.	The ESBWR will be based on a single earthquake (SSE) design.	3.7.1 and 3.7.3
3.7.2		None	
3.7.3	II.9—For multiply supported equipment use envelope RS and;	Independent Support Motion Response Spectrum methods acceptable for use.	3.7.3.9
3.7.3	Combine responses from inertia effects with anchor displacements by absolute sum.	Combine responses from inertia effects with anchor displacements by SRSS.	3.7.3.9
3.7.3	II.2 – Determination of number of OBE cycles	The ESBWR is based on a single earthquake (SSE) design, two SSE events with 10 peak stress cycles per event are used.	3.7.3.2
3.7.4		None	

Table 1.9-3 [RAI 3.6-22, only section 3 shown in this table]

Summary of Differences from SRP Section 3

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Section/Subsection Where Discussed
3.8.1		None	
3.8.2		None	
3.8.3		None	
3.8.4		None	
3.8.5		None	
3.9.1		None	
3.9.2		None	
3.9.3		None	
3.9.4		None	
3.9.5		None	
3.9.6		None	
3.10		None	
3.11		None	

Table 1.9-20 [RAI 3.6-22, only section 3 shown in this table]

NRC Standard Review Plans and Branch Technical Positions Applicability to ESBWR

SRP No.	SRP Title or BTP	Appl. Rev.	Issued Date	ESBWR Appli-cable?	Comments
Chapter 3 Design of Structures, Components, Equipment, and Systems					
3.2.1	Seismic Classification	1	07/1981	Yes	
3.2.2	System Quality Group Classification	1	07/1981	Yes	
	Appendix A (Formerly BTP RSB 3-1)	1	07/1981	Yes	
	Appendix B (Formerly BTP RSB 3-2)	1	07/1981	Yes	
	Appendix C	1	07/1981	No	PWR Only
	Appendix D	1	07/1981	—	Never issued
3.3.1	Wind Loadings	2	07/1981	Yes	
3.3.2	Tornado Loadings	2	07/1981	Yes	
3.4.1	Flood Protection	2	07/1981	Yes	
3.4.2	Analysis Procedures	2	07/1981	Yes	
3.5.1.1	Internally Generated Missiles (Outside Containment)	2	07/1981	Yes	
3.5.1.2	Internally Generated Missiles (Inside Containment	2	07/1981	Yes	
3.5.1.3	Turbine Missiles	2	07/1981	Yes	
3.5.1.4	Missiles Generated by Natural Phenomena	2	07/1981	Yes	
	BTP ASB 3-2	2	07/1981	—	Superseded by RG 1.117
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	07/1981	Yes	
3.5.1.6	Aircraft Hazards	2	07/1981	Yes	
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2	07/1981	Yes	
3.5.3	Barrier Design Procedures	1	07/1981	Yes	
	Appendix A	0	07/1981	Yes	

Table 1.9-20 [RAI 3.6-22, only section 3 shown in this table]

NRC Standard Review Plans and Branch Technical Positions Applicability to ESBWR

SRP No.	SRP Title or BTP	Appl. Rev.	Issued Date	ESBWR Appli-cable?	Comments
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	3	Draft 04/1996	Yes	
	BTP SPLB-3-1	3	Draft 04/1996	Yes	
	Appendix A to SPLB 3-1	3	Draft 04/1996	Yes	
	Appendix B to SPLB 3-1	3	Draft 04/1996	Yes	
	Appendix C to SPLB 3-1	3	Draft 04/1996	Yes	
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	2	Draft 04/1996	Yes	
	BTP EMEB-3-1	2	Draft 04/1996	Yes	
3.6.3	Leak-Before-Break Evaluation Procedures	0	Draft 03/1987	—	Not credited. ~~Option available for possible future use during COL~~
3.7.1	Seismic Design Parameters	2	08/1989	Yes	
	Appendix A	0	08/1989	Yes	
3.7.2	Seismic System Analysis	2	08/1989	Yes	
	Appendix A	0	08/1989	Yes	
3.7.3	Seismic Subsystem Analysis	2	08/1989	Yes	
3.7.4	Seismic Instrumentation	1	07/1981	Yes	
3.8.1	Concrete Containment	1	07/1981	Yes	
	Appendix	0	07/1981	Yes	
3.8.2	Steel Containment	1	07/1981	Yes	applies only to Drywell Head

Table 1.9-20 [RAI 3.6-22, only section 3 shown in this table]

NRC Standard Review Plans and Branch Technical Positions Applicability to ESBWR

SRP No.	SRP Title or BTP	Appl. Rev.	Issued Date	ESBWR Appli-cable?	Comments
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1	07/1981	Yes	
3.8.4	Other Seismic Category I Structures	1	07/1981	Yes	
	Appendix A	0	07/1981	Yes	
	Appendix B	0	07/1981	Yes	
	Appendix C	0	07/1981	Yes	
	Appendix D	0	07/1981	Yes	
3.8.5	Foundations	1	07/1981	Yes	
3.9.1	Special Topics for Mechanical Components	3	Draft 04/1996	Yes	
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	3	Draft 04/1996	Yes	
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	2	Draft 04/1996	Yes	
	Appendix A	1	04/1984	Yes	
3.9.4	Control Rod Drive Systems	2	04/1984	Yes	
3.9.5	Reactor Pressure Vessel Internals	3	Draft 04/1996	Yes	
3.9.6	Inservice Testing of Pumps and Valves	3	Draft 04/1996	Yes	
3.9.7	Risk-Informed Inservice Testing	0	08/1998	—	COL
3.9.8	Review of Risk-Informed Inservice Inspection of Piping	0	09/2003	—	COL
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3	Draft 04/1996	Yes	
3.11	Environmental Qualification of Mechanical and Electrical Equipment	3	Draft 04/1996	Yes	

Table 1.10-1 [RAI 3.6-22, only section 3 shown in this table]

Summary of COL Items

Subject / Description of Item	Section
Provide a Description of the Switchyard	3.2.4 and Table 3.2-1
Site-Specific Design Basis Wind and Tornado	3.3.3.1
Ensure Remainder of Plant Structures, Systems and Components that are not Designed for Tornado Loads are Analyzed for Site-Specific Loadings	3.3.3.2
Evaluate Exposure to Water Spray	3.4.1.3
Detailed Flooding Evaluation	3.4.3
Submit Turbine System Maintenance Program including Probability Calculations of Turbine Missile Generation or Volumetrically Inspect All Low Pressure Turbine Rotors Every Other Refueling Outage	3.5.1.1.1.2
Evaluation of Nonsafety-related Structures, Systems, and Components (not housed in a tornado structure)	3.5.1.4
Confirm Low Probability of Site Proximity Missiles (Except Aircraft)	3.5.1.5
Missiles Generated by Natural Phenomena from Remainder of Plant Structures, Systems, and Components	3.5.4.1
Site Proximity Missiles and Aircraft Hazards	3.5.4.2
Impact of Failure of Nonsafety-Related Structures, Systems and Components	3.5.4.3
Turbine System Maintenance Program	3.5.4.4
Protection of Main Steamline Isolation Valves and Feedwater Isolation and Check Valves from Postulated Pipe Failures	3.6.1.3
~~Leak Before Break Evaluation Report~~ (deleted)	~~3.6.3~~
Details of Pipe Break Analysis Results and Protection Methods	3.6.5
Seismic Design Parameters	3.7.5.1
Seismic Analysis of EBAS Building	3.7.5.2
Structural Integrity Pressure Test of Containment Structure	3.8.1.7.1
Other Seismic Category I Structures	3.8.4
Foundation Waterproofing	3.8.6.1
Site-Specific Physical Properties and Foundation Settlement	3.8.6.2
Structural Integrity Pressure Result	3.8.6.3
Identification of Seismic Category I Structures	3.8.6.4

Table 1.10-1 [RAI 3.6-22, only section 3 shown in this table]

Summary of COL Items

Subject / Description of Item	Section
Risk-Informed In-Service Testing	3.9.7
Risk-Informed In-Service Inspection of Piping	3.9.8
Reactor Internals Vibration Analysis, Measurement and Inspection Program	3.9.9.1
ASME Class 2 or 3 or Quality Group D Components with 60-Year Design Life	3.9.9.2
Pump and Valve Inservice Testing Program	3.9.9.3
Audit of Design Specification and Design Reports	3.9.9.4
Valves for Process Radiation Monitoring System, Containment Monitoring System and HVAC Systems in Reactor, Control and Fuel Buildings	Table 3.9-8
Equipment Qualification Records	3.10.4
Dynamic Qualification Report	3.10.4
Verify Gamma and Beta Doses Assumed in Analysis are Bounding	3.11.4
Environmental Qualification Document (EQD)	3.11.5
Environmental Qualification Records	3.11.5
Drywell Pressurization: Vent Filter Design	3B.3.1.1
Gravity-Driven Cooling System Drain Pipe Design	3B.7.2
Lower Drywell Spillover Pipes	3B.7.3
~~Leak Before Break Evaluation Report~~ (deleted)	~~3.E.1.1~~
Radiation Environment Conditions Inside Containment Vessel for Accident Conditions	Table 3H-11
Radiation Environment Inside Reactor Building for Accident Conditions	Table 3H-12
Radiation Environment Conditions Inside Control Room Zone for Accident	Table 3H-13
Alternate Evaluation of Postulated Ruptures in High Energy Pipes	3J.1

[NOTES IN GREEN HIGHLIGHT ABOVE SUBSECTION HEADINGS THAT CONTAIN CHANGES]

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING [RAI 3.6-22]

This section deals with the structures, systems, components and equipment in the ESBWR Standard Plant.

Subsections 3.6.1 and 3.6.2 describe the design bases and protective measures which ensure that (1) the containment, (2) safety-related systems, components and equipment, and (3) other safety-related structures are adequately protected from the consequences associated with a postulated rupture of high-energy piping or crack of moderate-energy piping both inside and outside the containment.

Before delineating the criteria and assumptions used to evaluate the consequences of piping failures inside and outside of containment, it is necessary to define a pipe break event and a postulated piping failure:

- Pipe Break Event—Any single postulated piping failure occurring during normal plant operation and any subsequent piping failure and/or equipment failure that occurs as a direct consequence of the postulated piping failure.
- Postulated Piping Failure—Longitudinal or circumferential break or rupture postulated in high-energy fluid system piping or through-wall leakage crack postulated in moderate-energy fluid system piping. The terms used in this definition are explained in Subsection 3.6.2.

Structures, systems, components and equipment that are required to shut down the reactor and mitigate the consequences of a postulated piping failure, without off-site power, are defined as safety-related and are designed to Seismic Category I requirements.

The dynamic effects that may result from a postulated rupture of high-energy piping include (1) missile generation, (2) pipe whipping, (3) pipe break reaction forces, (4) jet impingement forces, (5) compartment, subcompartment, and cavity pressurizations, (6) decompression waves within the ruptured pipes, and (7) seven types of loads identified with a Loss-of-Coolant-Accident (LOCA).

~~Subsection 3.6.3 and Appendix 3E describe the implementation of the Leak Before Break (LBB) evaluation procedures as permitted by the broad scope amendment to General Design Criterion 4 (GDC 4) published in Reference 3.6-1. A LBB report shall be prepared with the stress report for the LBB-qualifiable piping in accordance with the guidelines presented in Appendix 3E. The LBB-qualified piping is excluded from pipe breaks, which are required to be postulated by Subsections 3.6.1 and 3.6.2, for design against their potential dynamic effects. However, such piping is included in postulation of pipe cracks for their effects as described in Subsections 3.6.1 and 3.6.2.~~

3.6.3 Leak-Before-Break Evaluation Procedures [RAI 3.6-22]

(deleted)

~~This subsection reviews the use of leak-before-break technology to exclude the dynamic effects of postulated pipe ruptures from the design basis of plant structures, systems and components (SSC) as discussed in SRP 3.6.3. General Design Criterion 4 (GDC4) of Appendix A to 10 CFR 50 allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of the pipe ruptures as discussed in SRP 3.6.2 Draft R2. The dynamic effects are defined in the introduction to DCD Section 3.6, above. Three forms of piping failure (full flow area circumferential and longitudinal breaks, and through-wall leakage crack) are postulated in accordance with Subsection 3.6.2 and Branch Technical Position EMEB 3-1 of NUREG-0800 (Standard Review Plan) for their dynamic and environmental effects.~~

~~In accordance with the modified GDC 4, (Reference 3.6-1), the mechanistic Leak-Before-Break (LBB) approach, justified by appropriate fracture mechanics techniques, is recognized as an acceptable procedure under certain conditions to exclude design against the dynamic effects from postulation of breaks in high energy piping. The LBB approach is not used to exclude postulation of cracks and associated effects as required in Subsection 3.6.2.2. As mentioned in Section 3.6, a report shall be prepared demonstrating LBB qualification of the piping. The report shall be prepared in accordance with the guidelines presented in Appendix 3E in conjunction with the stress report of the piping. The qualified piping, referred to in this DCD Tier 2 as the LBB-qualified piping, is excluded from pipe breaks, which are required to be postulated by Subsections 3.6.1 and 3.6.2, for design against their potential dynamic effects.~~

~~The following subsections describe (1) certain design bases where the LBB approach is not recognized by the NRC as applicable for exclusion of pipe breaks, and (2) certain conditions which limit the LBB applicability. Appendix 3E provides guidelines for LBB applications describing in detail the following necessary elements of an LBB report to be submitted by a COL applicant for NRC approval: (1) fracture mechanics methods, (2) leak rate prediction methods, (3) leak detection capabilities, and (4) typical special considerations for LBB applicability. The LBB application approach described in this subsection and Appendix 3E is consistent with that documented in Draft SRP 3.6.3 (Reference 3.6-2) and NUREG-1061 (Reference 3.6-3).~~

3.6.3.1 Scope of LBB Applicability [RAI 3.6-22]

(deleted)

~~The LBB approach is not used to replace existing regulations or criteria pertaining to the design bases of the Emergency Core Cooling System (Section 6.3), containment system (Section 6.2), or environmental qualification (Section 3.11). However, consistent with modified GDC 4, the design bases for dynamic qualification of mechanical and electrical equipment (Section 3.10) may exclude the dynamic load or vibration effects resulting from postulation of breaks in the LBB-qualified piping. This is also reflected in a note to Table 3.9-2 for ASME components. The LBB-qualified piping may not be excluded from design bases for environmental qualification unless the regulation permits it at the time of LBB qualification. For clarification, it is noted that the LBB approach is not used to relax the design requirements of the containment system that includes the Containment~~

~~Vessel (CV), vent system (vertical flow channels and horizontal vent discharges), drywell zones, suppression pool (wetwell), vacuum breakers, CV penetrations, and drywell head.~~

3.6.3.2 Conditions for LBB Applicability[RAI 3.6-22]

(deleted)

~~The LBB approach is not applicable to piping systems where operating experience has indicated particular susceptibility to failure from the effects of intergranular stress corrosion cracking (IGSCC), waterhammer, thermal fatigue, or erosion. Necessary preventive or mitigation measures are used and necessary analyses are performed, as discussed below, to avoid concerns for these effects. Other concerns, such as creep, brittle cleavage-type failure, potential indirect source of pipe failure, and deviation of as-built piping configuration, are also addressed.~~

- ~~Degradation by erosion, erosion/corrosion and erosion/cavitation caused by unfavorable flow conditions and water chemistry is examined. The evaluation is based on the industry experience and guidelines. Additionally, fabrication wall thinning of elbows and other fittings is considered in the purchase specification to assure that the code minimum wall requirements are met. These evaluations demonstrate that these mechanisms are not potential sources of pipe rupture.~~
- ~~The ESBWR plant design involves operation below 371°C (700°F) in ferritic steel piping and below 427°C (800°F) in austenitic steel piping. This assures that creep and creep-fatigue are not potential sources of pipe rupture.~~
- ~~The design also assures that the piping material is not susceptible to brittle cleavage-type failure over the full range of system operating temperatures (i.e., the material is on the upper shelf).~~
- ~~The ESBWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to IGSCC. The material of major high-energy piping in the primary containment is carbon steel or ferritic steel, except for the austenitic stainless reactor water cleanup piping in the containment.~~
- ~~A systems evaluation of potential waterhammer is made to assure that pipe rupture caused by to this mechanism is unlikely. Waterhammer is a generic term including various unanticipated high frequency hydrodynamic events such as steam hammer and water slugging. To demonstrate that waterhammer is not a significant contributor to pipe rupture, reliance on historical frequency of waterhammer events in specific piping systems coupled with a review of operating procedures and conditions is used for this evaluation. The ESBWR design includes features such as vacuum breakers and jockey pumps, coupled with improved operational procedures to reduce or eliminate the potential for waterhammer identified by past experience. Certain anticipated waterhammer events, such as a closure of a valve, are accounted for in the Code design and analysis of the piping.~~

- ~~The systems evaluation also addresses a potential for fatigue cracking or failure from thermal and mechanical induced fatigue. Based on past experience, the piping design avoids potential for significant mixing of high and low temperature fluids or mechanical vibration. The startup and preoperational monitoring assures avoidance of detrimental mechanical vibration.~~
- ~~Based on experience and studies by Lawrence Livermore Laboratory, potential sources of indirect pipe rupture are extremely unlikely. Inservice inspection and testing of snubbers (if used) are performed to provide for a low snubber failure rate.~~
- ~~Initial LBB evaluation is based on the design configuration and stress levels that are acceptably higher than those identified by the initial analysis. This evaluation is reconciled when the as-built configuration is documented and the Code stress evaluation is reconciled. It is assured that the as-built configuration does not deviate significantly from the design configuration to invalidate the initial LBB evaluation, or a new evaluation coupled with necessary configuration modifications is made to assure applicability of the LBB procedure.~~

3.6.6 References [RAI 3.6-22]

3.6-1 USNRC, "Modification of General Design Criterion 4, Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture," Federal Register, Volume 52, No. 207, Rules and Regulations, Pages 41288 through 41295, October 27, 1987.

3.6-2 USNRC, "Standard Review Plan; Public Comments Solicited," Federal Register, Volume 52, No. 167, Notices, Pages 32626 to 32633, August 28, 1987.

3.6-3 USNRC, "Evaluation of Potential for Pipe Breaks, Report of the US NRC Piping Review Committee," NUREG-1061, Volume 3, November 1984.

3.6-4 ANSI/ANS-58.2-1988 "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture."

3.6-5 USNRC, "Standard Review Plan for the Review of Safety Analysis reports for Nuclear Power Plants," NUREG-0800, Section 3.6.1. "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", Draft Revision 3, April 1996.

3.6-6 USNRC, "Standard Review Plan for the Review of Safety Analysis reports for Nuclear Power Plants", NUREG-0800, Section 3.6.2 "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Draft Revision 2, April 1996.

3.6-7 ~~USNRC, "Standard Review Plan for the Review of Safety Analysis reports for Nuclear Power Plants," NUREG-0800, Section 3.6.3 "Leak-Before-Break Evaluation Procedures," Draft, March 1987.~~deleted.

3.6-8 10 CFR 50 "Domestic licensing of production and utilization facilities."

3.6-9 10 CFR 100 "Reactor site criteria."

Table 3.2-1 [RAI 3.6-22, only the affected part of table shown below]

Classification Summary

Principal Components[1]	Safety Class.[2]	Location[3]	Quality Group[4]	QA Req.[5]	Seismic Category[6]	Notes
4. Nitrogen accumulators (for ADS and manual actuation of SRVs)	3	CV	C	B	I	
5. Piping and valves (including supports) for main steamlines (MSL) and feedwater (FW) lines up to and including the outermost containment isolation valves	1	CV, RB	A	B	I	
6. Piping (including supports) for MSL from outermost isolation valve to and including seismic interface restraint and FW from outermost isolation valve to and including the shutoff valve	2	RB	B	B	I	Seismic interface restraints are located inside the seismic category I building.
7. Deleted.						
8. Piping and valves (including supports) from FW shutoff valve to the seismic interface restraint	2	RB	B	B	I	
9. Pipe whip restraints ~~—MSL/FW if needed~~	3	CV, RB	—	B	I or II	**Pipe Whip Restraints** —Pipe Whip Restraints are required on the Main Steam Line (MSL) and Feedwater (FW) piping ~~except where a "Leak-Before-Break" evaluation has been approved by the NRC~~.
10. Main steam drain piping and valves (including supports) within outermost containment isolation valves	1	CV, RB	A	B	I	(7)

Table 3.9-2 [3.6-22]

Load Combinations and Acceptance Criteria for Safety-Related, ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures

Plant Event	Service Loading Combination [1], [2], [3]	ASME Service Level [4]
1. Normal Operation (NO)	N	A
2. Plant/System Operating Transients (SOT)	(a) N + TSV (b) N + SRV[5]	B B
3. NO + SSE	N + SSE	B[11], [12]
4. Infrequent Operating Transient (IOT), ATWS, DPV	(a) N[6] + SRV[5] (b) N + DPV[7]	C[13] C[13]
5. SBL	N + SRV[8] + SBL~~[9]~~	C[13]
6. SBL or IBL + SSE	N + SBL (or IBL)~~[9]~~ + SSE + SRV[8]	D[13]
7. LBL + SSE	N + LBL~~[9]~~ + SSE	D[13]
8. NLF	N + SRV[5] + TSV[10]	D[13]

Notes:

(1) See Legend on the following pages for definition of terms. Refer to Table 3.9-1 for plant events and cycles information.

The service loading combination also applies to Seismic Category I Instrumentation and electrical equipment (refer to Section 3.10).

(2) For vessels, loads induced by the attached piping are included as identified in their design specification.

For piping systems, water (steam) hammer loads are included as identified in their design specification.

(3) The method of combination of the loads is in accordance with NUREG-0484, Revision 1.

(4) The service levels are as defined in appropriate subsection of ASME Section III, Division 1.

(5) The most limiting load combination case among SRV(1), SRV(2) and SRV (ALL). For main steam and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.

(6) The reactor coolant pressure boundary is evaluated using in the load combination the maximum pressure expected to occur during ATWS.

(7) This applies only to the Main Steam and Isolation Condenser systems. The loads from this event are combined with loads associated with the pressure and temperature concurrent with the event.

(8) The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (5) for main steam and branch piping.

(9) ~~The piping systems that are qualified to the leak-before-break criteria of Subsection 3.6.3 are excluded from the pipe break events to be postulated for design against LOCA dynamic effects, viz., SBL, IBL and LBL.~~(deleted)

(10) This applies only to the main steamlines and components mounted on it. The low probability that the TSV closure and SRV loads can exist at the same time results in this combination being considered under service level D.

(11) Applies only to fatigue evaluation of ASME Code Class 1 components and core support structures. See Dynamic Loading Event No. 13, Table 3.9-1, and Note 5 of Table 3.9-1 for number of cycles.

(12) For ASME Code Class 2 and 3 piping the following changes and additions to ASME Code Section III Subsection NC-3600 and ND-3600 are necessary and shall be evaluated to meet the following stress limits:

$$S_{SAM} = i\,\frac{M_c}{Z} \leq 3.0\,S_h \quad (\leq 2.0S_y) \qquad \text{Eq. (12a)}$$

Where: S_{SAM} is the nominal value of seismic anchor motion stress

M_c is the combined moment range equal to the greater of (1) the resultant range of thermal and thermal anchor movements plus one-half the range of the SSE anchor motion, or (2) the resultant range of moment due to the full range of the SSE anchor motions alone.

i and Z are defined in ASME Code Subsections NC/ND-3600

SSE inertia and seismic anchor motion loads shall not be included in the calculation of ASME Code Subsections NC/ND-3600 Equation (9), Service Levels A and B and Equations (10) and (11).

(13) ASME Code Class 1, 2 and 3 Piping systems, which are essential for safe shutdown under the postulated events are designed to meet the requirements of NUREG-1367. Piping system dynamic moments can be calculated using an elastic response spectrum or time history analysis.

Load Definition Legend for Table 3.9-2	
Normal (N)	Normal and/or abnormal loads associated with the system operating conditions, including thermal loads, depending on acceptance criteria.
SOT	System Operational Transient (Subsection 3.9.3.1).
IOT	Infrequent Operational Transient (Subsection 3.9.3.1).
ATWS	Anticipated Transient Without Scram.
TSV	Turbine stop valve closure induced loads in the main steam piping and components integral to or mounted thereon.
RBV Loads	Dynamic loads in structures, systems and components because of reactor building vibration (RBV) induced by a dynamic event.
NLF	Non-LOCA Fault.
SSE	RBV loads induced by safe shutdown earthquake.

Load Definition Legend for Table 3.9-2	
SRV(1), SRV(2)	RBV loads induced by safety/relief valve (SRV) discharge of one or two adjacent valves, respectively.
SRV (ALL)	RBV loads induced by actuation of all safety/relief valves, which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV (ADS)	RBV loads induced by the actuation of safety/relief valves in Automatic Depressurization Subsystem operation, which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.
DPV	Depressurization Valve opening induced loads in the stub tubes and Main Steam system piping and pipe-mounted equipment.
LOCA	The loss-of-coolant accident associated with the postulated pipe failure of a high-energy reactor coolant line. The load effects are defined by LOCA1 through LOCA7. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.
LOCA1	Pool swell (PS) drag/fallback loads on essential piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
LOCA2	Pool swell (PS) impact loads acting on essential piping and components located above the suppression pool water upper surface.
LOCA3	(a) Oscillating pressure induced loads on submerged essential piping and components during main vent clearing (VLC), condensation oscillations (COND), or chugging (CHUG), or (b) Jet impingement (JI) load on essential piping and components as a result of a postulated IBL or LBL event. Piping and components are defined essential, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without off-site power (refer to introduction to Subsection 3.6).
LOCA4	RBV load from main vent clearing (VLC).
LOCA5	RBV loads from condensation oscillations (COND).
LOCA6	RBV loads from chugging (CHUG).
LOCA7	Annulus pressurization (AP) loads due to a postulated line break in the annulus region between the RPV and shieldwall. Vessel depressurization loads on reactor internals (Subsection 3.9.2.4) and other loads due to reactor blowdown reaction and jet impingement and pipe whip restraint reaction from the broken pipe are included with the AP loads.
SBL	Loads induced by small break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a), LOCA4 and LOCA6. ~~See Note (9).~~
IBL	Loads induced by intermediate break LOCA (Subsection 3.9.3.1); the loads are: LOCA3(a) or LOCA3(b), LOCA4, LOCA5 and LOCA6. ~~See Note 9 of Table 3.9-2.~~
LBL	Loads induced by large break LOCA (Subsection 3.9.3.1); the loads are: LOCA1 through LOCA7. ~~See Note (9).~~

[ALL CHANGES TO APPENDIX 3E DUE TO RAI 3.6-22 RESPONSE]

3E. GUIDELINES FOR LEAK BEFORE BREAK APPLICATION

3E.1 INTRODUCTION

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~~As discussed in Subsection 3.6.3, this appendix provides detailed guidelines for addressing Leak Before Break (LBB) for specific piping systems. Also included in this appendix are the fracture mechanics properties of ESBWR piping materials and analysis methods, including the leak rate calculation methods.~~

~~Piping qualified by LBB is excluded from the non-mechanistic postulation requirements of a Double Ended Guillotine Break (DEGB) specified in Subsection 3.6.3. The LBB qualification means that the throughwall flaw lengths that are detectable by leakage monitoring systems (Subsection 5.2.5) are significantly smaller than the flaw lengths that could lead to pipe rupture or instability.~~

~~The fracture mechanics properties aspects required for evaluation in accordance with Subsection 3.6.3 are addressed in Section 3E.2. The fracture mechanics techniques and methods for the determination of critical flaw lengths and evaluation of flaw stability are described in Section 3E.3. The determination of flaw lengths for detectable leakages with margin is explained in Section 3E.4. A brief discussion on the leak detection capabilities is presented in Section 3E.5.~~

~~Material selection and the deterministic LBB evaluation procedure are discussed in this section.~~

3E.1.1 Material Selection Guidelines

(deleted)

~~The LBB approach is applicable to piping systems for which the materials meet the following criteria:~~

- ~~low probability of failure from the effects of corrosion (e.g., intergranular stress corrosion cracking); and~~
- ~~adequate margin before susceptibility to cleavage type fracture over the full range of systems operating temperatures where pipe rupture could have significant consequences.~~

~~The ESBWR plant design specifies use of austenitic stainless steel piping made of material (e.g., nuclear grade or low carbon type) that is recognized as resistant to Inter-Granular Stress Corrosion Cracking (IGSCC). The carbon steel or ferritic steels specified for the reactor pressure boundary are described in Subsection 3E.2.2. These steels are assured to have adequate toughness to preclude a fracture at operating temperatures. A COL applicant is expected to supply a detailed justification in the LBB evaluation report considering system temperature, fluid velocity and environmental conditions.~~

3E.1.2 Deterministic Evaluation Procedure

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~~The following deterministic analysis and evaluation is performed as an NRC-approved method to justify applicability of the LBB concept.~~

- ~~Use the fracture mechanics and the leak rate computational methods that are accepted by the NRC staff, or are demonstrated accurate with respect to other acceptable computational procedures or with experimental data.~~
- ~~Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including toughness and tensile data, long-term effects such as thermal aging, and other limitations.~~
- ~~Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combination. For each pipe size in the functional system, identify the location(s), which have the least favorable combination of stress and material properties for base metal, weldments and safe ends.~~
- ~~Postulate a throughwall flaw at the location(s) specified above. The size of the flaw should be large enough so that the leakage is assured detection with sufficient margin using the installed leak detection capability when pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described. For the estimation of leakage, the normal operating loads (i.e., deadweight, thermal expansion, and pressure) are to be combined based on the algebraic sum of individual values.~~

  ~~Using fracture mechanics stability analysis or limit load analysis described below, and normal plus Safe Shutdown Earthquake (SSE) loads, determine the critical crack size for the postulated throughwall crack. Determine crack size margin by comparing the selected leakage detection size crack to the critical crack size. Demonstrate that there is a margin of 2 between the leakage detection and critical crack sizes. The same load combination method selected below is used to determine the critical crack size.~~
- ~~Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage detection size crack does not experience unstable crack growth if 1.4 times the normal plus SSE loads are applied. Demonstrate that crack growth is stable and the final crack is limited such that a double-ended pipe break should not occur. The deadweight, thermal expansion, pressure, SSE (inertial), and Seismic Anchor Motion (SAM) loads are combined based on the same method used for the primary stress evaluation by the ASME Code. The SSE (inertial) and SAM loads are combined by Square Root of the Sum of the Squares (SRSS) method.~~
- ~~The piping material toughness (J-Resistance curves) and tensile (stress-strain curves) properties are determined at temperatures near the upper range of normal plant operation.~~

- ~~The specimen used to generate J-Resistance (J-R) curves is assured large enough to provide crack extensions up to an amount consistent with J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques are used as described in NUREG-1061, Volume 3, or in NUREG/CR-4575. Other techniques can be used if adequately justified.~~

- ~~The stress-strain curves are obtained over the range from the preoperational limit to maximum load.~~

- ~~Preferably, the materials tests should be conducted using archival materials for the pipe being evaluated. If archival material is not available, plant specific or industry wide generic material databases are assembled and used to define the required material tensile and toughness properties. Test material includes base and weld metals.~~

- ~~To provide an acceptable level of reliability, generic databases are reasonable lower bounds for compatible sets of material tensile and toughness properties associated with materials at the plant. To assure that the plant specific generic database is adequate, a determination is made to demonstrate that the generic database represents the range of plant materials to be evaluated. This determination is based on a comparison of the plant material properties identified above with those of the materials used to develop the generic database. The number of material heats and weld procedures tested are adequate to cover the strength and toughness range of the actual plant materials. Reasonable lower bound tensile and toughness properties from the plant specific generic database are to be used for the stability analysis of individual materials, unless otherwise justified.~~

  ~~Industry generic data bases are reviewed to provide a reasonable lower bound for the population of material tensile and toughness properties associated with any individual specification (e.g., A106, Grade B), material type (e.g., austenitic steel) or welding procedures.~~

  ~~The number of material heats and weld procedures tested should be adequate to cover the range of the strength and tensile properties expected for specific material specifications or types. Reasonable lower bound tensile and toughness properties from the industry generic database are used for the stability analysis of individual materials.~~

  ~~If the data are being developed from an archival heat of material, three stress-strain curves and three J-Resistance curves from the one heat of material is sufficient. The tests should be conducted at temperatures near that upper range of normal plant operation. Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest. The lower toughness should be used in the fracture mechanics evaluation. One J-R curve and one stress-strain curve for one~~

~~base metal and weld metal are considered adequate to determine temperature dependence.~~

- ~~There are certain limitations that currently preclude generic use of limit load analyses to evaluate leak-before-break conditions deterministically. However, a modified limit-load analysis can be used for austenitic stainless steel piping to demonstrate acceptable margins as described in Subsection 3E.3.3.~~

3E.2 Material Fracture Toughness Characterization

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~~This Subsection describes the fracture toughness properties and flow stress evaluation for the ferritic and austenitic stainless steel materials used in ESBWR plant piping, as required for evaluation according to Subsection 3E.1.2.~~

3E.2.1 Fracture Toughness Characterization

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~~When the Elastic-Plastic Fracture Mechanics (EPFM) methodology or the J-T methodology is used to evaluate the leak-before-break conditions with postulated throughwall flaws, the material toughness property is characterized in the form of J-integral Resistance curve (or J-R curve) (References 3E-1, 3E-2 and 3E-3). The J-R curve, schematically shown in Figure 3E-1, represents the material's resistance to crack extension. The onset of crack extension is assumed to occur at a critical value of J. Where the plane strain conditions are satisfied, initiation J is denoted by~~ $\cancel{J_{IC}}$~~. Plane strain crack conditions, achieved in test specimen by side grooving, generally provide a lower bound behavior for material resistance to stable crack growth.~~

~~Once the crack begins to extend, the increase of J with crack growth is measured in terms of slope or the nondimensional tearing modulus, T, expressed as:~~

$$\cancel{T = \frac{E}{(\sigma_f)^2} \cdot \frac{dJ}{da}} \qquad \cancel{(3E\text{-}1)}$$

~~The flow stress,~~ $\cancel{\sigma_f}$~~, is a function of the yield and ultimate strength, and E is the elastic modulus. Generally,~~ $\cancel{\sigma_f}$ ~~is assumed as the average of the yield and ultimate strength. The slope of the material J-R curve is a function of crack extension Δa. Generally, the slope decreases with crack extension thereby giving a convex upward appearance to the material J-R curve in Figure 3E-1.~~

~~To evaluate the stability of crack growth, it is convenient to represent the material J-R curve in the J-T space as shown in Figure 3E-1. The resulting curve is labeled as J-T material. Crack instability is predicted at the intersection point of the J/T material and J/T applied curves.~~

~~The crack growth variably involves some elastic unloading and distinctly nonproportional plastic deformation near the crack tip. J-integral is based on the deformation theory of plasticity (References 3E-4 and 3E-5), which inadequately models both of these aspects of plastic behavior. In order to use J-integral to characterize crack growth (i.e., to assure J-controlled crack growth), the following sufficiency condition, in terms of a nondimensional parameter proposed by Hutchinson and Paris (Reference 3E-6), is used:~~

$$\cancel{\omega = \frac{b}{j} \cdot \frac{dJ}{da} >> 1} \qquad \cancel{(3E\text{-}2)}$$

~~where b is the remaining ligament. Reference 3E-7 suggests that ω > 10 would satisfy the J-controlled growth requirements. However, if the requirements of this criteria are strictly followed, the amount of crack growth allowed would be very small in most test specimen geometries. Use of such a material J-R curve in J/T evaluation would result in grossly underpredicting the instability loads for large diameter pipes where considerable stable crack growth is expected to occur before reaching the instability point. To overcome this difficulty, Ernst (Reference 3E-8) proposed a modified J-integral~~ ~~J_{mod},~~ ~~which was shown to be effective even when limits on ω were grossly violated. The Ernst correction essentially factors in the effect of crack extension in the calculated value of J. This correction can be determined experimentally by measuring the usual parameters: load, displacement, and crack length.~~

~~The definition of~~ ~~J_{mod}~~ ~~is~~

$$J_{mod} = J - \int_{a_0}^{a} \left| \frac{\partial}{\partial a}\left(J - G_e\right) \right|_{\delta_{pl}} da \qquad \text{(3E-3)}$$

~~where:~~

~~J = is based on deformation theory of plasticity;~~

~~G_e = is the linear elastic Griffith energy release rate of elastic J,~~ ~~J_{el};~~

~~δ_{pl} = is the nonlinear part of the load-point displacement (or simply the total minus the elastic displacement); and~~

~~a_0, a = are the initial and current crack length, respectively.~~

~~For the particular case of the compact tension specimen geometry, the preceding equation and the corresponding rate take the form:~~

$$J_{mod} = J + \int_{a_0}^{a} \gamma \cdot \frac{J_{pl}}{b} \cdot da \qquad \text{(3E-4)}$$

~~where~~ ~~J_{pl}~~ ~~is the nonlinear part of the deformation theory J, b is the remaining ligament and γ is~~

$$\gamma = \left(1 + 0.76\,\frac{b}{W}\right) \qquad \text{(3E-5)}$$

~~Consequently, the modified material tearing modulus~~ ~~T_{mod}~~ ~~can be defined as:~~

$$T_{mod} = T_{mat} + \frac{E}{(\sigma_f)^2}\left(\frac{\gamma}{b} \cdot J_{pl}\right) \qquad \text{(3E-6)}$$

~~Because in most of the test J-R curves the ω > 10 limit was violated, all of the material J-T data were recalculated in the~~ ~~J_{mod}, T_{mod}~~ ~~format. The~~ ~~J_{mod}, T_{mod}~~ ~~calculations were performed up to crack extension of a = 10% of the original ligament in the test specimen. The J-T curves were then extrapolated to larger J values using the method recommended in NUREG-1061, Vol. 3 (Reference 3E-9). The~~ ~~J_{mod} – T_{mod}~~ ~~approach is used in this~~

~~appendix for illustrative purposes. It should be adopted if justified based on its acceptability by the technical literature. A J_D approach is another more justifiable approach.~~

~~For terminology see References 3E-1 through 3E-3 and 3E-9.~~

3E.2.2 Carbon Steels and Associated Welds

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~~The carbon steels used in the ESBWR reactor coolant pressure boundary piping are SA 106 Gr. B or SA 333 Gr. 6. The first specification covers seamless pipe and the second one pertains to both seamless and seam-welded pipe, although only seamless pipe will be used for ESBWR reactor coolant pressure boundary piping. The corresponding material specifications used for carbon steel flanges, fittings and forgings are equivalent to the piping specifications.~~

~~While the chemical composition requirements for a pipe per SA 106 Gr. B and SA 333 Gr. 6 are identical, the latter is subjected to two additional requirements: (1) a normalizing heat treatment which refines the grain structure and (2) a Charpy test at -45.6°C (-50°F) with a specified minimum absorbed energy of 85.5 Nm (13 ft-lb). The electrodes and filler metal requirements for welding carbon steel to carbon or low alloy steel are as specified in Table 3E-1.~~

~~A comprehensive test program was undertaken at GE to characterize the carbon steel base and weld material toughness properties. The next section describes the scope and the results of this program.~~

3E.2.2.1 Fracture Toughness Test Program

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~~The test program consisted of generating true stress-true strain curves, J-Resistance curves and the Charpy V-notch tests. Two materials were selected: (1) SA333 Gr. 6, 16-in. diameter Schedule 80 pipe and (2) SA516, Gr. 70, 1-1/4 in. thickness plate. Table 3E-2 shows the chemical composition and mechanical property test information provided by the material supplier. The materials were purchased to the same specifications as those to be used in the ESBWR applications.~~

~~To produce a circumferential butt weld, the pipe was cut in two pieces along a circumferential plane and welded back using the shielded metal arc process. The weld prep was a single V design with a backing ring. The preheat temperature was 93.3°C (200°F).~~

~~The plate material was cut along the longitudinal axis and welded back using the submerged arc weld (SAW) process. The weld prep was of a single V type with one side as vertical and the other side at 45 degrees. A backing plate was used during the welding with a clearance of 0.64 cm (1/4 inch) at the bottom of the V. The interpass temperature was maintained at less than 260°C (500°F).~~

~~Both the plate and the pipe welds were x-rayed according to Code (Reference 3E-10) requirements and were found to be satisfactory.~~

~~It is well-known that carbon steel base materials show considerable anisotropy in fracture toughness properties. The toughness depends on the orientation and direction of propagation of the crack in relation to the principal direction of mechanical working or gain flow. Thus, the selection of proper orientation of Charpy and J-R curve test specimen is important. Figure 3E-2 shows the orientation code for rolled plate and pipe specimen as given in ASTM Standard E399 (Reference 3E-11). Because a throughwall circumferential crack configuration is of most interest from the Double Ended Guillotine Break (DEGB) point of view, the L-T specimen in a plate and the L-C specimen in a pipe provide the appropriate toughness properties for that case. On the other hand, T-L and C-L specimens are appropriate for the axial flaw case.~~

~~Charpy test data are reviewed first because they provide a qualitative measure of the fracture toughness.~~

~~Charpy Tests~~

~~The absorbed energy or its complement, the lateral expansion measured during a Charpy V-notch test provides a qualitative measure of the material toughness. For example, in the case of austenitic stainless steel flux weldments, the observed lower Charpy energy relative to the base metal was consistent with the similar trend observed in the J-Resistance curves. The Charpy tests in this program were used as preliminary indicators of relative toughness of welds, heat-affected zones (HAZs) and the base metal.~~

~~The carbon steel base materials exhibit considerable anisotropy in the Charpy energy as illustrated by Figure 3E-3 from Reference 3E-12. This anisotropy is associated with development of grain flow due to mechanical working. The Charpy orientation C in Figure 3E-3 (orientations LC and LT in Figure 3E-2) is the appropriate one for evaluating the fracture resistance to the extension of a throughwall circumferential flaw. The upper shelf Charpy energy associated with axial flaw extension (orientation A in Figure 3E-3) is considerably lower than that for the circumferential crack extension.~~

~~A similar trend in the base metal Charpy energies was also noted in this test program. Figure 3E-4 and Figure 3E-5 show the pipe and plate material Charpy energies for the two orientations as a function of temperature. The tests were conducted at six temperatures ranging from room temperature to 288°C (550°F). From the trend of the Charpy energies as a function of temperature in Figure 3E-4 and Figure 3E-5 it is clear that even at room temperature the upper shelf conditions have been reached for both the materials.~~

~~No such anisotropy is expected in the weld metal because it does not undergo any mechanical working after its deposition. This conclusion is also supported by the available data in the technical literature. The weld metal Charpy specimens in this test program were oriented the same way as the LC or LT orientations in Figure 3E-2. The Heat Affected Zone (HAZ) Charpy specimens were also oriented similarly.~~

~~Figure 3E-6 shows a comparison of the Charpy energies from the SA333 Gr. 6 base metal, the weld metal and the HAZ. In most cases two specimens were used. Considerable scatter in the weld and HAZ Charpy energy values is seen. Nevertheless, the average energies for the weld metal and the HAZ seem to fall at or above the average base metal values. This indicates that, unlike the stainless steel flux weldments, the fracture toughness of carbon steel weld and HAZ, as measured by the Charpy tests, is at least equal to the carbon steel base metal.~~

~~The preceding results and the results of the stress-strain tests discussed in the next section or other similar data are used as a basis to choose between the base and the weld metal properties for use in the J-T methodology evaluation.~~

~~Stress-Strain Tests~~

~~The stress-strain tests were performed at three temperatures: room temperature, 177°C (350°F), and 288°C (550°F). Base and weld metal from both the pipe and the plate were tested. The weld specimens were in the as-welded condition. The standard test data obtained from these tests are summarized in Table 3E-3.~~

~~An examination of Table 3E-3 shows that the measured yield strength of the weld metal, as expected, is considerably higher than that of the base metal. For example, the 288°C (550°F) yield strength of the weld metal in Table 3E-3 ranges from 358.6 MPa (52 ksi) to 406.8 MPa (59 ksi), whereas the base metal yield strength is only 234.5 MPa (34 ksi). The impact of this observation in the selection of appropriate material (J/T) curve is discussed in later sections.~~

~~Figure 3E-7 through Figure 3E-10 show the plots of the 288°C (550°F) and 177°C (350°F) stress-strain curves for both the pipe and the plate used in the test. As expected, the weld metal stress-strain curve in every case is higher than the corresponding base metal curve. The Ramberg-Osgood format characterization of these stress-strain curves is given in Subsection 3E.3.2 where appropriate values of α and n are also provided.~~

~~J-R Curve Tests~~

~~The test temperatures selected for the J-R curve tests were: room temperature, 177°C (350°F), and 288°C (550°F). Both the weld and the base metal were included. Due to the curvature, only the 1T plan compact tension (CT) specimens were obtained from the 0.41 m (16-in). diameter test pipe. Both 1T and 2T plan test specimens were prepared from the test plate. All of the CT specimens were side-grooved to produce plane strain conditions.~~

~~Table 3E-4 shows some details of the J-R curve tests performed in this test program. The J-R curve in the LC orientation of the pipe base metal and in the LT orientation of the plate base metal represent the material's resistance to crack extension in the circumferential direction. Thus, the test results of these orientations were used in the LBB evaluations. The orientation effects are not present in the weld metal. As an example of the J-R curve obtained in the test program, Figure 3E-11 shows the plot of J-R curve obtained from specimen OWLC-A.~~

3E.2.2.2 Material (J/T) Curve Selection

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~~The normal operating temperatures for most of the carbon steel piping in the reactor coolant pressure boundary in the ESBWR generally fall into two categories: 274°C (528°F) to 288°C (550°F) and 216°C (420°F). The latter temperature corresponds to the operating temperature of the feedwater piping system. The selections of the appropriate material (J/T) curves for these two categories are discussed next.~~

~~**Material J/T Curve for 288°C (550°F)**~~

~~A review of the test matrix in Table 3E-4 shows that five tests were conducted at 288°C (550°F). Two tests were on the weld metal, two were on the base metal, and one was on the heat-affected zone. Figure 3E-12 shows the plot of material~~ $\cancel{J_{mod}, T_{mod}}$ ~~values calculated from the J-Δa values obtained from the 288°C (550°F) tests. The value of flow stress,~~ $\cancel{\sigma_f}$, ~~used in the tearing modulus calculation (Equation 3E-1) was 358.5 MPa (52.0 ksi) based on data shown in Table 3E-3. To convert the deformation J and dJ/da values obtained from the J-R curve into~~ $\cancel{J_{mod}, T_{mod}}$. ~~Equations 3E-4 and 3E-6 were used. Only the data from the pipe weld (Specimen ID OWLC-A) and the plate base metal (Specimen ID BMLI-12) are shown in Figure 3E-12. A few unreliable data points were obtained in the pipe base metal (Specimen ID OBLC-3) J-R curve test because of a malfunction in the instrumentation. Therefore, the data from this test were not included in the evaluation. The J-R curves from the other two 288°C (550°F) tests were evaluated as described in the next paragraph. For comparison purposes, Figure 3E-12 also shows the SA106 carbon steel J-T data obtained from the J-R curve reported by Gudas (Reference 3E-13). The curve also includes extrapolation to higher J values based on the method recommended in NUREG 1061, Vol. 3 (Reference 3E-9).~~

~~The~~ $\cancel{J_{mod} - T_{mod}}$ ~~data for the plate weld metal and the plate HAZ were evaluated. A comparison shows that these data fall slightly below those for the plate base metal shown in Figure 3E-12. On the other hand, as noted in Subsection 3E.2.2.1, the yield strength of the weld metal and the HAZ is considerably higher than that of the base metal. The material stress-strain and J-T curves are the two key inputs in determining the instability load and flaw values by the (J/T) methodology. Calculations performed for representative throughwall flaw sizes showed that the higher yield strength of the weld metal more than compensates for the slightly lower J-R curve and, consequently, the instability load and flaw predictions based on base metal properties are smaller (i.e., conservative). Accordingly, it was concluded that the material (J-T) curve shown in Figure 3E-12 is the appropriate one to use in the LBB evaluations for carbon steel piping at 288°C (550°F).~~

~~**Material J/T Curve for 216°C (420°F)**~~

~~Because the test temperature of 177°C (350°F) can be considered reasonably close to the 216°C (420°F), the test J-R curves for 177°C (350°F) were used in this case. A review of the test matrix in Table 3E-4 shows that three tests were conducted at 177°C (350°F). The~~ $\cancel{J_{mod}, T_{mod}}$ ~~data for all three tests were reviewed. The flow stress value used in the~~

~~tearing modulus calculation was 372.4 MPa (54 ksi) based on Table 3E-3. Also reviewed were the data on SA106 carbon steel at 300°F reported by Gudas (Reference 3E-13).~~

~~Consistent with the trend of the 288°C (550°F) data, the 177°C (350°F) weld metal (J-T) data fell below the plate and pipe base metal data. This probably reflects the slightly lower toughness of the SAW weld in the plate. The (J/T) data for the pipe base metal fell between the plate base metal and the plate weld metal. Based on the considerations similar to those presented in the previous section, the pipe base metal J-T data, although they may lie above the weld J-T data, were used for selecting the appropriate (J-T) curve. Accordingly, the curve shown in Figure 3E-13 was developed for using the (J-T) methodology in evaluations at 216°C (420°F).~~

3E.2.3 Stainless Steels and Associated Welds

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~~The stainless steels used in the ESBWR reactor coolant pressure boundary piping are either nuclear grade or low carbon Type 304 or 316. These materials and the associated welds are highly ductile and, therefore, undergo considerable plastic deformation before failure can occur. Toughness properties of Type 304 and 316 stainless steels have been extensively reported in the open technical literature and are, thus, not discussed in detail in this section. Due to high ductility and toughness, modified limit load methods can be used to determine critical crack lengths and instability loads (Subsection 3E.3.3).~~

3E.3 FRACTURE MECHANICS METHODS

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~~This Subsection deals with the fracture mechanics techniques and methods for the determination of critical flaw lengths and instability loads for materials used in ESBWR. These techniques and methods comply with criteria described in Subsection 3E.1.2.~~

3.E.3.1 Elastic-Plastic Fracture Mechanics or (J/T) Methodology

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~~Failure in ductile materials such as highly tough ferritic materials is characterized by considerable plastic deformation and significant amount of stable crack growth. The EPFM approach outlined in this Subsection considers these aspects. Two key concepts in this approach are (1) J-integral (References 3E-14 and 3E-15) which characterizes the intensity of the plastic stress-strain field surrounding the crack tip and (2) the tearing instability theory (References 3E-16 and 3E-17) which examines the stability of ductile crack growth. A key advantage of this approach is that the material fracture toughness characteristic is explicitly factored into the evaluation.~~

3E.3.1.1 Basic (J/T) Methodology

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~~Figure 3E-14 schematically illustrates the J/T methodology for stability evaluation. The material (J/T) curve in Figure 3E-14 represents the material's resistance to ductile crack extension. Any value of J falling on the material R-curve is denoted as J_{mat} and is a function solely of the increase in crack length Δa. Also defined in Figure 3E-14 is the "applied" J, which for given stress-strain properties and overall component geometry, is a function of the applied load P and the current crack length, a. Hutchinson and Paris (Reference 3E-17) also define the following two nondimensional parameters:~~

$$\cancel{T_{applied} = \frac{E}{(\sigma_f)^2} \cdot \frac{\partial J_{applied}}{\partial a}}$$

$$\cancel{T_{mat} = \frac{E}{(\sigma_f)^2} \cdot \frac{dJ_{mat}}{da}} \qquad \text{~~(3E-7)~~}$$

~~where E is Young's modulus and σ_f is an appropriate flow stress.~~

~~Intersection point of the material and applied (J/T) curves denotes the instability point. This is mathematically stated as:~~

$$\cancel{J_{applied}(a, P) = J_{mat}(a))} \qquad \text{~~(3E-8)~~}$$

$$\cancel{T_{applied} < T_{mat}(\text{stable})}$$

$$\cancel{T_{applied} > T_{mat}(\text{unstable})} \qquad \text{~~(3E-9)~~}$$

~~The load at instability is determined from the J versus load plot also shown schematically in Figure 3E-14. Thus, the three key curves in the tearing stability evaluation are: $J_{applied}$ versus $T_{applied}$, J_{mat} versus T_{mat} and $J_{applied}$ versus load. The determination of appropriate~~

~~J_{mat} versus T_{mat} or the material (J/T) curve has been already discussed in Subsection 3E.2.1. The $J_{applied}$ – $T_{applied}$ or the (J/T) applied curve can be easily generated through perturbation in the crack length once the $J_{applied}$ versus load information is available for different crack lengths. Therefore, only the methodology for the generation of $J_{applied}$ versus load information is discussed in detail.~~

3E.3.1.2 J Estimation Scheme Procedure

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~~The $J_{applied}$ or J as a function of load was calculated using the GE/EPRI estimation scheme procedure (References 3E-18 and 3E-19). The J in this scheme is obtained as sum of the elastic and fully plastic contributions:~~

$$J = J_e + J_p \qquad (3E\text{-}10)$$

~~The material true stress-strain curve in the estimation scheme is assumed to be in the Ramberg-Osgood format:~~

$$\left(\frac{\varepsilon}{\varepsilon_0}\right) = \left(\frac{\sigma}{\sigma_0}\right) + \alpha\left(\frac{\sigma}{\sigma_0}\right)^n \qquad (3E\text{-}11)$$

~~where σ_0 is the material yield stress, $\varepsilon_0 = \sigma/E_0$, and α and n are obtained by fitting the preceding equation to the material true stress-strain curve.~~

~~The estimation scheme formulas to evaluate the J-integral for a pipe with a throughwall circumferential flaw subjected to pure tension or pure bending are as follows:~~

~~**Tension**~~

$$J = f_1\left(a_e, \frac{R}{t}\right)\frac{P^2}{E} + \alpha\sigma_0\varepsilon_0 c\left(\frac{a}{b}\right)h_1\left(\frac{a}{b}, n, \frac{R}{t}\right)\left[\frac{P}{P_0}\right]^{n+1} \qquad (3E\text{-}12)$$

~~where:~~

$$f_1\left(\frac{a}{b}, n, \frac{R}{t}\right) = \frac{aF^2\left(\frac{a}{b}, n, \frac{R}{t}\right)}{4\pi R^2 t^2}$$

$$P_0 = 2\sigma_0 Rt\left[\pi - \gamma - 2\,\mathrm{asin}\left(\frac{1}{2}\sin\gamma\right)\right]$$

~~**Bending**~~

$$J = f_1\left(a_e, \frac{R}{t}\right)\frac{M^2}{E} + \alpha\sigma_0\varepsilon_0 c\left(\frac{a}{b}\right)h_1\left(\frac{a}{b}, n, \frac{R}{t}\right)\left[\frac{M}{M_0}\right]^{n+1} \qquad (3E\text{-}13)$$

~~where:~~

$$f_1\left(\frac{a}{b}, n, \frac{R}{t}\right) = \pi a\left(\frac{R}{I}\right)^2 F^2\left(\frac{a}{b}, n, \frac{R}{t}\right)$$

~~M_θ = $M_0\left[\cos\left(\frac{\gamma}{2}\right) - \frac{1}{2}\sin(\gamma)\right]$~~

~~The non-dimensional functions f and h are given in Reference 3E-19.~~

~~While the calculation of J for given α, n, σ_0 and load type is reasonably straightforward, one issue that needs to be addressed is the tearing instability evaluation when the loading includes both the membrane and the bending stresses. The estimation scheme is capable of evaluating only one type of stress at a time.~~

~~This aspect is addressed next.~~

3E.3.1.3 Tearing Instability Evaluation Considering Both the Membrane and Bending Stresses

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~~Based on the estimation scheme formulas and the tearing instability methodology just outlined, the instability bending and tension stresses can be calculated for various throughwall circumferential flaw lengths. Figure 3E-15 shows a schematic plot of the instability stresses as a function of flaw length. For the same stress level, the allowable flaw length for the bending is expected to be larger than the tension case.~~

~~When the applied stress is a combination of the tension and bending, a linear interaction rule is used to determine the instability stress or conversely the critical flaw length. The application of linear interaction rule is certainly conservative when the instability load is close to the limit load. The applicability of this proposed rule should be justified by providing a comparison of the predictions by the proposed approach (or an alternate approach) with those available for cases where the membrane and bending stresses are treated together.~~

~~The interaction formulas follow: (See Figure 3E-15)~~

~~**Critical Flaw Length**~~

$$\cancel{a_c = \frac{(\sigma_t)}{\sigma_t + \sigma_b} a_{c,t} + \frac{(\sigma_b)}{\sigma_t + \sigma_b} a_{c,b}} \qquad \text{(3E-14)}$$

~~where:~~

~~σ_t = applied membrane stress~~

~~σ_b = applied bending stress~~

~~$a_{c,t}$ = critical flaw length for a tension stress of ($\sigma_t + \sigma_b$)~~

~~$a_{c,b}$ = critical flaw length for a bending stress of ($\sigma_t + \sigma_b$)~~

~~**Instability Bending Stress**~~

$$\cancel{S_b = \left(1 - \frac{\sigma_t}{\sigma'_t}\right)\sigma'_b} \qquad \text{(3E-15)}$$

~~where:~~

~~S_b = instability bending stress for flaw length, a, in the presence of membrane stress, σ_t~~

~~σ_t = applied membrane stress~~

~~σ'_t = instability tension stress for flaw length, a~~

~~σ'_b = instability bending stress for flaw length, a~~

~~Once the instability bending stress, S_b, in the presence of membrane stress, σ_t, is determined, the instability load margin corresponding to the detectable leak-size crack (as required by LBB criterion in Subsection 3.6.3) can be calculated as follows:~~

$$\frac{\sigma_t + S_b}{\sigma_t + \sigma_b} \qquad \text{(3E-16)}$$

~~It is assumed in the preceding equation that the uncertainty in the calculated applied stress is essentially associated with the stress because of applied bending loads and that the membrane stress, which is generally due to the pressure loading, is known with greater certainty. This method of calculating the margin against loads is also consistent with the definition of load margin employed in Paragraph IWB-3640 of Section XI of Reference 3E-20.~~

3E.3.2 Application of (J/T) Methodology to Carbon Steel Piping

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~~From Figure 3E-3, it is evident that carbon steels exhibit transition temperature behavior marked by three distinct stages: lower shelf, transition, and upper shelf. The carbon steels generally exhibit ductile failure mode at or above upper shelf temperatures. This would suggest that a net-section collapse approach may be feasible for the evaluation of postulated flaws in carbon steel piping. Such a suggestion was also made in a review report prepared by the Naval Research Lab (Reference 3E-21). Low temperature (i.e., less than 51.7°C (125°F)) pipe tests conducted by GE (Reference 3E-22) and by Vassilaros (Reference 3E-23) which involved circumferentially cracked piping subjected to bending and/or pressure loading, also indicate that a limit load approach is feasible. However, test data at high temperatures, especially involving large diameter pipes, are currently not available. Therefore, a (J/T) based approach is used in the evaluation.~~

3E.3.2.1 Determination of Ramberg-Osgood Parameters for 288°C (550°F) Evaluation

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~~Figure 3E-7 shows the true stress-true strain curves for the carbon steels at 288°C (550°F). The same data is plotted here in Figure 3E-16 in the Ramberg-Osgood format. It is seen that, unlike the stainless steel case, each set for stress-strain data (i.e., data derived from one stress-strain curve) follows approximately a single slope line. Based on~~

~~the visual observation, a line representing $\alpha = 2$, $n = 5$ in Figure 3E-16 was drawn as representing a reasonable upper bound to the data shown.~~

~~The third parameter in the Ramberg-Osgood format stress-strain curve is σ_θ, the yield stress. Based on the several internal GE data on carbon steels, such as SA 333 Gr. 6 and SA 106 Gr. B, a reasonable value of 288°C (550°F) yield strength was judged as 238.6 MPa (34,600 psi). To summarize, the following values are used in this appendix for the (J/T) methodology evaluation of carbon steels at 288°C (550°F):~~

~~α = 2.0~~

~~n = 5.0~~

~~σ_θ = 238.6 MPa (34,600 psi)~~

~~E = 1.79×10^5 MPa (26×10^6 psi)~~

3E.3.2.2 Determination of Ramberg-Osgood Parameters for 216°C (420°F) Evaluation

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~~Figure 3E-17 shows the Ramberg-Osgood (R-O) format plot of the 177°C (350°F) true stress-strain data on the carbon steel base metal. Also shown in Figure 3E-17 are the GE data and SA 106 Gr. B at 204°C (400°F). Because the difference between the ASME Code Specified minimum yield strength at 177°C (350°F) and 216°C (420°F) is small, the 177°C (350°F) stress-strain data were considered applicable in the determination of R-O parameters for evaluation at 216°C (420°F).~~

~~A review of Figure 3E-17 indicates that the majority of the data associated with any one test can be approximated by one straight line.~~

~~It is seen that some of the data points associated with the yield point behavior fall along the y-axis. However, these data points at low strain level were not considered significant and, therefore, were not included in the R-O fit.~~

~~The 177°C (350°F) yield stress for the base material is given in Table 3E-3 as 261.2 MPa (37.9 ksi). Because the difference between the ASME Code specified minimum yield strengths of pipe and plate carbon steels at 216°C (420°F) and 177°C (350°F) is roughly 6.18 MPa (0.9 ksi), the σ_o value for use at 216°C (420°F) are chosen as 261.2-6.18 MPa (37.9-0.9 ksi) or 255.0 MPa (37 ksi). In summary, the following values of R-O parameters are used for evaluation of 216°C (420°F):~~

~~σ_θ = 255.1 MPa (37,000 psi)~~

~~α = 5.0~~

~~n = 4.0~~

3E.3.3 Modified Limit Load Methodology for Austenitic Stainless Steel Piping

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~~Reference 3E-24 describes a modified limit load methodology that may be used to calculate the critical flaw lengths and instability loads for austenitic stainless steel piping and associated welds. If appropriate, this or an equivalent methodology may be used in place of the (J/T) methodology described in Subsection 3E.3.1.~~

3E.3.4 Bimetallic Welds

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~~For joining austenitic stainless steels to ferritic steels, Ni-Cr-Fe Alloys 82 is generally used for weld metals (in selected cases stainless steel types 309L/308L may be used). The procedures recommended in Section 3E.3.3 for the austenitic stainless steel welds are also applicable to these weld metals. This is justified based on the common procedures adopted for flaw acceptance in the ASME Code Section XI, Article IWB-3600 and Appendix C, for both types of the welds. If other types of bimetallic metals are used, proper procedures should be used with generally acceptable justifications.~~

3E.5 LEAK RATE CALCULATION METHODS

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~~Leak rates of high pressure fluids through cracks in pipes are a complex function of crack geometry, crack surface roughness, applied stresses, and inlet fluid thermodynamic state. Analytical predictions of leak rates essentially consist of two separate tasks: calculation of the crack opening area, and the estimation of the fluid flow rate per unit area. The first task requires the fracture mechanics evaluations based on the piping system stress state. The second task involves the fluid mechanics considerations in addition to the crack geometry and its surface roughness information. Each of these tasks is now discussed separately considering the type of fluid state in ESBWR piping.~~

3E.4.1 Leak Rate Estimation for Pipes Carrying Water

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~~EPRI-developed computer code PICEP (Reference 3E-25) may be used in the leak rate calculations. The basis for this code and comparison of its leak rate predictions with the experimental data is described in References 3E-26 and 3E-27. This code has been used in the successful application of LBB to primary piping system of a PWR. The basis for flow rate and crack opening area calculations in PICEP is briefly described first. A comparison with experimental data is shown next.~~

~~Other methods (e.g., Reference 3E-28) may be used for leak rate estimation at the discretion of the applicant.~~

3E.4.1.1 Description of Basis for Flow Rate Calculation

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~~The thermodynamic model implemented in PICEP computer program assumes the leakage flow through pipe cracks to be isentropic and homogeneous, but it accounts for non-equilibrium "flashing" transfer process between the liquid and vapor phases.~~

~~Fluid friction caused surface roughness of the walls and curved flow paths has been incorporated in the model. Flows through both parallel and convergent cracks can be treated. The model uses some approximations and empirical factors, which were confirmed by comparison against test data because of the complicated geometry within the flow path.~~

~~For given stagnation conditions and crack geometries, the leak rate and exit pressure are calculated using an iterative search for the exit pressure starting from the saturation pressure corresponding to the upstream temperature and allowing for friction, gravitational, acceleration and area change pressure drops. The initial flow calculation is performed when the critical pressure is lowered to the backpressure without finding a solution for the critical mass flux.~~

~~A conservative methodology was developed to handle the phase transformation into a two-phase mixture or superheated steam through a crack. To make the model continuous, a correction factor was applied to adjust the mass flow rate of a saturated mixture to be equal to that of a slightly sub-cooled liquid. Similarly, a correction factor was developed to ensure continuity as the steam became superheated. The superheated model was developed by applying thermodynamic principles to an isentropic expansion of the single phase steam.~~

~~The code can calculate flow rates through fatigue or IGSCC cracks and has been verified against data from both types. The crack surface roughness and the number of bends account for the difference in geometry of the two types of cracks. The guideline for predicting leak rates through IGSCCs when using this model was based on obtaining the number of turns that give the best agreement for Battelle Phase II test data of Collier et al. (Reference 3E-29). For fatigue cracks, it is assumed that the crack path has no bends.~~

3E.4.1.2 Basic for Crack Opening Area Calculation

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~~The crack opening area in PICEP code is calculated using the estimation scheme formulas. The plastic contribution to the displacement is computed by summing the contributions of bending and tension alone, a procedure that underestimates the displacement from combined tension and bending. However, the plastic contribution is expected to be insignificant because the applied stresses at normal operation are generally such that they do not produce significant plasticity at the cracked location.~~

3E.4.1.3 Comparison Verification with Experimental Data

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~~Figure 3E-18 from Reference 3E-27 shows a comparison PICEP prediction with measured leak rate data. It is seen that PICEP predictions are virtually always conservative (i.e., the leak flow rate is under-predicted).~~

3E.4.2 Flow Rate Estimation for Saturated Steam

3E.4.2.1 Evaluation Method

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~~The calculations for this case were based on the maximum two-phase flow model developed by Moody (Reference 3E-30). However, in an LBB report, a justification should be provided by comparing the predictions of this method with the available experimental data, or a generally acceptable method, if available, should be used. The Moody model predicts the flow rate of steam-water mixtures in vessel blowdown from pipes (see Figure 3E-19). A key parameter that characterized the flow passage in the Moody analysis is fL/D_h, where f is the coefficient of friction, L, the length of the flow passage and D_h, the hydraulic diameter. The hydraulic diameter for the case of flow through a crack is 2δ where δ is the crack opening displacement and the length of the flow passage is t, the thickness of the pipe. Thus, the parameter fL/D_h in the Moody analysis was interpreted as ft/2δ for the purpose of this evaluation.~~

~~Figure 3E-20 shows the predicted mass flow rates by Moody for fL/D_h of 0 and 1. Similar plots are given in Reference 3E-30 for additional fL/D_h values of 2 through 100. Because the steam in the ESBWR main steam lines would be essentially saturated, the mass flow rate corresponding to the upper saturation envelope line is the appropriate one to use. Table 3E-5 shows the mass flow rates for a range of fL/D_h values for a stagnation pressure of 6.62 MPa (960 psi) which is roughly equal to the pressure in an ESBWR piping system carrying steam.~~

~~A major uncertainty in calculating the leakage rate is the value of f. This is discussed next.~~

3E.4.2.2 Selection of Appropriate Friction Factor

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~~Typical relationships between Reynolds number and relative roughness ε/D_h, the ratio of effective surface protrusion height to hydraulic diameter, were relied upon in this case. Figure 3E-21, from Reference 3E-31, graphically shows such a relationship for pipes. The ε/D_h ratio for pipes generally ranges from 0 to 0.50. However, for a fatigue crack consisting of rough fracture surfaces represented by a few mils, the roughness height ε at some location may be almost as much as δ. In such cases, ε/D_h would seem to approach one-half. There are no data or any analytical model for such cases, but a crude estimate based on the extrapolation of the results in Figure 3E-21 would indicate that f may be of the order of 0.1 to 0.2. For this evaluation an average value of 0.15 was used with the modification as discussed next.~~

~~For blowdown of saturated vapor, with no liquid present, Moody states that the friction factor should be modified according to~~

$$\cancel{f_g = f_{GSP}\left(\frac{\upsilon_f}{\upsilon_g}\right)^{1/3}} \qquad \cancel{(3E\text{-}17)}$$

~~where:~~

~~f_g = modified friction factor~~

~~f_{GSP} = factor for single phase~~

~~$\frac{\upsilon_f}{\upsilon_g}$ = liquid/vapor specific volume ratio evaluated at an average static pressure in the flow path~~

~~This correction is necessary because the absence of a liquid film on the walls of the flow channel at high quality makes the two-phase flow model invalid as it stands. The average static pressure in the flow path is going to be something in excess of 3.31 MPa (480 psia) if the initial pressure is 6.62 MPa (960 psia); this depends on the amount of flow choking and can be determined from Reference 3E-30. However, a fair estimate of $(v_f/v_g)^{1/3}$ is 0.3, so the friction factor for saturated steam blowdown may be taken as 0.3 of that for mixed flow.~~

~~Based on this discussion, a coefficient of friction of 0.15 x 0.3 =0.45 was used in the flow rate estimation. Currently experimental data are unavailable to validate this assumed value of coefficient of friction.~~

3E.4.2.3 Crack Opening Area Formulation

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~~The crack opening areas were calculated using LEFM procedures with the customary plastic zone correction. The loadings included in the crack opening area calculations were: pressure, weight, and thermal expansion.~~

~~The mathematical expressions given by Paris and Tada (Reference 3E-32) are used in this case. The crack opening areas for pressure (A_p) and bending stresses (A_b) were separately calculated and then added together to obtain the total area (A_c).~~

~~For simplicity, the calculated membrane stresses from weight and thermal expansion loads were combined with the axial membrane stress, σ_p, due to the pressure.~~

~~The formulas are summarized below:~~

$$\cancel{A_p = \frac{\sigma_p}{E}(2\pi R t)G_p(\gamma)} \qquad \cancel{(3E\text{-}18)}$$

~~where:~~

~~σ_p = axial membrane stress caused by pressure, weight and thermal expansion loads~~

~~E = Young's modulus~~

~~R = pipe radius~~

~~t = pipe thickness~~

~~λ = shell parameter = $\frac{a}{\sqrt{Rt}}$~~

~~a = half crack length~~

$$\cancel{G_p(\lambda) = \lambda^2 + 0.16\lambda^4 (0 \le \lambda \le 1)}$$

$$= 0.02 + 0.81\lambda^2 + 0.30\lambda^3 + 0.03\lambda^4 (1 \le \lambda \le 5) \quad (3E\text{-}19)$$

$$A_b = \frac{\sigma_b}{E} \cdot \pi \cdot R^2 \cdot \frac{(3 + \cos\theta)}{4} I_t(\theta) \quad (3E\text{-}20)$$

~~where:~~

~~σ_b = bending stress caused by weight and thermal expansion loads~~

~~θ = half crack angle~~

$$\cancel{I_t(\theta) = 2\theta^2 \left[1 + \left(\frac{\theta}{\pi}\right)^{3/2} \left\{ 8.6 - 13.3\left(\frac{\theta}{\pi}\right) + 24\left(\frac{\theta}{\pi}\right)^2 \right\} + \left(\frac{\theta}{\pi}\right)^3 \right.}$$

$$\left. \left\{ 22.5 - 75\left(\frac{\theta}{\pi}\right) + 205.7\left(\frac{\theta}{\pi}\right)^2 - 247.5\left(\frac{\theta}{\pi}\right)^3 + 242\left(\frac{\theta}{\pi}\right)^4 \right\} \right] (0 < \theta < 100^0) \quad \cancel{(3E\text{-}21)}$$

~~The plastic zone correction was incorporated by replacing a and θ in these formulas by a_{eff} and θ_{eff} which are given by~~

$$\cancel{\theta_{eff} = \theta + \frac{(K_{total})^2}{(2\pi R \sigma_Y)^2}}$$

$$a_{eff} = \theta_{eff} \cdot R \quad (3E\text{-}22)$$

~~The yield stress, σ_y, was conservatively assumed as the average of the code specified yield and ultimate strength. The stress intensity factor, K_{total}, includes contribution caused by both the membrane and bending stress and is determined as follows:~~

$$\cancel{K_{total} = K_m + K_b} \quad \cancel{(3E\text{-}23)}$$

~~where:~~

$$\cancel{K_m = \sigma_p \cdot \sqrt{a} . F_p(\lambda)}$$

$$\cancel{F_p(\lambda) = \left(1 + 0.3225\lambda^2\right)^{1/2} (0 \le \lambda \le 1)}$$

$$\cancel{= 0.9 + 0.25\lambda (1 \le \lambda \le 5)}$$

$$\cancel{K_b = \sigma_b \cdot \sqrt{\pi a} \cdot F_b(\theta)}$$

$$\cancel{F_b(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{3/2} - 13.6\left(\frac{\theta}{\pi}\right)^{5/2} + 20\left(\frac{\theta}{\pi}\right)^{7/2} (0 \le \theta \le 100^0)}$$

~~The steam mass flow rate, M, shown in Table 3E-5 is a function of parameter, ft/2δ. Once the mass flow rate is determined corresponding to the calculated value of this parameter, the leak rate in gpm can then be calculated.~~

3E.5 LEAK DETECTION CAPABILITIES

(deleted)

~~A complete description of various leak detection systems is provided in Subsection 5.2.5. The leakage detection system gives separate considerations to: leakage within the drywell and leakage external to the drywell. The limits for reactor coolant leakage are described in Subsection 5.2.5.4.~~

~~The total leakage in the drywell consists of the identified leakage and the unidentified leakage. The identified leakage is that from pumps, valve stem packings, reactor vessel head seal and other seals, which all discharge to the equipment drain sump. The Technical Specifications (TS) limit on the identified leak rate is 114 Liters/min (25 gpm).~~

~~The unidentified leak rate in the drywell is the portion of the total leakage received in the drywell sumps that is not identified as previously described. As specified in subsection 5.2.5.2, the detection capability for unidentified leak rate is 3.8 Liters/min (1 gpm). To cover uncertainties in leak detection capability, although it meets Regulatory Guide 1.45 guidelines, a margin factor of 10 is required per Reference 3E-24 to determine a reference leak rate. A reduced margin factor may be used if accounts can be made of effects of sources of uncertainties such as plugging of the leakage crack with particulate material over time, leakage prediction, measurement techniques, personnel, and frequency of monitoring. For the piping in drywell, a reference leak rate of 37.85 L/min (10 gpm) may be used, unless a smaller rate can be justified.~~

~~The sensitivity and reliability of leakage detection systems used outside the drywell must be demonstrated to be equivalent to Regulatory Guide 1.45 systems. Methods that have been shown to be acceptable include local leak detection, for example, visual observation or instrumentation. Outside the drywell, the leakage rate detection and the margin factor depend upon the design of the leakage detection systems.~~

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Table 3E-1

Electrodes and Filler Metal Requirements for Carbon Steel Welds

(deleted)

~~Base Material~~	~~P-No.~~	~~Process~~	~~Electrode or Specification~~	~~Filler Metal Classification~~
~~Carbon Steel to Carbon Steel; or~~	~~P-1 to P-1, P-3~~	~~SMAW~~	~~SFA-5.1~~	~~E7018~~
~~Low Alloy Steel~~	~~P-4 or P-5~~	~~GTAW PAW~~	~~SFA-5.18~~	~~E70S-2, E70S-3~~
		~~GMAW~~	~~SFA-5.18 SFA-5.20~~	~~E70S-2, E70S-3, E70S-6 E70T-1~~
		~~SAW~~	~~SFA-5.17~~	~~F72EM12K, F72EL12~~

Table 3E-2

Supplier Provided Chemical Composition and Mechanical Properties Information

(deleted)

		~~Chemical Composition~~					~~Mechanical Property~~		
~~Material~~	~~Product Form~~	~~C~~	~~Mn~~	~~P~~	~~S~~	~~Si~~	~~Sy MPa (ksi)~~	~~Su MPa (ksi)~~	~~Elongation (%)~~
~~SA 333 Gr. 6 Heat #52339~~	~~16 in. Sch. 80 Pipe~~	~~0.12~~	~~1.18~~	~~0.01~~	~~0.026~~	~~0.27~~	~~303.4 (44.0)~~	~~465.4 (67.5)~~	~~42.0~~
~~SA 516 Gr. 70 Heat #E18767~~	~~1.0 in. Plate~~	~~0.18~~	~~0.98~~	~~0.017~~	~~0.0022~~	~~0.25~~	~~320.5 (46.5)~~	~~486.1 (70.5)~~	~~31.0~~

~~Notes:~~

~~(1) Pipe was normalized at 898.9°C (1650°F). Held for 2 hours and air-cooled.~~

~~(2) Plate was normalized at 926.7°C (1700°F) for one hour and air-cooled.~~

Table 3E-3

Standard Tension Test Data at Temperature

(deleted)

~~Specimen Number~~	~~Material~~	~~Test Temperature~~	~~0.2% YS MPa (ksi)~~	~~UTS (%)~~	~~Elongation (%)~~	~~RA (%)~~
~~OW1~~	~~Pipe Weld~~	~~RT~~	~~455.7 MPa (66.1)~~	~~81.6~~	~~32~~	~~77.2~~
~~OW2~~	~~Pipe Weld~~	~~288°C (550°F)~~	~~406.8 MPa (59.0)~~	~~93.9~~	~~24~~	~~56.7~~
~~ITWL2~~	~~Plate Weld~~	~~288°C (550°F)~~	~~365.4 MPa (53.0)~~	~~91.4~~	~~34~~	~~51.3~~
~~IBL1~~	~~Plate Base~~	~~RT~~	~~309.6 MPa (44.9)~~	~~73.7~~	~~38~~	~~51.3~~
~~IBL2~~	~~Plate Base~~	~~177°C (350°F)~~	~~261.3 MPa (37.9)~~	~~64.2~~	~~34~~	~~68.9~~
~~IBL3~~	~~Plate Base~~	~~288°C (550°F)~~	~~235.2 MPa (34.1)~~	~~69.9~~	~~29~~	~~59.4~~
~~OB1~~	~~Pipe Base~~	~~RT~~	~~300.9 MPa (43.6)~~	~~68.6~~	~~41~~	~~67.8~~
~~OB2~~	~~Pipe Base~~	~~177°C (350°F)~~	~~291.0 MPa (42.2)~~	~~74.9~~	~~21~~	~~55.4~~
~~OB3~~	~~Pipe Base~~	~~288°C (550°F)~~	~~238.6 MPa (34.6)~~	~~78.2~~	~~31~~	~~55.4~~

Table 3E-4

Summary of Carbon Steel J-R Curve Tests

(deleted)

~~Number~~	~~Specimen ID~~	~~Size~~	~~Description~~	~~Temperature~~
~~(1)~~	~~OWLC-A~~	~~1T~~	~~Pipe Weld~~	~~288°C (550°F)~~
~~(2)~~	~~OBCL-1~~	~~1T~~	~~Pipe Base C-L Orientation~~	~~RT~~
~~(3)~~	~~OBLC2~~	~~1T~~	~~Pipe Base L-C Orientation~~	~~288°C (550°F)~~
~~(4)~~	~~OBLC3-B~~	~~1T~~	~~Pipe Base L-C Orientation~~	~~177°C (350°F)~~
~~(5)~~	~~BML-4~~	~~1T~~	~~Plate Base Metal, L-T Orientation~~	~~RT~~
~~(6)~~	~~BML4-14~~	~~2T~~	~~Plate Base Metal, L-T Orientation~~	~~RT~~
~~(7)~~	~~BML2-6~~	~~2T~~	~~Plate Base Metal, L-T Orientation~~	~~177°C (350°F)~~
~~(8)~~	~~BML1-12~~	~~2T~~	~~Plate Base Metal, L-T Orientation~~	~~288°C (550°F)~~
~~(9)~~	~~WM3-9~~	~~2T~~	~~Plate Weld Metal~~	~~RT~~
~~(10)~~	~~XWM1-11~~	~~2T~~	~~Plate Weld Metal~~	~~177°C (350°F)~~
~~(11)~~	~~WM2-5~~	~~2T~~	~~Plate Weld Metal~~	~~288°C (550°F)~~
~~(12)~~	~~HAZ~~	~~(Non-standard)~~ ~~Width = 7.09 cm (2.793")~~	~~Heat-Affected Zone, Plate~~	~~RT~~
~~(13)~~	~~OWLC-7~~	~~1T~~	~~Pipe Weld~~	~~RT~~

~~Notes:~~

~~(1) Pipe base metal, SA333 Gr. 6~~

~~(2) Plate base metal, SA516 Gr. 70~~

Table 3E-5

Mass Flow Rate Versus fl/D_h Values

(deleted)

~~fl/D_h~~	~~Mass Flow Rate, kg/s m^2 (lbm/sec-ft^2 M)~~
~~0~~	~~18540 (3800)~~
~~1~~	~~10740 (2200)~~
~~2~~	~~7810 (1600)~~
~~3~~	~~5615 (1150)~~
~~4~~	~~4490 (920)~~
~~5~~	~~3904 (800)~~
~~10~~	~~2830 (580)~~
~~20~~	~~1950 (400)~~
~~50~~	~~1270 (260)~~
~~100~~	~~903 (185)~~

(deleted)
Figure 3E-19. Pipe Flow Model

(deleted)
Figure 3E-20. Mass Flow Rates for Steam/Water Mixtures

(deleted)
Figure 3E-21. Friction Factors for Pipes