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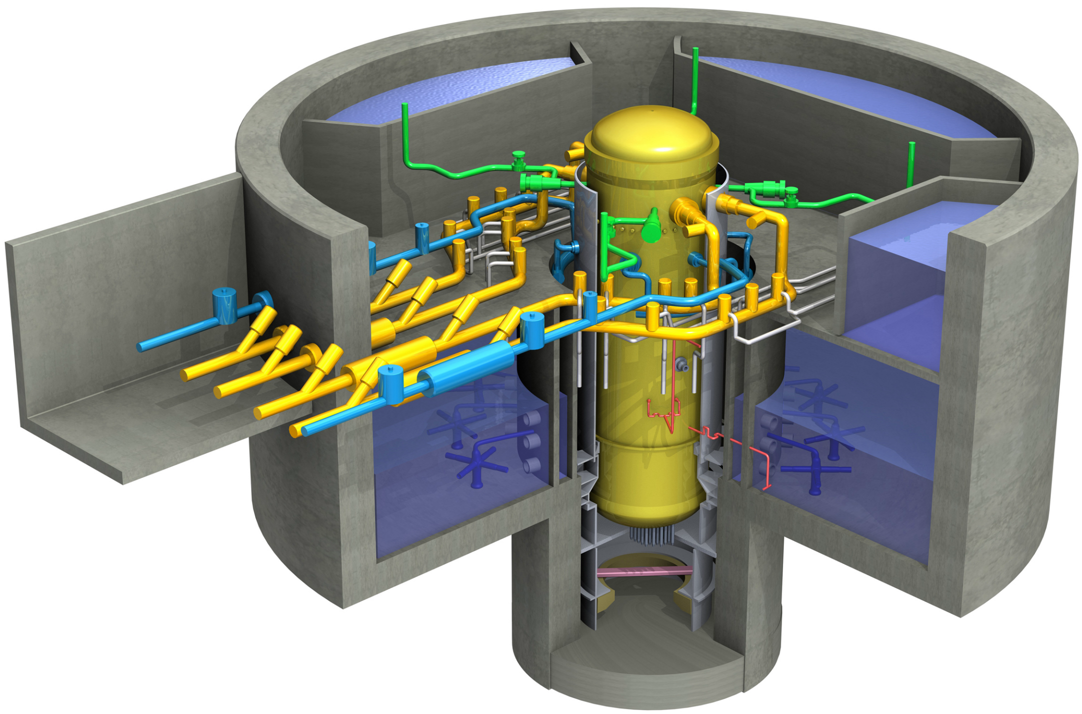

ESBWR Design Control Document
Tier 2

Chapter 1
Introduction and General Description of Plant
Appendices 1A-1D

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APPENDIX 1A RESPONSE TO TMI RELATED MATTERS

Table 1A-1 addresses the Three Mile Island (TMI) Action Plan Items listed in 10 CFR 50.34(f). Because the ESBWR includes design features different from the active plants considered in 10 CFR 50.34(f), consideration is given to all issues, in order to identify comparable ESBWR features which may address the issues.

1A.1 REFERENCES

1A-1 U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U. S. NRC Report NUREG-0660, Vols. 1 and 2, May 1980.

1A-2 U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," U. S. NRC Report NUREG-0737, November 1980.

1A-3 U. S. Nuclear Regulatory Commission, "Licensing Requirements for Pending Construction Permits and Manufacturing License Applications," NUREG-0718, Revision 1, June 1981.

1A-4 Letter from D. B. Waters, Chairman, BWR Owners' Group, to D. G. Eisenhut, NRC, "BWR Owners' Group Evaluation of NUREG-0737 Requirements II.K.3.16 and II.K.3.18," March 31, 1981.

1A-5 GE Nuclear Energy, "Regulatory Relaxation for BWR Post-Accident Sampling Stations (PASS)," NEDO-32991-A, Class I (Non-proprietary), August 2001.

1A-6 U.S. Nuclear Regulatory Commission, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Regulatory Guide 1.97, Revision 4, June 2006.

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(1)(i)	II.B.8	Levels 1 (Plant), 2 (Containment) & 3 (Site) PRAs to confirm meeting NRC Safety Goals.	A plant specific Probabilistic Risk Assessment (PRA) performed on the ESBWR design evaluates the plant in terms of core damage frequency and containment integrity. The PRA supports the design effort and establishes the capability of the design to meet established safety goals. The PRA contains Level 1 (Plant), Level 2 (Containment), and Level 3 (Site) PRA evaluations including internal and external events. In addition the PRA identified a number of design changes to improve the design of the ESBWR that have been incorporated.	ESBWR PRA, Chapter 19
10 CFR 50. 34(f)(1)(ii)	II.E.1.1	PWR Auxiliary Feedwater System evaluation.	Applicable to PWRs only. The ESBWR does not have comparable systems.	N/A
10 CFR 50. 34(f)(1)(iii)	II.K.2.16 and II.K.3.25	Reactor Coolant Pump Seal damage.	ESBWR has no Reactor Coolant Pump; ESBWR is a passive plant and utilizes natural circulation to drive coolant flow.	N/A
10 CFR 50. 34(f)(1)(iv)	II.K.3.2	Power Operated Relief Valves	Applicable to PWRs only. The ESBWR does not use this type of relief valve.	N/A

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(1)(v)	II.K.3.13	Separate High Pressure Coolant Injection (HPCI), High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) system initiation levels such that RCIC initiates at a higher water level than HPCI/HPCS.	The comparable ESBWR systems are the Automatic Depressurization System (ADS) / Gravity Driven Cooling System (GDCS) and the Isolation Condenser System (ICS). The ICS initiates at a higher level (Level 2) than the ADS/GDCS (Level 1). High pressure inventory control and reactor decay heat removal following reactor isolation for the ESBWR occur by means of the Isolation Condenser System (ICS). The ESBWR ICS replaces the traditional High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems found in most BWRs, thus eliminating concerns about cold water injection and system initiation. The ICS initiates automatically on high reactor pressure or low reactor water level (Level 2). ICS also initiate on Loss of All Feedwater or on closure of the Main Steam Isolation Valves (MSIVs) whenever the reactor mode switch is in the RUN position. ESBWR low pressure inventory control is via the GDCS in conjunction with the ADS, which initiates at a lower water level (Level 1) than the ICS.	5.4.6, 5.4.7, 6.3.2.7, 6.3.2.8, 7.1.6, 7.3.1, 7.3.5 and 7.4.4.

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(1)(vi)	II.K.3.16	Perform a study to identify practical system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only).	One of the key design criteria of the ESBWR is that Safety Relief Valves (SRVs) should not need to open during any Anticipated Operational Occurrences (transients) or Design Basis Accidents (DBAs) to protect against overpressure. SRVs are only expected to open in the event of an Anticipated Transient Without Scram (ATWS) or beyond design basis events. This is achieved through the use of the ICS. General Electric and the BWR Owners' Group responded to this requirement for earlier BWR models. Based on a review of the existing operating information on the challenge rate of relief valves, they concluded that the BWR/6 product line had already achieved the "order of magnitude" level of reduction in SRV challenge rate. The principal reason for this reduction is that the BWR/6 uses direct acting SRVs, not the pilot-operated design used in some earlier BWRs. The ESBWR uses a modern SRV design that has been proven not to experience the performance problems observed in earlier BWRs.	5.2.2

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(1)(vii)	II.K.3.18	Perform a feasibility and risk assessment study to determine the optimum Automatic Depressurization System (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWRs only)	The ESBWR ADS does not require manual activation to ensure adequate core cooling. Actuation of ADS equipment is performed automatically upon receipt of a persistent low reactor water level signal without need for operator action. Manual actuation is also possible. Automatic ADS complements manual ADS. Subsection 7.3.1.1 describes the logic and sequencing of the ADS in detail. For the above reasons, this TMI issue is considered resolved for the ESBWR design.	5.2.2.2, 6.3.2.8, and 7.3.1.1.
10 CFR 50. 34(f)(1)(viii)	II.K.3.21	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present (Applicable to BWR's only).	The comparable ESBWR systems are the ADS / GDCS and the ICS. The ADS is made up of SRVs and squib-activated DPVs. When the Depressurization Valves (DPVs) are actuated there is no way to manually stop depressurization and GDCS operation. Since the core is never uncovered the issues relating to spray cooling are not applicable to the ESBWR. This TMI item applies to low pressure inventory control systems (Core Spray and Low Pressure Coolant Injection) that can be stopped by the operator. Once the ESBWR low pressure injection system, GDCS, is initiated, the operator does not have the	5.4.6, 6.3.2.7, 6.3.2.8

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			ability to stop it from completing the initiation sequence. Therefore, this TMI item is not applicable to the ESBWR.	
10 CFR 50. 34(f)(1)(ix)	II.K.3.24	Provide space cooling for RCIC, HPCI/HPCS systems for 2 hours following complete loss of offsite power. (Applicable to BWR's only).	The ESBWR ICS replaces the traditional HPCI and RCIC Systems found in most BWRs. The ICS does not rely on active pumps to remove excess sensible and core decay heat. Each isolation condenser is located in a subcompartment of the IC/PCCS pool, and requires no additional space cooling other than that provided by the surrounding water in the IC/PCCS pool. If all of the safety-related power supplies used to start the isolation condensers were to fail, then all available isolation condensers automatically start into operation because of the "fail open" actuation of the condensate return bypass valves on loss of electrical power to the solenoids which control the pneumatically actuated valves. Therefore, this TMI item is considered resolved for the ESBWR design.	5.4.6, 5.4.7
10 CFR 50. 34(f)(1)(x)	II.K.3.28	Ensure that ADS valves, accumulators and associated equipment will be capable of	The ESBWR ADS is made up of SRVs and squib-activated DPVs. When the DPVs are actuated there is no way to close the DPVs	5.2.2.2, 6.3.2.8, 7.1.6.1, and

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		performing its intended functions during and following an accident.	until the valves are refurbished. The ADS utilizes the SRVs and the DPVs for depressurization of the reactor. Each of the ADS SRVs is equipped with a pneumatic accumulator and check valve for the ADS and manual opening functions. These accumulators assure that the valves can be opened following failure of the gas supply to the accumulators. The accumulator capacity is sufficient for one actuation at drywell design pressure. The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology. The DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The SRVs and DPVs and associated controls and actuation circuits are located or protected so that their function cannot be impaired by consequential effects of accidents. ADS components are qualified to withstand the harsh environments postulated for design basis accidents inside the containment, including high temperature,	7.3.1.1.

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			pressure, and radiation environments.	
10 CFR 50. 34(f)(1)(xi)	II.K.3.45	Evaluate depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only)	The ESBWR ADS DPVs are sized such that vessel depressurization and cooldown is slow enough that vessel integrity limits are not exceeded. A comprehensive thermal analysis was performed considering the effect of blowdown and the GDCS reflooding. Hypothetical ESBWR Accidents are much slower than those of previous BWR Product Line plants. Slower depressurization methods than a full ADS actuation are only possible in situations where nonsafety-related sources of high pressure makeup water remain available during the depressurization. Design basis accident events require the full ADS actuation.	5.3.2.1, 5.3.2.2, and 5.3.3.
10 CFR 50. 34(f)(1)(xii)	II.B.8	Include a hydrogen control system that satisfies the requirements of 10 CFR 50.34 (f)(2)(ix). As a minimum consider hydrogen ignition and post-accident inerting.	It is GEH's position that this TMI item has been superseded by the revisions to 10 CFR 50.44. The ESBWR utilizes a nitrogen inerted containment and a passive autocatalytic recombiner system to comply with this regulation and therefore complies with this TMI item.	6.2.5

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(i)	I.A.4.2	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs	Simulator capability that correctly models the control room and includes the capability to simulate small-break Loss-of-Coolant-Accidents (LOCAs) is used to perform the reactor operator training discussed in Section 13.2.	13.2 and 18.10
10 CFR 50. 34(f)(2)(ii)	I.C.9	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures.	Plant procedures are discussed in Section 13.5.	13.2.1, 13.3, 13.5, 18
10 CFR 50. 34(f)(2)(iii)	I.D.1	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts.	State-of-the-art human factor principles have been incorporated into the ESBWR control-room design. The design of the ESBWR control room utilizes accepted human factors engineering principles, incorporating the results of a full systems analysis similar to that described in Appendix B of NUREG-0700. An integrated program plan, entitled “Design of Controls, Instrumentation and Man-Machine Interfaces,” was prepared and implemented to incorporate human factors engineering principles and to achieve an integrated design of the control and instrumentation systems and operator interfaces of the	18

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			ESBWR. This plan and the associated procedures provided guidance for the conduct of the ESBWR control and instrumentation and Man-Machine Interface Systems (MMIS) design development activities including definition of the standard design features of the control room MMIS described in Subsection 18.2.2. Chapter 18 describes the ESBWR MMIS design goals and bases, the standard MMIS design features and the detailed MMIS design and implementation process, with embedded design acceptance criteria, for the ESBWR standard plant operator interface. A detailed control-room design review specified in NUREG-0737 is not required by Standard Review Plan (SRP) Section 18.1.	
10 CFR 50. 34(f)(2)(iv)	I.D.2	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	The ESBWR Control Room Design incorporates the features that display to operators a set of parameters responding to the symptom driven emergency procedure guidelines defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded.	18.1.1, 7.1.3.3.5, 7.1.4.4, 7.1.5.1.2, 7.1.5.2.4.1, 7.1.5.3.1, 7.1.6.1 and 7.8.3.1

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			The principal functions of the Safety Parameter Display System (SPDS), as required by Supplement 1 to NUREG-0737, are integrated into the control room operator interface design, as permitted by SRP Section 18. The ESBWR control room operator interface design incorporates the SPDS function as part of the plant status summary information which is continuously displayed on the fixed-position displays on a large display panel, and also incorporates the use of on-screen control video display units (VDUs), independent of the plant computer, for control and monitoring of both safety-related and nonsafety-related systems. Other VDUs, driven by the plant computer, are available for monitoring of safety-related systems and monitoring and control of nonsafety-related systems.	
10 CFR 50. 34(f)(2)(v)	I.D.3	Provide for automatic indication of the bypassed and operable status of safety systems.	ESBWR design of I&C provides automatic indication of the bypasses and inoperable status of safety systems.	7.2.1.4, 7.1.4.4, 7.1.5.3.1, 7.1.6.1 (Q-DCIS); 7.2.1.3.1, 7.2.1.3.6

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
				(RPS); 7.2.2.3.1, 7.2.2.3.6 (NMS); 7.2.3.3.1, 7.2.3.3.6 (SPTM); 7.3.1.1.3.1, 7.3.1.1.3.6 (ADS); 7.3.1.2.3.1, 7.3.1.2.3.6 (GDCS); 7.3.3.3.1, 7.3.3.3.6 (LD&IS); 7.3.4.3.1, 7.3.4.3.6 (CRHS); 7.3.5.3.1 (SSLC/ESF); 7.3.6.3.1, 7.3.6.3.6 (VB); 7.4.1.3.1 (SLC);

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
				7.4.2.3.1 (RSS); 7.4.4.3.1 (ICS); 7.4.5.3.1, 7.4.5.3.6 (HP CRD); 7.5.1.3.1 (PAM); 7.5.2.3.1, 7.5.2.3.6 (CMS); 7.5.3.3.1, 7.5.3.3.6 (PRMS); 7.6.1.3.1, 7.6.1.3.5 (HP/LP Interlocks)
10 CFR 50. 34(f)(2)(vi)	II.B.1	Provide the capability of high point venting of noncondensable gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated	During reactor operation the ESBWR design provides continuous venting from the Reactor Pressure Vessel (RPV) head and the isolation condensers driven by the differential pressure between the primary system pressure and a downstream steamline location, where the non-condensable gases	5.4.6 and 5.4.12

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or unacceptable challenge to containment integrity.	are extracted and swept to the main condenser, where the gasses are extracted. The capability to vent the ESBWR reactor coolant system when the vessel is isolated is provided by the safety/relief valves and reactor vessel head vent line. The head vent line is isolated from the Equipment and Floor Drain System (EFDS) with two normally closed valves during reactor power operation. These vent and purge lines are not required to assure natural circulation core cooling.	
10 CFR 50. 34(f)(2)(vii)	II.B.2	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect equipment from the radiation environment.	The Alternate Source Term (AST) contained in Reg. Guide 1.183 has superseded the TID-14844 source term. The AST is used for radiation design issues in the ESBWR. Reviews of ESBWR spaces requiring post-accident access reveal that each area has low post-LOCA radiation levels. A review of the radiation and shielding of the ESBWR post-accident operations has been made. It has been found that there is adequate access to areas potentially requiring post-accident access and that safety equipment is adequately protected. An evaluation of post-accident radioactive	Appendix 1B, 3.1.2, 3.11, 12.3.5, 12.3.6, Figures 12.3-43 through 12.3-51, and 15.4.1.3.

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			sources concluded that the ESBWR design limits potential radiation exposure from accidents both to plant personnel and to the public by the use of passive safety features and holdup in the containment. Potential releases in the radwaste building are contained by isolating the radwaste building atmosphere and containing any water releases in the building, which is seismically qualified and designed to prevent any potential water releases from high activity areas. Additional details relating to plant radiation sources can be found in Section 12.2. The locations requiring access to mitigate the consequences of an accident during the post-accident period are the control room, the technical support center, electrical equipment rooms for divisions 1, 2, 3 and 4, the Standby Liquid Control system pump room, nonsafety-related Distributed Control and Information System (DCIS) rooms, the remote shutdown panel rooms, the health physics facility (counting room), diesel generator control rooms, Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) cross-tie line spectacle flanges and manual	

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			isolation valves, and the IC/PCCS and fuel pool refill valves. Each area has low post-LOCA radiation levels. The results of the evaluations are reflected in the radiation zone maps (Figures 12.3-43 through 12.3-51) and demonstrate that personnel doses will be within regulatory guidelines. Areas in the reactor building requiring post-accident access are all located off the controlled access way and contamination is limited to air infiltration from the containment and contaminated systems. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from monitoring systems such as the Process Sampling System (PSS). An environmental qualification program for safety-related mechanical and electrical equipment to demonstrate their capability to perform their required functions when exposed to the environmental conditions (including accident and post-accident conditions) in their respective locations is described in Section 3.11. Radiation shielding is designed to keep radiation doses to equipment below levels at which	

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			disabling radiation damage occurs.	
10 CFR 50. 34(f)(2)(viii)	II.B.3	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.	The ability for the ESBWR to obtain and analyze samples is based on the approach provided in NEDO-32991-A (Reference 1A-5), which provides the basis for exceptions to NUREG-0737 (Reference 1A-2) and Regulatory Guide 1.97 (Reference 1A-6). The ESBWR Containment Monitoring System (CMS) and Process Sampling System (PSS) provide contingency plans for obtaining and analyzing highly radioactive reactor coolant, suppression pool, and containment atmospheric samples. The Process Sampling System described in Subsection 9.3.2 meets the requirements of this position with the following exception. The upper limit of activity in the samples at the time they are taken is as follows: Liquid sample 1 Ci/g (37,000 MBq/ml); and Gas Sample 0.1 Ci/cm^3(3700 MBq/ml) The CMS monitors the atmosphere in the containment for high gross gamma radiation levels and for high concentration levels of oxygen and hydrogen during post-accident conditions. Also, these three parameters are	7.5.2, 7.5.3, 9.3.2 and 11.5.

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			monitored during normal reactor operations to evaluate the integrity and safe conditions of the containment. A detailed description of the PSS and the CMS can be found in Subsections 9.3.2 and 7.5.2 respectively. Means to reduce radiation exposure are provided, such as shielding, remotely operated valves, and sample transporting casks. Acceptance Criterion II.K.5 of SRP Subsection 9.3.2 requires the capability of sampling liquids of 10 Ci/g. The ESBWR design has the capability of sampling liquids of 1Ci /g. Sampling and area radiation measurement would be performed. If levels are above safe limits, handling of samples is delayed. The Process Radiation Monitoring System (PRMS) identifies the various gaseous and liquid process streams and effluent release paths or points to be monitored and sampled, and defines the required instrumentation for detection and measurement of the radioactive contents of these streams. The PRMS alerts operating personnel to excessive radiation levels and automatically initiates the required protection action to	

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			isolate radioactivity releases to the environs. The PRMS is designed for operability during and following an accident. A detailed description of the PRMS can be found in Subsection 7.5.3 and in Section 11.5.	

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(ix)	II.B.8	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide with reasonable assurance that: (A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion. (B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of	It is GEH's position that this TMI item has been superseded by the revisions to 10 CFR 50.44. The ESBWR utilizes a nitrogen inerted containment and a passive autocatalytic recombiner system to comply with this regulation and therefore complies with this TMI item.	6.2.5

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		appropriate mitigating features. (C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system. (D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.		
10 CFR 50. 34(f)(2)(x)	II.D.1	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves … for all fluid conditions expected under operating conditions, transients and accidents.	The overpressure protection system, of which the SRVs are a part, is capable of accommodating the most severe pressurization Anticipated Operational Occurrence (transient). The ESBWR pressurization is mild relative to previous BWR designs because of the large steam	5.2.2.2, 6.3.2.8, 7.1.6.1, 7.3.1.1, 7.3.5.3.1, and 14.2.8.1.1.

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		Consideration of ATWS conditions shall be included.	volume in the chimney and vessel head, which mitigates the pressurization and does not result in opening of relief valves prior to isolation condenser initiation. A detailed description of the safety evaluation of Anticipated Operational Occurrences (transients) for the overpressure protection system can be found in Subsection 5.2.2. The inspection and testing of applicable SRVs utilizes a quality assurance program, which complies with Appendix B of 10 CFR 50. The number for SRVs provided is sufficient to limit the pressurization of the RPV to less than code allowables in the event of an ATWS. The SRVs are tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The conducted tests include hydrostatic, steam leakage, full flow pressure and blowdown, and response time testing. The valves are installed with setpoints as received from the factory. The valve manufacturer certifies that design and performance requirements, including	

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Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			capacity and blowdown, have been met. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic initiation signals for power actuation of each ADS SRV are verified during the preoperational test program described in Subsection 14.2.8.1.1. It is not feasible to test the SRV setpoints while the valves are in place. The valves can be removed for maintenance or bench testing and reinstalled during normal plant shutdowns. The valves will be tested to check set pressure in accordance with the requirements of the plant Technical Specifications. The external and flange seating surfaces of the SRVs are 100% visually inspected when any valve is removed for maintenance or bench testing during normal plant shutdown. The ESBWR ADS is made up of SRVs and squib-activated DPVs. When the DPVs are actuated there is no way to close the DPVs until the valves are refurbished.	

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TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(xi)	II.D.3	Provide direct indication of relief and safety valve position (open or closed) in the control room.	Direct indication of SRV and DPV position (open or closed) is provided in the main control room. SRV position is indicated in the control room in full compliance with this requirement.	7.1.6.1, 7.3.1.1 and 7.3.5.3.1
10 CFR 50. 34(f)(2)(xii)	II.E.1.2	Provide automatic and manual auxiliary feedwater (AFW) system initiation and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have an auxiliary feedwater system.	N/A
10 CFR 50. 34(f)(2)(xiii)	II.E.3.1	Provide pressurizer heater power supply and associated motive and control power … . (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have comparable systems.	N/A
10 CFR 50. 34(f)(2)(xiv)	II.E.4.2	Provide containment isolation systems that: (A) Ensure all non-essential systems are isolated automatically by the containment isolation system, (B) For each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) Do not result in reopening of the containment isolation valves on	The ESBWR Containment Isolation function meets the NRC requirements, including the post-TMI requirements. In general, this means that two barriers are provided. Redundancy and physical separation are required in the electrical and mechanical design of the containment isolation system to ensure that no single failure in the system prevents it from performing its intended functions. Electrical redundancy is provided	3.1.5, 5.2.5, 6.2.4, 7.1.6.1, 7.3.3, and 7.4.5.3.6

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		resetting of the isolation signal, (D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operations, and (E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.	for each set of isolation valves, such that the unavailability of any two safety-related electrical divisions will not prevent isolation from occurring. Electrical cables for isolation valves in the same line are routed separately. Cables are selected and based on the specific environment to which they may be subjected (e.g., magnetic fields, high radiation, high temperature and high humidity). Classification of structures, systems and components for the ESBWR design is addressed in Section 3.2 and identified in Table 3.2-1. The basis for classification is also presented in Section 3.2. The containment isolation function, automatically closes fluid penetrations of fluid systems that are not required for emergency operation. The design of the control systems for automatic containment isolation valves ensures that resetting the isolation signal does not result in the automatic reopening of containment isolation valves. Actuation of the containment isolation system is automatically initiated by the Leak Detection and Isolation System (LD&IS) at	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			specific limits defined for reactor plant operation. The LD&IS (described in Subsections 5.2.5 and 7.3.3) is designed to detect, monitor and alarm leakage inside and outside the containment, and automatically initiates the appropriate protective action to isolate the source of the leak. Various plant variables are monitored, including pressure, and these are used in the logic to isolate the containment. The drywell pressure is monitored by four divisional channels, using pressure transmitters to sense the drywell atmospheric pressure from four separate locations. A pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within the drywell. A high pressure indication is alarmed in the main control room, initiates reactor scram and, with the exception of the MSIVs, closes the containment isolation valves in certain designated process lines. The alarm and initiation setpoints of the LD&IS are set to initiate containment isolation at the minimum values compatible with normal operating conditions for containment penetrations containing process lines that are not required for emergency operation. The value for this setpoint is based on the	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			analytical limit used in safety analyses. All ESBWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main purge valves are fail-closed and are verified to be closed at a frequency interval of 31 days as defined in the plant technical specifications. All purge and vent valves are pneumatically operated, fail closed and receive containment isolation signals. Bleed valves and makeup valves can be remote manually opened in the presence of an isolation signal, by utilizing override control if continued inerting is necessary. In the ESBWR design, redundant primary containment isolation valves (purge and vent) close automatically upon receipt of an isolation signal from the LD&IS. The LD&IS is a four-divisional system designed to detect and monitor leakage from the reactor coolant pressure boundary, and, in certain cases, isolates the source of the leak by initiating closure of the appropriate containment isolation valves. Various plant variables are monitored, including radiation level, and these are used in the logic to initiate alarms and the required control signals for containment isolation. High	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			radiation levels detected in the reactor building Heating, Ventilation and Air Conditioning (HVAC) air exhaust or in the refueling area air exhaust automatically isolates the containment purge and vent isolation valves.	
10 CFR 50. 34(f)(2)(xv)	II.E.4.4	Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions.	The ESBWR design includes a capability for containment purging/venting designed to minimize the purging time consistent with As Low As Reasonably Achievable (ALARA) principles for occupational exposure. The system provides high assurance that the purge system will reliably isolate under accident conditions	7.3.3, 7.7.7.3.1 and 6.2.5.2
10 CFR 50. 34(f)(2)(xvi)	II.E.5.1	Establish a design criterion for the allowable number of actuation cycles of the ECCS and RPS consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only).	Applicable to B&W designs only. The ESBWR design includes criteria for the number of actuation cycles for the passive cooling components, which include both Anticipated Operational Occurrences (transients) and accidents.	3.9
10 CFR 50. 34(f)(2)(xvii)	II.F.1	Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C)	The Containment Monitoring System provides the ability to measure containment pressure, containment water level, containment hydrogen and oxygen levels,	7.5.2, and 7.5.3

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples.	and radiation levels. The Process Radiation Monitoring System monitors the radiation levels in gaseous streams at their release points.	
10 CFR 50. 34(f)(2)(xviii)	II.F.2	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as … a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's.	The detection of conditions indicative of inadequate core cooling is provided in the ESBWR design by the direct water level instrumentation system. Coolant level in the RPV is measured by both wide range and fuel zone instruments. The four divisions of wide range instruments cover the range from above the core to the main steam lines. The four channels of fuel zone instruments cover the range from below the core to the top of the steam separator. The RPV water level is the primary variable indicating the availability of adequate core cooling. Indication of water level by the differential pressure method is acceptable	4.6, 7.5.1, 7.7.1.2.2 and 18.1

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			(without diverse methods of sensing and indication) because adequate redundancy and unambiguity is provided from the bottom of the core support plate to the centerline of the main steam lines. The ESBWR has addressed the issue regarding erroneously high water level indication upon vessel depressurization due to the release of dissolved non-condensable gases in the reference leg. The ESBWR water level instrumentation system design includes a constant metered addition of purge water from the Control Rod Drive (CRD) hydraulic system to prevent the build-up of dissolved gasses in the fixed leg. This is consistent with the approved ABWR design as well as the modifications made by the majority of the BWR fleet.	
10 CFR 50. 34(f)(2)(xix)	II.F.3	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.	The ESBWR is designed in accordance with Regulatory Guide 1.97, (Reference 1A-6). A detailed assessment of the Regulatory Guide is found in Section 7.5.	T 1.9-21, and 7.5.1
10 CFR 50. 34(f)(2)(xx)	II.G.1	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: … . (Applicable to PWRs only)	Applicable to PWRs only. The ESBWR does not have comparable design features to PWR pressurizers.	N/A

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(xxi)	II.K.1.22	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only)	There are no short term manual actions which must be taken. Sufficient systems exist to automatically mitigate the consequences of a loss of feedwater event. An analysis was performed for a loss of feedwater event. The sequence of events is described within Section 15.2.5.3, and is summarized below. In the event that the main feedwater system is not operable, a reactor scram and initiation of the ICS will occur either due to 1) A detected Loss of All Feedwater, or 2) Reactor water level will fall due to void collapse, boil-off and absence of makeup water. When Level 3 is reached, a reactor scram is automatically initiated. Reactor water level continues to decrease due to void collapse, boil-off until the low-low level setpoint, Level 2, is reached. At this point, reactor isolation also occurs, but with a time delay for the MSIVs. When an initiation signal is received by the ICS, the condensate return valves will open in 30 seconds, placing the ICS in full operation at which time water level stabilizes. (High pressure CRD makeup if available will prevent the water level from	7.2, 7.3, 7.4.4 and 15.2.5.3

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			falling to a point where ADS and GDCS are initiated.) If not enough of the four isolation condensers are operable, the SRVs will open on high vessel pressure approximately five minutes later. The SRVs open and close to maintain vessel pressure. When reactor low water Level 1 is reached, an ADS timer is initiated. When the ADS timer is timed out, the ADS and Standby Liquid Control (SLC) system actuation sequence is initiated, and the GDCS timer is initiated. When the GDCS timer is timed out, the GDCS injection valves open. Vessel pressure then decreases below the static head of GDCS, and the GDCS reflooding flow into the vessel begins. The core remains covered throughout the sequence of events and no core heatup occurs.	
10 CFR 50. 34(f)(2)(xxii)	II.K.2.9	Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only).	The ESBWR does not have a system comparable to the B&W Integrated Control System.	N/A

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(xxiii)	II.K.2.10	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on the loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only).	The ESBWR Anticipated Operational Occurrences (transients) are generally much slower than even previous BWR designs. However, due to limited high pressure make-up, a reactor trip and initiation of the Isolation Condenser Systems (ICS) will occur in response to a Loss of All Feedwater event. These are anticipatory trips actuated directly on loss of power to two of the four main feedwater pumps. The ESBWR includes as part of the standard plant design 110% bypass capacity for the main turbine. Scram only occurs on a turbine trip if an insufficient number of bypass valves open within a prescribed time period.	7.1.6, 7.2.1, 7.3, 7.4.4, 10.4.4.1, and 15.2.5.3.
10 CFR 50. 34(f)(2)(xxiv)	II.K.3.23	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWRs only).	Recording of water level is included in the Main Control Room (MCR). Water level measurements are from the wide and fuel range water level instruments. See the discussion of 10 CFR 50.34(f)(2)(xviii) for more detail.	7.5.1
10 CFR 50. 34(f)(2)(xxv)	III.A.1.2	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a near-site	Space for the Technical Support Center is included in the Standard Design on the ground floor of the Electrical Building. The space provided is in conformance with	Figure 1.2-26, and 13.3

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		Emergency Operations Facility.	NUREG-0696 Provisions for an onsite Operational Support Center, and a near-site Emergency Operations Facility are discussed in Section 13.3.	
10 CFR 50. 34(f)(2)(xxvi)	III.D.1.1	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.	Leakage is reduced to as-low-as-practical levels for all required post-accident systems outside the containment that could contain highly radioactive fluid using a program that consists of: • Monitoring drain sumps to ascertain gross leakage occurring from systems included in this program. • Inspecting miscellaneous components (e.g., vents, drains, valve packing, valve packing leakoffs, pump packing, pump gland seal leakoffs, etc.) for leakage during initial system startup as part of the system preoperational test, and reducing any detected leakage to as-low-as-practical levels. After fuel load these same components are monitored as part of a surveillance test program. • Performing indirect inspections or a suitable substitute in situations where it is not possible, practical or permissible	5.4.7 and Chapter 16 TS Subsection 5.5.2

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			(e.g., due to high radiation) to make direct inspections. Examples of indirect inspection techniques include inspecting floor areas for wetting and monitoring the associated equipment or floor drain sumps for excessive flow or fill rates. The following ESBWR systems perform the design functions mentioned in the clarification section of NUREG-0737, Item III.D.1.1, and could contain radioactive material outside the primary containment boundary: 1) Isolation Condenser System 2) Fuel and Auxiliary Pools Cooling System 3) Containment Monitoring System 4) Reactor Water Cleanup/Shutdown Cooling System The portion of ICS outside containment is submerged during normal operation. Consequently, it is not accessible to plant personnel under post-accident conditions or for routine surveillance during normal plant operation.	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(2)(xxvii)	III.D.3.3	Provide for monitoring of inplant radiation and airborne radioactively as appropriate for a broad range of routine and accident conditions.	The ESBWR provides three systems to monitor area radiation and airborne radioactivity: Containment Monitoring System , Process Radiation Monitoring System , and Area Radiation Monitoring System.	7.5.2, 7.5.3, and 7.5.4
10 CFR 50. 34(f)(2)(xxviii)	III.D.3.4	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release, and make necessary design provisions to preclude such problems.	Safe occupancy of the ESBWR control room during abnormal conditions is provided for in the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a design basis accident for the duration of the accident. The Control Room Habitability Area HVAC Subsystem has redundant equipment, which includes High Efficiency Particulate Air (HEPA) filters, Carbon filters and radiation detectors with appropriate alarms and interlocks. If any hazards exist at the normal control room ventilation intake, habitability is assured by a redundant set of safety-related Emergency Filter Units that, upon isolation of the control room habitability area, provide a positive air purge. In the unlikely event that the control room must be vacated and access is restricted,	3.1.2, 6.4.2, 9.4.1, and 15.4

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
			instrumentation and controls are provided outside the control room, which can be utilized to initiate reactor shutdown, maintain a safe shutdown condition and achieve subsequent cold shutdown of the reactor.	
10 CFR 50. 34(f)(3)(i)	I.C.5	Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant.	The ESBWR design engineers are continually involved in reviewing industry experience from sources such as NRC Bulletins, Licensee Event Reports, NRC request for information letters to holders of operating licenses for nuclear power reactors, Federal Register information, and generic letters. A general description of the administrative procedures involved is provided in Section 18.3.2.	13.2.3, 13.5.2 and 18.3.2
10 CFR 50. 34(f)(3)(ii)	I.F.1	Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR Part 50 includes all structures, systems, and components important to safety.	The ESBWR Quality Assurance Plan is described in Chapter 17. Structures, systems, and components are classified as described in Section 3.2.	3.2, and 17

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(3)(iii)	I.F.2	Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities.	The Quality Assurance Program described in Chapter 17 meets the requirements of issue I.F.2 as they apply to the design of the ESBWR.	17

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
10 CFR 50. 34(f)(3)(iv)	II.B.8	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system.	The Containment Inerting System provides the capability to vent the containment via a pathway that connects the wetwell airspace to the Reactor Building / Fuel Building stack. This pathway can be opened in the event that the operators determine that venting is required and provides a fission product release at an elevated point at a time prior to containment structural failure. Having the connection point in the wetwell airspace forces the escaping fission products through the suppression pool. In a core damage event initiated by an Anticipated Operational Occurrence (transient), in which the vessel does not fail, fission products are directed to the suppression pool via the SRVs, DPVs via the connecting vents, ICS or PCCS, scrubbing any potential release. Therefore there is no need for any dedicated penetrations to be provided. However, the Containment Inerting System is supplemented by providing a manual, hardened, and elevated vent path.	6.2.5.2 and Figure 6.2-29
10 CFR 50. 34(f)(3)(v)	II.B.8	Provide preliminary design information at a level of detail consistent with that normally required at the construction permit	The ESBWR has a concrete containment that meets the requirements of this provision. Compliance with Reg. Guide 1.7	6.2.5

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		stage of review sufficient to demonstrate that: (II.B.8) (A)(1) Containment integrity will be maintained (i.e., for steel containment by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instablity is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a	demonstrates that this issue has been satisfactorily addressed. See further detailed discussion in the response to 10 CFR 50.34(f)(2)(ix).	

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions. (2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. (B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system		

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		(assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category. (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.		
10 CFR 50. 34(f)(3)(vi)	II.E.4.1	For plant designs with external hydrogen recombiners, provide	The ESBWR does not have external hydrogen recombiners, therefore, this	N/A

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.	requirement is not applicable.	
10 CFR 50.34 (f)(3)(vii)	II.J.3.1	Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the	The ESBWR design team has developed a management plan for the ESBWR project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels. The procedures to be used in the construction and operation phases of the plant are discussed in Section 13.5. Startup procedures are discussed in Section 14.2.	13.5, 14.2

Table 1A-1

TMI Action Plan Items

Regulation	TMI Item	Description	ESBWR Resolution	Associated Location(s)
		preparation and implementation of procedures necessary to guide the effort.		

APPENDIX 1B PLANT SHIELDING TO PROVIDE ACCESS TO AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION [II.B.2]

1B.1 INTRODUCTION

GE Hitachi Nuclear Energy (GEH) has performed a review of the ESBWR post-accident environment in response to Item II.B.2 of NUREG-0737 (Reference 1B-1). This attachment discusses the results of that review.

1B.2 SUMMARY OF SHIELDING DESIGN REVIEW

Several alternatives are potentially available to the designer to assure continued equipment availability and performance under post-accident conditions. One alternative is to provide redundant systems and/or components that are qualified to operate in the expected environment and thus preclude the need for operator access. Another is to provide operator access to conduct the operations and to maintain the equipment. This latter alternative would generally be accompanied by appropriate shielding and, in many cases, would be difficult if not impossible to carry out.

GEH has taken the first approach and furthermore has designed the plant so that most responses to transient conditions are automatic, including achieving and maintaining safe-shutdown conditions. The design basis for the ESBWR is to require safety-related equipment to be appropriately environmentally qualified and capable of being operated from the control room. As a result of this design philosophy and as shown by this review, no changes are necessary to assure that personnel access is adequate or that safety equipment is not degraded because of post-accident operation.

As part of the design of the ESBWR, it was necessary to establish the environmental conditions for qualification of safety-related equipment. A result of this design work was an environmental requirement establishing the integrated dose that the equipment must be able to withstand. These values are listed in Appendix 3H.

Another aspect of the review was the manner in which the safety-related equipment is arranged and operated during normal and abnormal operation and postulated accidents. The essence of the ESBWR is to achieve and maintain a safe shutdown condition for all postulated accident conditions with operator actions being conducted from outside the containment zones, principally from the control room.

The purpose of this review is first to verify that, where equipment access is required, it is reasonably accessible outside the containment zones. Secondly, the review verifys that inaccessible equipment is environmentally qualified and can be operated from the control room.

The results of the review are:

(1) The period of interest begins with the plant in a safe shutdown condition. Thus, the various safety-related systems needed to achieve safe shutdown conditions have performed as expected, and only the safety-related systems and auxiliaries, as described later, are required to maintain this condition.

(2) Based upon the accident source terms of Regulatory Guides 1.183 and 1.7 and Standard Review Plan Section 15.0.1 (References 1B-2, 1B-3 and 1B-4, respectively) including normal operations, exposures for equipment requiring post-accident access are enveloped based upon the Table below:

Area	Gamma (Gy)	Beta (Gy)
Containment	$2x10^6$	$2x10^7$

Each actual area is environmentally qualified to the area specific envelope as defined in Appendix 3H.

(3) It is not necessary for operating personnel to have access to any place other than the control room, technical support center, electrical equipment rooms for divisions 1, 2, 3 and 4, the Standby Liquid Control (SLC) system pump room, nonsafety-related Distributed Control and Information System (DCIS) rooms, remote shutdown panels, health physics facility (counting room), diesel generator control rooms, RWCU/SDC cross-tie line spectacle flanges and manual isolation valves, and Isolation Condenser/Passive Containment Cooling System (IC/PCCS) and fuel pool refill valves as described in Appendix 19A (19A.3, Criterion B) to operate the equipment of interest during the 100-day environmental qualification period required by Appendix 3H. The control room and the technical support center are designed to be physically accessible post-accident. The ability to occupy the technical support center post-accident is subject to the presence of acceptable radiation levels at those locations.

(4) Access to radwaste control panels or equipment is not required, but the Radwaste Building (RW) is accessible, since containment sump discharges are isolated. Thus, fission products are not transported to the Radwaste Building. The ESBWR does not have a containment isolation reset control area. These functions are provided in the control room.

(5) Following an accident, access is available to electrical equipment rooms containing motor control centers and to corridors in the upper Reactor Building (RB) (see post-accident radiation zone maps in Subsection 12.3.6). This is based on radiation shine from the containment. While not necessary to maintain safe shutdown, such access is useful in extending system functionality and facilitating plant recovery.

(6) The safety-related power supplies identified in Table 1B-5 are accessible. However, access is not necessary. Nonsafety-related diesel generators are also available and accessible to provide power.

1B.3 CONTAINMENT DESCRIPTION AND POST-ACCIDENT OPERATIONS

1B.3.1 Description of Containment

The ESBWR design includes many features to assure that personnel occupancy is not unduly limited and that safety-related equipment is not degraded by post-accident radiation fields. These features are detailed in other DCD locations. Consequently, only a brief summary description and a reference to other DCD locations are provided here for emphasis.

The configuration of the pressure suppression containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the containment atmosphere following an accident is thereby substantially reduced compared to the Reg. Guide 1.183 source terms by this scrubbing process. The Passive Containment Cooling System (PCCS) condensing function contributes to reduce many of the airborne fission products.

Containment leakage is limited to less than 0.35% of the weight in the containment free volume per day.

1B.3.2 Post-Accident Access of Areas and Systems

This section addresses any area that may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas that must be accessible after an accident are the control room and technical support center.

Areas requiring post-accident access also include consideration (in accordance with NUREG-0737, II.B.2) of the containment isolation reset control area, manual ECCS alignment area, motor control centers and radwaste control panels. However, the ESBWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room or are not applicable for the passive ECC systems. Areas requiring post-accident access that are normally areas of mild environment allowing unlimited access are not reviewed for access.

Systems specific to the ESBWR that may require post-accident access are those for long-term core cooling, fission product control and combustible gas monitoring, as well as the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1B.3.3 Post-Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10 CFR 52.47(a)(2)(iv) limits.

Safety-related systems are required for scram and to achieve a safe shutdown condition. However, they are not necessarily needed to maintain safe shutdown. The systems identified in Section 1B.5 are the systems used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the manual external connections for the IC/PCCS and fuel pools makeup.

1B.4 DESIGN REVIEW BASES

1B.4.1 Radioactive Source Term and Dose Rates

The radioactive source term used is equivalent to the source terms recommended in Reg. Guide 1.183 and Standard Review Plan 15.0.1 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <0.15 mSv/hr (15 mrem/hr).

Design dose rates for personnel in areas requiring post-accident access are such that the guidelines of General Design Criterion (GDC) 19 [i.e., <0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2] are not exceeded for the duration of the accident, based upon expected occupancy and protection.

1B.4.2 Accidents Used as the Basis for the Specified Radioactivity Release

The various accidents and associated potential for fuel rod failure are addressed in Sections 15.3 and 15.4. This chapter also provides the accident parameters. Of those accidents, only the design basis accident (DBA) LOCA is assumed to produce 100% failed fuel rods under NRC worst-case assumptions. The fuel handling accident is the only other DBA postulated as leading to failed fuel rods with the potential consequence of radioactivity releases comparable to the 10 CFR 52.47(a)(2)(iv) limits.

For the fuel handling accident, the reactor is either shutdown and cooled or is operating normally if the accident is in the spent fuel storage pool. The total activity released to the environment and the calculated exposures are provided in Subsection 15.4.1. The exposures are within the limits of 10 CFR 52.47(a)(2)(iv). Thus, recovery is possible well within the specified 100-day equipment qualification period defined in Appendix 3H. ECCS equipment is not affected by the fuel handling accident. This accident is not considered further.

Although a DBA-LOCA would not actually uncover the core or lead to fuel damage (see Section 6.3), core wide fuel failure is assumed such that this accident produces the limiting conditions of interest for this design review. In this accident the reactor is depressurized and reactor water mixes with suppression pool water in the process of keeping the fuel covered and cooled.

1B.4.3 Availability of Offsite Power

The availability of offsite power is not influenced by plant accident conditions. Various 10 CFR 50 Appendix A General Design Criteria (i.e., GDCs 33, 34, 35, 38, 41 and 44) require that safety functions be capable of being performed from either onsite or offsite power sources while the other power source is assumed unavailable. Consequently, loss of offsite power is typically assumed as occurring coincident with the beginning of many accident sequences.

However, continued absence of offsite power for the accident duration is not realistic. While restoration of offsite power is not a necessary condition for maintaining core cooling, its availability permits operation of other plant systems that would not otherwise be permitted by emergency power restrictions (e.g., operation of the pneumatic air system, nonsafety-related HVAC systems and other systems useful to plant cleanup and recovery).

Based on the PRA in Chapter 19, the probability for offsite power recovery is estimated to be very high in 8 hours, while the AC power is not needed for at least 72 hours following an accident, and even longer with operator action for IC/PCCS and fuel pools makeup after 72 hours.

1B.4.4 Radiation Qualification Conditions

The safety-related equipment requiring review for qualification is only that necessary for post-accident operations and for providing information for assuring post-accident control.

In 10 CFR 50.46, the long-term cooling capability is given as follows: "...decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core." A 100-day period has been selected as a sufficient extended period permitting site and facility response to terminate the event.

As part of the design review process, a set of reference conditions is necessary for comparing expected post-accident radiation exposures. Appendix 3H defines the environmental conditions for safety-related equipment zones for periods of 60 years normal operations, including anticipated tests and abnormal events, and 100 days following the DBA-LOCA. These conditions are upper bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. In effect, these are specification values, and equipment is qualified to meet or exceed these values.

Radiation sources in the containment are the same as the Table 1B-1 design basis values for water sources. For airborne radiation sources, the plant design basis of Table 1B-1 for air is used. Containment leakage is assumed to occur in each of the individual Reactor Building compartments. As previously noted, no credit has been taken for the radio-halogen scrubbing, which is an inherent feature of the BWR.

1B.5 RESULTS OF THE REVIEW

1B.5.1 Systems Required Post-Accident

This section establishes the various equipment required to function following an accident along with their locations. The expected habitability conditions and access and control needs are identified for the required post-accident period.

1B.5.1.1 Necessary Post-Accident Functions and Systems

Following an accident and assuming that immediate plant recovery is not possible, the following functions defined in Reference 1B-5 are necessary:

(1) Reactivity control;

(2) Reactor core cooling;

(3) Reactor coolant pressure boundary integrity (if not already breached by the initiating event);

(4) Containment integrity; and

(5) Radioactive effluent control.

Reactivity control is a short-term function and is achieved when the reactor is shutdown. The remaining functions are achieved in the longer-term post-accident period by use of:

(1) The Emergency Core Cooling System (ECCS) (for reactor core cooling);

(2) The Passive Containment Cooling System (PCCS) (for containment heat removal);

(3) The fission product removal and control system and auxiliaries (for radioactive effluent control, see Subsection 1B.5.1.4); and

(4) Instrumentation and controls and power for accident monitoring and functioning of the necessary systems and associated habitability systems.

Tables 1B-2 through 1B-5 are generated to show:

- What major equipment and systems are required to function and thereby define the systems for review; and
- The redundant equipment locations by divisional isolated room or area and containment or building.

1B.5.1.2 Emergency Core Cooling and Residual Heat Removal Systems

The Gravity-Driven Cooling System (GDCS) provides cooling for the fuel under accident conditions as described in Subsection 6.3.2.7. After it is initiated the GDCS requires no auxiliary support or post-accident instrumentation.

The Automatic Depressurization System (ADS) function is described in Subsection 6.3.2.8. A postulated small break accident could require the depressurization function until the GDCS is initiated. In the case of a small break accident, the majority of the fission products would be released via the safety/relief valves to the suppression pool and hence to the containment, rather than direct mixing through the suppression pool vents, as would occur following a DBA-LOCA. In either case, the distribution of fission products is assumed the same as for the DBA-LOCA even though, realistically, a significant portion of halogens and solid fission products would be retained in the reactor pressure vessel. Thus, the results as they apply to the ADS are conservative. The pneumatic nitrogen supply for the ADS is supplied by the SRV accumulators included in Table 1B-2. The hand-operated nitrogen reserve supply valves are accessible outside the containment, if needed, to mitigate a large nitrogen leak.

Credit is also taken during a LOCA for the water volume stored in the discharge lines of the Isolation Condenser System (ICS) and for operation of the Standby Liquid Control (SLC) system.

Containment cooling is provided by the PCCS. PCCS is a passive system that requires no operator action. The PCCS function cools the air volumes and discharges the resulting condensate to the GDCS pools so that it is available for return to the reactor pressure vessel. Additional containment cooling can be obtained by manually initiating the nonsafety-related Fuel and Auxiliary Pools Cooling System (FAPCS) in its drywell spray mode. Controls for initiating drywell spray are available in the main control room.

The fuel pool cooling function is also provided in the event that a recently unloaded fuel batch requires continued cooling during the post-accident period. The spent fuel pool contains sufficient inventory to ensure no operator action is required during the first 72 hours. After that period, either makeup water must be supplied to the spent fuel pool or the FAPCS must be initiated. The FAPCS equipment is environmentally qualified, so access is not required and redundancy is included in system components.

The locations of selected ECCS equipment and instrument transmitters are included in Table 1B-2. These listings do not represent all the types of this equipment that are environmentally qualified, safety-related, or included in the systems classified in Table 3.2-1. It does, however, represent principal components that are needed to operate, generally during post-accident operations. For example, (after depressurization) only the GDCS discharge valves need to open to direct water to the reactor. Similarly, the instrument transmitters shown are those that would provide information for system initiation and monitoring of long-term system performance post-accident. Control room instrumentation is not listed because it is in an accessible area where radiation degradation would not be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under the Nuclear Boiler System (B21) are those for ECCS functions or monitoring reactor vessel level.

1B.5.1.3 Flammability Control

Flammability control in the containment is achieved by an inert atmosphere during all plant operating modes except operator access for refueling and maintenance. The Containment Inerting System gas supply is described within Subsection 6.2.5.2. The Containment Monitoring System (CMS) measures and records oxygen and hydrogen concentrations in the containment under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.5.2). Table 1B-3 lists the principal combustible gas monitoring components and their locations. Long-term flammability control following design basis accidents is provided by the Passive Autocatalytic Recombiner System.

1B.5.1.4 Fission Product Removal and Control System

The ESBWR has a main control room filter system that performs a safety-related function following a design basis accident. The main control room is provided with emergency filter units to maintain a safe control room atmosphere following a design basis accident as discussed in Section 6.4.

The CMS described in the previous section also measures and records containment area radiation under post-accident conditions. Table 1B-4 lists the fission product removal control components and locations.

1B.5.1.5 Instrumentation and Control, Power and Habitability Systems

Post-accident instrumentation and control system equipment is listed with the applicable equipment in Tables 1B-2, 1B-3 and 1B-4. Additional instrumentation and control equipment is included with the power and habitability systems equipment listed in Table 1B-5. Instrumentation requirements for post-accident monitoring (PAM) are discussed in Subsection 7.5.1.

The ESBWR does not need, and therefore does not have, any safety-related standby diesel generators. Storage batteries are the standby power source for safety-related electric power.

Habitability systems ensure that the operator can remain in the control room and take appropriate action for post-accident operations. The Control Building includes the instrumentation and controls necessary for operating the systems required under post-accident conditions.

1B.6 REFERENCES

1B-1 U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," U. S. NRC Report NUREG-0737.

1B-2 U. S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183.

1B-3 U. S. Nuclear Regulatory Commission, "Control of Combustible Gas Concentrations in Containment," Regulatory Guide 1.7.

1B-4 U. S. Nuclear Regulatory Commission, "Radiological Consequence Analyses Using Alternate Source Terms," U. S. NRC Report NUREG-0800, Standard Review Plan (SRP) Section 15.0.1.

1B-5 American National Standards Institute / American Nuclear Society, "Criteria for Accident Monitoring Functions in Light Water Reactors," ANSI/ANS 4.5.

Table 1B-1

Radiation Source Comparison

Activity Group	% Core Inventory Released		
	Reg. Guide 1.183	Reg. Guide 1.7	Plant Design Basis
Air			
Noble Gases	100	100	100 *
Halogens	30	—	30 *
All Remaining	—	—	—
Water			
Noble Gases	0	—	0
Halogens	—	50	50 **
All Remaining	—	1	1

* Uniformly mixed within the containment boundary

** Uniformly mixed in the suppression pool and reactor coolant

Table 1B-2

Post-Accident Emergency Core Cooling Systems and Auxiliaries

Equipment	MPL	Location
Automatic Depressurization System (ADS)		
Safety Relief Valves	B21	Upper Drywell (C)
SRV Accumulators	B21	Upper Drywell (C)
Depressurization Valves	B21	Upper Drywell (C)
Gravity-Driven Cooling System (GDCS)		
Discharge Valves	E50	Upper Drywell (C)
Isolation Condensers		
Steam Supply Valves	B32	Upper Drywell (C)
Condensate Discharge Valves	B32	Upper Drywell (C)
Condenser Units	B32	IC/PCCS Pools (RB)
Standby Liquid Control (SLC) system		
Inboard Isolation Valves	C41	Upper Drywell (C)
Outboard Isolation Valves	C41	SLC Rooms (RB)
Accumulators	C41	SLC Rooms (RB)
Initiation / Monitoring Instrumentation		
Reactor Water Level	B21	Instrument Rack Rooms (RB)
Reactor Pressure	B21	Instrument Rack Rooms (RB)
Drywell Pressure	T62	Instrument Rack Rooms (RB)
Wetwell Pressure	T62	Instrument Rack Rooms (RB)

MPL — Master Parts List Number designated for the system

(C) — Containment

(RB) — Reactor Building

Table 1B-3

Post-Accident Containment Systems and Auxiliaries

Equipment	MPL	Location
(Deleted)		
Containment Monitoring System (CMS)		
Hydrogen, Oxygen Elements	T62	CMS Rooms (RB)
Gas Measurement	T62	CMS Rooms (RB)
Gas Elements	T62	CMS Rooms (RB)
Drywell Gas Valve	T62	CMS Rooms (RB)
Wetwell Gas Valve	T62	CMS Rooms (RB)
Gas Supply	T62	CMS Rooms (RB)
Passive Containment Cooling System (PCCS)		
Condenser Units	T15	IC/PCCS Pools (RB)
Condensate Discharge	T15	GDCS Pools (C)
Non-condensable Gas Discharge	T15	Suppression Pool Wetwell (C)
Passive Autocatalytic Recombiner System (PARS)		
Individual PARS units	T49	Drywell and Suppression Pool Wetwell (C)

MPL — Master Parts List Number designated for the system

(C) — Containment

(RB) — Reactor Building

Table 1B-4

Post-Accident Fission Product Removal and Control Systems and Auxiliaries

Equipment	MPL	Location
Control Room Habitability Area HVAC Subsystem		
	U77	(CB)
Stacks		
Radiation (Ion/Scint.)	D11	RB/FB Stack
Radiation (Ion/Scint.)	D11	TB Stack
Radiation (Ion/Scint.)	D11	RW Stack

MPL — Master Parts List Number designated for the system

(CB) — Control Building

RB/FB — Reactor Building / Fuel Building

TB — Turbine Building

RW — Radwaste Building

Table 1B-5

Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries

Equipment	MPL	Location
Instrumentation and Controls		
Post-Accident I&C	H11-Post-Accident	Control & Panel Rooms (CB)
Safety-Related Distributed Control and Information System	C63	(CB and RB)
Power		
DC Supply	R16-Storage Batteries	(RB)
AC Low Voltage and I&C Supply Systems	R13 Post-Accident	(RB)
Control Building HVAC		
Detection of high airborne radioactivity	D21	(CB)
Detection of smoke	U43	(CB)
Isolation of Control Room Habitability Area	U77	(CB)
Control Room Habitability Area HVAC Subsystem controls	U77	(CB)

MPL — Master Parts List Number designated for the system

(RB) — Reactor Building

(CB) — Control Building

APPENDIX 1C INDUSTRY OPERATING EXPERIENCE

1C.1 EVALUATION

Industry operating experience information is routinely made available to or distributed by GEH design and modifications personnel. The more important industry-wide issues are routinely addressed in NRC Generic Letters and Bulletins.

All of the Generic Letters and Bulletins covering January 1, 1980 through February 24, 2005 were reviewed. Of those, the Generic Letters and Bulletins that are potentially applicable to the ESBWR design or operations are addressed in Tables 1C-1 and 1C-2, respectively. For each of those Generic Letters and Bulletins, Tables 1C-1 and 1C-2 provide the location(s) where the topic of the Generic Letter or Bulletin topic is addressed or a summary conclusion of its effect on the ESBWR. Generic Letter and Bulletin topics deemed not applicable to the ESBWR design or operations (topics pertaining to other reactor types or BWR design features, e.g., a Reactor Recirculation Pump issue) are generally not included in Tables 1C-1 and 1C-2. Also, Generic Letter and Bulletin topics related to identified regulatory or industry developed resolutions are not included in Tables 1C-1 and 1C-2, to avoid repetition.

Selected Generic Letters and Bulletins are noted as being within the scope of the COL applicant as indicated below in Section 1C.2 and in Tables 1C-1 and 1C-2.

1C.2 COL INFORMATION

1C.1-1-A Handling of Safeguards Information

COL Applicant will address requirements of Generic Letter 82-39 regarding the handling of safeguards information. (Table 1C-1, No. 82-39)

1C.1-2-A Emergency Preparedness and Response Actions

COL Applicant will address requirements of IE Bulletin 2005-02 regarding emergency preparedness and response actions for security-based events. (Table 1C-2, No. 2005-02)

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
80-30	4/10/80	Clarification of the Term "Operable" as it Applies to Single Failure Criterion for Safety Systems Required by TS	Chapter 16
80-34	4/25/80	Clarification of NRC Requirements for Emergency Response Facilities at Each Site	The ESBWR includes provisions for a Technical Support Center (TSC). The Operational Support Center (OSC) and Emergency Operating Facility (EOF) are described in Section 13.3.
80-110	12/15/80	Periodic Updating of Final Safety Analysis Reports (FSARs)	Not Applicable. Administrative communication.
80-113	12/22/80	Control of Heavy Loads	Subsection 9.1.5. See also Section 1.11 (Item A-36).
81-03	2/26/81	Implementation of NUREG-0313, Rev. 1	Subsections 5.2.3.4.1 and 5.3.1.4
81-04	2/25/81	Emergency Procedures and Training for Station Blackout Events	The ESBWR does not require emergency AC power to achieve safe shutdown. Therefore, this issue is not applicable to the ESBWR Standard Plant design. See Section 1.11 (Item A-44) and Subsection 15.5.5.
81-07	2/3/81	Control of Heavy Loads	Subsection 9.1.5. See also Section 1.11 (Item A-36).
81-10	2/18/81	Post-TMI Requirements for the Emergency Operations Facility	See response to GL 80-34.
81-11	2/29/81	Comments on NUREG-0619	Not applicable. The ESBWR does not include a CRD return nozzle on the RPV.
81-20	4/10/81	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Not applicable. The ESBWR utilizes Fine Motion Control Rod Drives (FMCRDs) and does not have CRD withdraw lines and scram discharge volume.
81-37	12/29/81	ODYN Code Reanalysis Requirements	Not applicable, ESBWR does not use the ODYN Code

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
81-38	11/10/81	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	The Radwaste Building includes space for processing and storage of low level waste. Storage space is provided for 6 months worth of waste. Section 11.4
82-09	4/20/82	Environmental Qualification of Safety-Related Electrical Equipment	Section 3.11
82-21	10/6/82	Technical Specifications for Fire Protection Audits	Not applicable. No longer controlled by Technical Specifications.
82-23	10/30/82	Inconsistency Between Requirements of 10CFR73.40(d) and Standard Technical Specifications for Performing Audits of Safeguard Contingency Plans	Not applicable. No longer controlled by Technical Specifications.
82-27	11/15/82	Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."	The suppression pool is provided with sufficient temperature instrumentation to monitor the temperature rise during testing that adds heat to the pool. The Tech Spec limits on pool temperature are established in accordance with the applicable design limits. Subsections 5.2.2.1, 7.2.3 and Chapter 16 TS Bases 3.3.1.1
82-33	12/17/82	Supplement 1 to NUREG-0737 – Requirements for Emergency Response Capability	These requirements have been incorporated into Reg. Guide 1.97 and the ESBWR conforms with this Reg. Guide. Table 1.9-21 and Appendix 1A
82-39	12/22/82	Problems with the Submittals of 10 CFR 73.21 Safeguards Information for Licensing Review	Not Applicable. Administrative communication. GEH has an approved procedure for handling Safeguards Information including how to mail such information to authorized recipients. COL Applicant will address requirements of Generic Letter 82-39 regarding the handling of safeguards information. (COL 1C.1-1-A)

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
83-05	2/8/83	Safety Evaluation of "Emergency Procedure Guidelines" Revision 2," NEDO-24934, June 1982	See Section 18.9 for a discussion of Emergency Operating Procedures development. Revision 2 of the Emergency Procedure Guidelines has been superseded by newer revisions.
83-07	2/16/83	The Nuclear Waste Policy Act of 1982	The requirements of this generic letter have been incorporated into CFR or other SRP requirements.
83-13	3/2/83	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems	Subsection 9.4.1
83-28	7/8/83	Required Actions Based on Generic Implications of Salem ATWS Events	Superseded by 10 CFR 50.62, see Section 15.5.4 for the ATWS event evaluation
83-33	10/19/83	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	Subsection 9.5.1 and Appendices 9A and 9B
84-15	7/2/84	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	Not applicable, the ESBWR does not have/need emergency diesel generators.
84-23	10/26/84	Reactor Vessel Water Level Instrumentation in BWRs	Subsection 7.7.1
85-01	1/9/85	Fire Protection Policy Steering Committee Report	Subsection 9.5.1 and Appendices 9A and 9B
85-06	1/16/85	Quality Assurance Guidance for ATWS Equipment that is not Safety-Related	Table 3.2-1, Subsection 17.1.22
86-10	4/24/86	Implementation of Fire Protection Requirements	Subsection 9.5.1 and Appendices 9A and 9B
86-10, Supp 1	3/25/1994	Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area	Subsection 9.5.1 and Appendices 9A and 9B
87-06	3/13/87	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	Not applicable. As described in Appendix 3K, the ESBWR does not need or have pressure isolation valves between the reactor coolant pressure boundary and a low pressure piping system.

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
87-09	6/4/87	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operations and Surveillance Requirements	Chapter 16 TS Section 3.0, consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.1).
88-01	1/25/88	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	Subsection 5.2.3.4 and Chapter 16B Bases B 3.4.2
88-14	8/8/88	Instrument Air Supply System Problems Affecting Safety-Related Equipment Past Related Correspondence: IE Notice 87-28, Supp. 1 NUREG-1275, Volume 2	Subsection 9.3.6
88-15	9/12/88	Electric Power Systems — Inadequate Control Over Design Process Past Related Correspondence: IE Notice 88-45	Most of the issues described in GL 88-15 are not applicable to the ESBWR. Section 8.3
88-16	10/3/88	Removal of Cycle-Specific Parameter Limits from Technical Specifications	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.1). Chapter 16 TS Subsection 5.6.3
88-18	10/20/88	Plant Record Storage on Optical Disks Past Related Correspondence: NUREG-0800 Chapter 17; Reg. Guide 1.28, Rev. 3	This generic letter is not involved with operational experience, so no evaluation is required. Section 17.2
88-20	11/23/88	Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)	Chapter 19
88-20, Supp 1	8/29/89	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR Para. 50.54(f)	Chapter 19
88-20, Supp 2	4/4/90	Accident Management Strategies for Consideration in the Individual Plant Examination Process	Chapter 19

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
88-20, Supp 3	7/6/90	Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities	Chapter 19
88-20, Supp 4	6/28/91	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)	Chapter 19
88-20, Supp 5	9/8/95	Individual Plant Examination of External Events for Severe Accident Vulnerabilities	Chapter 19
89-01	1/31/89	Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.1). Chapter 16 TS Subsections 5.5.1 and 5.5.3
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products Past Related Correspondence: EPRI-NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications." Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10, Information Notices 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	Addressed as part of Quality Assurance Program for Operations. See Section 17.2
89-04	4/3/89	Guidance on Developing Acceptable Inservice Testing Program	GL 89-04 only applies to pumps and valves. See Subsection 3.9.6
89-04, Supp 1	4/4/95	Guidance on Developing Acceptable Inservice Testing Programs	Subsection 3.9.6
89-06	4/12/89	Task Action Plan Item I.D.2 – Safety Parameter Display System 10 CFR 50.34(f)	Appendix 1A and Chapter 18

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	Section 13.6
89-07 Supp 1	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	Section 13.6
89-08	5/2/89	Erosion/Corrosion-Induced Pipe Wall Thinning	Subsections 6.6.7 and 10.3.6
89-10	6/28/89	Safety-Related Motor-Operated Valve Testing and Surveillance	Subsections 3.9.6 and 3.9.6.1
89-10, Supp 1	6/13/90	Results of Public Workshop	Subsections 3.9.6 and 3.9.6.1
89-10, Supp 3	10/25/90	Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves	Subsections 3.9.6 and 3.9.6.1
89-10, Supp 4	2/12/92	Consideration of Valve Mispositioning in Boiling Water Reactors	Subsections 3.9.6 and 3.9.6.1
89-10, Supp 5	6/28/93	Inaccuracy of Motor-Operated Valve Diagnostic Equipment	Subsections 3.9.6 and 3.9.6.1
89-10, Supp 6	3/8/94	Information on Schedule and Grouping, and Staff Responses to Additional Public Questions	Subsections 3.9.6 and 3.9.6.1
89-13	7/18/89	Service Water System Problems Affecting Safety-Related Equipment	Not applicable. ESBWR has no safety-related service water and applies water quality standards to the use of water for safety functions. As indicated in Subsection 9.2.5, long-term makeup water for the IC/PCCS and spent fuel pools is required to meet fire protection system water quality standards.
89-13, Supp 1	4/4/90	Service Water System Problems Affecting Safety-Related Equipment	Not applicable. ESBWR has no safety-related service water and makes no use of untreated water for safety functions. See above entry for GL 89-13.

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
89-14	8/21/89	Line Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.1) Chapter 16 TS Section 3.0
89-15	8/21/89	Emergency Response Data System	Section 13.3
89-16	9/1/89	Installation of a Hardened Wetwell Vent	The ESBWR does not need a dedicated Hardened Wetwell Vent, however the Containment Inerting System is supplemented by providing a manual, hardened, and elevated vent path as discussed in Subsection 6.2.5.4.
89-18	9/6/89	Resolution of Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants"	Section 1.11 and 19A.6
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47, "Safety Implication of Control Systems in LWR Nuclear Power Plants", Pursuant to 10 CFR 50.54(f)	Section 1.11 (Item A-47), Subsections 5.4.5 and 7.7.3
89-22	10/19/89	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed By The National Weather Service	Table 2.0-1 and Table 3G.1-2
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	Subsection 3.9.3.7.1, Item (3)b
91-03	03/06/91	Reporting of Safeguards Events	Section 13.6
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	Chapter 16 TS Sections 3.1-3.9 are already written to support 24-month fuel cycles.
91-05	04/09/91	Licensee Commercial-Grade Procurement and Dedication Programs	Addressed as part of Quality Assurance Program for Operations. See Section 17.2

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
91-06	04/29/91	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies," Pursuant to 10 CFR 50.54(f)	Subsection 8.3.2 and Section 1.11 (Issues A-30 and 128). Operating and maintenance procedures developed in accordance with Subsection 13.5.2 and Section 18.9 ensure that the required tests are adequately addressed.
91-10	07/08/91	Explosive Searches at Protected Area Portals	Issue is Not Applicable to Commercial Nuclear Power Plants.
91-11	07/18/91	Resolution of Generic Issues 48, "LCOs for Class 1E Vital Instrument Buses," and 49, "Interlocks and LCOs for Class 1E Tie Breakers" Pursuant to 10 CFR 50.54(f)	Section 1.11 (Issues 48, 49 and 128)
91-14	09/23/91	Emergency Telecommunications	Subsection 9.5.2.2
91-15	09/23/91	Operating Experience Feedback, Solenoid-Operated Valve Problems at U.S. Reactors	Subsection 3.9.6.1
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	Section 13.6
91-17	10/17/91	Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"	Refer to Subsection 3.9.3, American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 Components, Component Supports and Core Support Structures, for further details. Addressed as part of the design, material selection, procurement, fabrication and maintenance processes for bolted connections. Section 1.11 (Issue 29)
92-01r1	03/06/92	Reactor Vessel Structural Integrity	Subsection 5.3.2 and 5.3.3
92-04	8/19/92	Resolution of the Issues Related to Reactor Vessel Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)	The ESBWR includes a continuous purge of water to the reference leg to prevent the buildup of non-condensable gases. The CRD Hydraulics provides this flow. Subsections 4.6.1.2.4 and 7.7.1.2

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
92-08	12/17/92	Thermo-Lag 330-1 Fire Barriers Past Related Correspondence: IE Notices 92-01 and 92-01 Supplement 1.	This GL only applies to Thermo-Lag 330-1 fire barrier systems. The design intent for ESBWR is to provide strict physical separation between the redundant safety-related divisions. If it is determined that a fire-wrapping material is required in some locations, an alternative material qualified per the guidance of RG 1.189 will be used. Subsection 9.5.1, Section 17.2
93-05	9/27/93	Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation	Not Applicable. Administrative communication. Lessons from the Tech Spec Improvement programs have been factored into the proposed ESBWR Tech Specs. Chapter 16
93-06	10/25/93	Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas"	The ESBWR only uses highly combustible gases in any safety-related area for reference gas in the H_2/O_2 monitors. This calibration gas is only used periodically and normally valved out of service. The H_2 bottles are located in a nonsafety-related structure. The lines to the H_2 monitors are very small and would limit the flow in the event of a break. Subsection 9.5.1 and Section 1.11 (Issue 106)
93-08	12/29/93	Relocation of Technical Specification Tables Of Instrument Response Time Limits	Not Applicable. Administrative communication.
94-01	5/31/94	Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators	Not Applicable. Maintenance communication. The ESBWR does not have safety-related emergency diesel generators. There are no TS surveillance requirements for the nonsafety-related diesel generators.
94-02	7/11/94	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in BWRs	The ESBWR addresses the concerns of Thermal-Hydraulic Instability. Section 4.3 and Appendix 4D

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
94-03	7/25/94	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors	Controls on material properties and welding parameters are placed on all stainless material used in the RPV including the shroud. Subsection 5.2.3.4.1
95-07	8/17/95	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	The number of safety-related valves in the ESBWR is much smaller than previous designs. The number safety-related valves that need to open to perform their function is even smaller. Many of the safety-related valves that need to open are squib actuated and not subject to this phenomenon. Globe valves are generally used in the other applications. In any case, GL 89-10 Supplement 6 now covers this issue and the ESBWR complies with the guidance of this document. Subsection 3.9.6
96-01	1/10/96	Testing of Safety-Related Logic Circuits	Plant procedures ensure that the safety-related logic circuitry is adequately covered in the surveillance procedures as described in Generic Letter 96-01. Chapter 16
96-03	1/31/96	Relocation of The Pressure Temperature Limit Curves And Low Temperature Overpressure Protection System Limits	Consistent with current Standard Technical Specifications (NUREG-1434, Rev. 3.1) Chapter 16 TS Subsection 5.6.4
96-04	6/26/96	Boraflex Degradation in Spent Fuel Pool Storage Racks	Not Applicable. Procurement communication. The equipment specification for the racks at the time of the order will be consistent with the latest regulatory guidance. Subsection 9.1.2
96-05	9/18/96	Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves	Subsection 3.9.6.1

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
96-06	9/30/96	Assurance of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions	PCCS provides containment air cooling during design basis accidents as described in Subsections 6.2.1 and 6.2.2, and is not subject to water hammer effects. The Chilled Water System provides cooling water to the Drywell Cooling System during normal operation, and is isolated on a LOCA signal as discussed in Subsections 9.2.7.5 and 6.2.4.3.2.1. Fluid-filled piping associated with containment penetrations that automatically isolate during DBAs is designed in accordance with ASME Code Section III to accommodate thermal transient loadings as described in Subsection 3.9.3.4 and Table 3.9-2.
96-06, Supp 1	11/13/97	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	Subsections 6.2.1, 6.2.2 and 6.2.4.3.2.1 and 9.2.7.5
97-04	10/7/97	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	Not applicable, the ESBWR does not use pumps for ECCS or safety-related containment cooling functions.
98-01	5/11/98	Year 2000 Readiness of Computer Systems at Nuclear Power Plants	Outdated concern
98-01, Supp 1	1/14/99	Year 2000 Readiness of Computer Systems at Nuclear Power Plants	Outdated concern
98-04	7/14/98	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	Applicable only to the suppression pool as described in Subsection 9.1.3. The GDCS pools do not have the debris transport mechanisms that the Suppression Pool is subject to. The PCCS pools are not subject to LOCA debris. There is no safety-related containment spray.
99-02	6/3/99	Laboratory Testing of Nuclear-Grade Activated Charcoal	Chapter 16 TS Subsection 5.5.13.c

Table 1C-1

Operating Experience Review Results Summary – Generic Letters

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
03-01	6/12/03	Control Room Habitability	The verification requirements are in accordance with the applicable regulatory guidance and standards. See Section 6.4.
06-02	2/1/2006	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	Subsection 8.2.2.1 and ninth bullet in Subsection 8.2.3
06-03	4/10/2006	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	Subsection 9.5.1 and Appendices 9A and 9B

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
79-02r2	11/8/79	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	Subsection 3.9.3.7
79-08	4/14/79	Events Relevant to Boiling WaterReactors Identified During Three Mile Island Incident	Appendix 1A
80-01	1/11/80	Operability of ADS Valve Pneumatic Supply	The design of the pneumatic supply to the ADS valves addresses the concerns with the potential loss of loss of pneumatic pressure. In addition, the ESBWR ADS design includes a diverse type of depressurization valves (DPVs) that do not rely on pneumatic supply.
80-03	2/6/80	Loss of Charcoal from Standard Type II, 2 Inch, Tray Absorber Cells	Subsection 9.4.1
80-05	3/10/80	Vacuum Condition Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks	Not applicable to BWRs
80-06	3/13/80	ESF Reset Controls	Subsections 6.2.4.1 and 7.1.6.6.1.3
80-08	4/7/80	Examination of Containment Liner Penetration Welds	Containment penetrations are designed to ASME Code Section III and Section XI requirements for design and accessibility of welds for in-service inspection to meet 10 CFR 50 Appendix A, General Design Criterion 16 (see Subsections 3.1.2.7 and 3.1.4.10). The application of ASME Code Section XI, to in-service examination of containment penetration welds is discussed in Subsection 3.6.2.1.1 and Section 6.6.
80-10	5/6/80	Contamination of Non-radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment	Subsections 9.2.1 and 9.2.2
80-12	5/9/80	Decay Heat Removal System Operability	Not applicable to BWRs

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
80-13	5/12/80	Cracking in Core Spray Spargers	Not Applicable. The ESBWR does not have a core spray sparger or any comparable RPV internal structures that rely on flow distribution across the core to perform their post-accident design function.
80-15	6/18/80	Possible Loss of Emergency Notification System with Loss of Offsite Power	Subsection 9.5.2.2
80-20	7/31/80	Failures of Westinghouse Type W-2 Spring Return to Neutral Control Switches	Not applicable to new equipment
80-21	11/6/80	Valve Yokes Supplied by Malcolm Foundary Company, Inc.	Not applicable to new equipment
80-22	9/11/80	Automatic Industries, Model 200-500-008 Sealed Source Containers	Not applicable to new equipment
80-24	11/21/80	Prevention of Damage due to Water Leakage Inside Containment	Not applicable. The ESBWR Containment is cooled using the Chilled Water System (CWS), which is a closed loop system. Potential leakage inside containment from other systems is monitored by the Equipment and Floor Drain System. Subsection 9.3.3.
80-25	12/19/80	Operating Problems with Target Rock SRVs at BWRs	Not applicable to the ESBWR design. Different valve type to be used. Subsection 5.4.13
81-01	1/27/81	Surveillance of Mechanical Snubbers	Subsection 3.9.3.7.1, Item (3)b
81-02	4/9/81	Failure of Gate Type Valves to Close Against Differential Pressure	Not applicable to BWRs
81-02, Supp 1	8/18/81	Failure of Gate Type Valves to Close Against Differential Pressure	Not applicable to BWRs

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
81-03	4/10/81	Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. (Asiatic clam) and Mytilus Sp. (Mussel)	Not applicable. Safety-related ESBWR systems do not rely on cooling water sources that can be blocked by these species.
82-04	12/3/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	Applies to a specific equipment supplier that is no longer selling primary containment electrical penetration assemblies. ESBWR primary containment electrical penetration assemblies are qualified to Institute of Electrical and Electronic Engineers (IEEE) 317 requirements in accordance with Regulatory Guide 1.63. See Subsection 3.11.1.1.
83-06	7/22/83	Non-Conforming Materials Supplied by Tube-Line Corp.	Not applicable, vendor supply issue
84-01	2/3/84	Cracks in Boiling Water Reactor Mark I Containment Vent Headers	Not applicable to the ESBWR containment design
84-03	8/24/84	Refueling Cavity Water Seal	The ESBWR utilizes permanently installed flexible bellows between the RPV and the refueling cavity. See Subsections 6.2.1.1.2, 9.1.4.21 and 12.4.4.
85-03	11/15/85	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	Subsection 3.9.6.1
85-03, Supp 1	4/27/88	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings Past Related Correspondence: IE Bulletin 85-03, IE Notice 86-29, and IE Notice 87-01	Subsection 3.9.6.1
86-01	5/23/86	Minimum Flow Logic Problems That Could Disable RHR Pumps	Not Applicable. The ESBWR does not have safety-related RHR pumps
86-03	10/8/86	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	Not Applicable. The ESBWR does not have ECCS pumps

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
87-01	7/9/87	Thinning of Pipe Walls in Nuclear Power Plants Past Related Correspondence: IE Notice 88-17	Subsection 6.6.7
87-02	11/6/87	Fastener Testing to Determine Conformance with Applicable Material Specifications	Addressed as part of Quality Assurance Program for Operations. See Section 17.2
87-02, Supp 1	4/22/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	Addressed as part of Quality Assurance Program for Operations. See Section 17.2
87-02, Supp 2	6/10/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	Addressed as part of Quality Assurance Program for Operations. See Section 17.2
88-04	5/5/88	Potential Safety-Related Pump Loss	Not applicable, the ESBWR does not have safety-related pumps
88-07	6/15/88	Power Oscillations in Boiling Water Reactors (BWRs) Past Related Correspondence: IE Notice 88-39	Sections 4.3 and 4D
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	Sections 4.3 and 4D
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	Not applicable, the vendor has corrected the problem and new transmitters have been changed to correct the problem.
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	Section 4B.1
92-01	6/24/92	Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage	Not Applicable. See evaluation result for Generic Letter 92-08 in Table 1C-1.

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
92-01, Supp 1	8/28/92	Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function	Not Applicable. See evaluation result for Generic Letter 92-08 in Table 1C-1.
93-02	5/11/93	Debris Plugging of Emergency Core Cooling Suction Strainers	Applicable only to the suppression pool as described in Subsection 9.1.3. The GDCS pools do not have the debris transport mechanisms that the Suppression Pool is subject to.
93-02, Supp 1	2/18/94	Debris Plugging of Emergency Core Cooling Suction Strainers	Applicable only to the suppression pool as described in Subsection 9.1.3. See above.
93-03	5/28/93	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs	The ESBWR includes a continuous purge of water to the reference leg to prevent the buildup of non-condensable gases. The CRD Hydraulics provides this flow. Subsections 4.6.1.2.4 and 7.7.1.2
94-01	4/14/94	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1	The FAPCS is designed to prevent the possibility of draining water from the Spent Fuel Storage Pool. Subsection 9.1.3
95-02	10/17/95	Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode	Applicable only to the suppression pool as described in Subsection 9.1.3.
96-02	4/11/96	Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment	Subsection 9.1.5
96-03	5/6/96	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	Applicable only to the suppression pool as described in Subsection 9.1.3. See response to IE Bulletin 93-02
96-04	7/5/96	Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks	Related to dry cask storage, which is not part of the ESBWR Standard Plant design.

Table 1C-2

Operating Experience Review Results Summary – IE Bulletins

No.	Issue Date	Title	Evaluation Result or Topic's Tier 2 Location(s)
2005-02	7/18/05	Emergency Preparedness and Response Actions for Security-Based Events	Site-specific, and thus not within the scope of a design certification. COL Applicant will address requirements of IE Bulletin 2005-02 regarding emergency preparedness and response actions for security-based events. (COL 1C.1-2-A)

APPENDIX 1D SUMMARY OF TIER 2* INFORMATION

1D.1 PLANT-SPECIFIC CHANGES TO CERTAIN DESIGNATED MATERIAL IN TIER 2

Certain information within sections of Tier 2 is designated as Tier 2* with brackets, italicized text, and an asterisk after the closing bracket. Table 1D-1 provides a summary of DCD locations that contain Tier 2* information. A plant-specific change to any of this Tier 2* designated information shall require NRC Staff approval prior to implementing the change. A request for departure from Tier 2* will be treated as a request for license amendment under 10 CFR 50.90 and 50.92.

1D.2 EXPIRATION OF TIER 2* INFORMATION

The requirement for prior NRC Staff approval of plant-specific changes will expire for some of the designated information, as indicated in Table 1D-1, when the plant first achieves 100% power.

Tier 2* material related to the following topics has no expiration date and may not be changed without prior NRC approval. A request for a departure will be treated as a request for a license amendment.

- Fuel mechanical and thermal-mechanical design evaluation reports, including fuel burnup limits (References 4.2-4 and 4.2-5).
- Control rod mechanical and nuclear design reports (References 4.2-8 and 4.2-9).
- Fuel nuclear design report (referenced in several locations in Chapters 4 and 15).
- Critical power correlation (Reference 4.4-12).
- Fuel licensing acceptance criteria (Appendix 4B).
- Control rod licensing acceptance criteria (Appendix 4C).
- Mechanical and structural design of spent fuel storage racks (Subsection 9.1.2.4 and Reference 9.1-2).

After the plant first achieves full power, Tier 2* material related to the following topics reverts to Tier 2 status and is thereafter subject to the same departure provisions that apply to Tier 2 material.

- ASME Boiler & Pressure Vessel Code, Section III.
- ACI 349 and ANSI/AISC–N690.
- Motor-operated valves.
- Equipment seismic qualification methods.
- Piping design acceptance criteria.
- Instrument Setpoint Methodology

- Safety-Related Distributed Control and Information System (Q-DCIS) performance specifications and architecture.
- Safety System Logic and Control (SSLC) hardware and software qualification.
- Human factors engineering design and implementation process.

Tier 2* material related to the following topics reverts to Tier 2 status after successful completion of testing for the first ESBWR unit, and is thereafter subject to the same departure provisions that apply to Tier 2 material.

- First of a kind testing for reactor stability (Subsection 14.2.8.2.7).
- Reactor Pre-Critical Heatup with RWCU/SDC (Subsection 14.2.8.2.35.1).
- Isolation Condenser System Heatup and Steady State Operation (Subsection 14.2.8.2.35.2).
- Power Maneuvering in the Feedwater Temperature Operating Domain (Subsection 14.2.8.2.35.3).
- Load Maneuvering Capability (Subsection 14.2.8.2.35.4).
- Defense-In-Depth Stability Solution Evaluation Test (Subsection 14.2.8.2.35.5).

Table 1D-1

Summary of Tier 2* Information

Location	Short Description of Tier 2* Information	Expiration
Chapter 1		
Table 1.6-1	Selected Licensing Topical Reports (LTRs) consistent with how they are marked at their referenced locations	As marked at their referenced locations later in this table
Table 1.9-22	Applicable Edition/Addenda for ASME Boiler and Pressure Vessel Code, Section III	First Full Power
Chapter 2		
Table 2.0-1	Standard Plant Site Parameters	First Full Power
Figure 2.0-1	Horizontal SSE Design Ground Spectra at Foundation Level	First Full Power
Figure 2.0-2	Vertical SSE Design Ground Response Spectra at Foundation Level	First Full Power
Chapter 3		
S3.6.2.1.1	Locations of Postulated Pipe Breaks	First Full Power
S3.6.2.1.2	Locations of Postulated Pipe Cracks	First Full Power
S3.6.2.5	Pipe Break Analysis Results and Protection Methods	First Full Power
S3.7	Seismic Design	First Full Power
S3.7.1.1.3	Single Envelope Ground Motion	First Full Power
S3.7.1.2	Percentage of Critical Damping Values	First Full Power
S3.7.1.3	Supporting Media for Category I Structures	First Full Power
S3.7.2	Seismic System Analysis	First Full Power
S3.7.2.1.2 b)	Response Spectrum Method, Multi-Supported System with ISMs	First Full Power
S3.7.2.1.3	Static Coefficient Method	First Full Power
S3.7.2.2	Natural Frequencies and Responses	First Full Power
S3.7.2.3	Procedures Used for Analytical Modeling	First Full Power
S3.7.2.4	Soil-Structure Interaction	First Full Power
S3.7.2.5	Development of Floor Response Spectra	First Full Power
S3.7.2.6	Three Components of Earthquake Motion	First Full Power

Table 1D-1

Summary of Tier 2* Information

Location	Short Description of Tier 2* Information	Expiration
S3.7.2.7	Combination of Modal Responses	First Full Power
S3.7.2.8.1	Turbine Building	First Full Power
S3.7.2.8.2	Radwaste Building	First Full Power
S3.7.2.8.3	Service Building	First Full Power
S3.7.2.8.4	Ancillary Diesel Building	First Full Power
S3.7.2.9	Effects of Parameter Variations on Floor Response Spectra	First Full Power
S3.7.2.10	Use of Equivalent Vertical Static Factors	First Full Power
S3.7.2.11	Methods Used to Account for Torsional Effects	First Full Power
S3.7.2.13	Analysis Procedure for Damping	First Full Power
S3.7.2.14	Determination of Seismic Category I Structure Overturning Moments	First Full Power
S3.7.3.1	Seismic Analysis Method	First Full Power
S3.7.3.2	Determination of Number of Earthquake Cycles	First Full Power
S3.7.3.3.1	Piping Systems	First Full Power
S3.7.3.3.2	Equipment	First Full Power
S3.7.3.3.3	Modeling of Special Engineered Pipe Supports	First Full Power
S3.7.3.5	Analysis Procedure for Damping	First Full Power
S3.7.3.6	Three Components of Earthquake Motion	First Full Power
S3.7.3.7	Combination of Modal Responses	First Full Power
S3.7.3.8	Interaction of Other Systems with Seismic Category I Systems	First Full Power
S3.7.3.9	Multiple-Supported Equipment and Components with Distinct Inputs	First Full Power
S3.7.3.10	Use of Equivalent Vertical Static Factors	First Full Power
S3.7.3.11	Torsional Effects of Eccentric Masses	First Full Power
S3.7.3.12	Effect of Differential Building Movements	First Full Power
S3.7.3.13	Seismic Category I Buried Piping, Conduits and Tunnels	First Full Power

Table 1D-1

Summary of Tier 2* Information

Location	Short Description of Tier 2* Information	Expiration
S3.7.3.14	Methods for Seismic Analysis of Seismic Category I Concrete Dams	First Full Power
S3.7.3.15	Methods for Seismic Analysis of Above-Ground Tanks	First Full Power
S3.7.3.16 (1)	Design of Small Branch and Small Bore Piping	First Full Power
S3.7.5	Site-Specific Information	First Full Power
Table 3.7-2	5%-Damped Target Spectra of Single Envelope Design Ground Motion at Foundation Level	First Full Power
Table 3.7-3	Summary of Methods of Seismic Analysis for Primary Building Structures	First Full Power
Figure 3.7-38	Single Envelope Spectrum Match – H1 Component	First Full Power
Figure 3.7-39	Single Envelope Spectrum Match – H2 Component	First Full Power
Figure 3.7-40	Single Envelope Spectrum Match – Vertical Component	First Full Power
Figure 3.7-41	Single Envelope Time Histories – H1 Component	First Full Power
Figure 3.7-42	Single Envelope Time Histories – H2 Component	First Full Power
Figure 3.7-43	Single Envelope Time Histories – Vertical Component	First Full Power
S3.8.1.1.1	Concrete Containment	First Full Power
S3.8.1.1.3	Containment Boundary	First Full Power
S3.8.1.2.2	Construction Codes of Practice	First Full Power
S3.8.1.2.3	General Design Criteria, Regulatory Guides, and Industry Standards	First Full Power
S3.8.1.3.6	Load Combinations for the Containment Structure and Liner Plate	First Full Power
S3.8.1.4.1.4	Corrosion Prevention	First Full Power
S3.8.1.5	Structural Acceptance Criteria	First Full Power
S3.8.1.6	Materials, Quality Control and Special Construction Techniques	First Full Power
S3.8.1.6.1	Concrete	First Full Power
S3.8.1.6.2	Reinforcing Steel	First Full Power
S3.8.1.6.3	Splices of Reinforcing Steel	First Full Power
S3.8.1.6.4	Liner Plate and Appurtenances	First Full Power

Table 1D-1

Summary of Tier 2* Information

Location	Short Description of Tier 2* Information	Expiration
S3.8.1.7.1	Structural Integrity Pressure Test	First Full Power
S3.8.1.7.3.12	Evaluation of Inaccessible Areas	First Full Power
S3.8.2.2.1	Codes, Standards and Regulatory Guides	First Full Power
S3.8.2.2.3	Code Compliance	First Full Power
S3.8.2.3	Loads and Load Combinations	First Full Power
S3.8.2.5	Structural Acceptance Criteria	First Full Power
S3.8.2.6	Materials, Quality Control and Special Construction Techniques	First Full Power
S3.8.3.1.1	Diaphragm Floor	First Full Power
S3.8.3.1.3	Reactor Shield Wall	First Full Power
S3.8.3.1.4	Vent Wall	First Full Power
S3.8.3.2	Applicable Codes, Standards, and Specifications	First Full Power
S3.8.3.3.2	Load Combination	First Full Power
S3.8.3.5.1	Diaphragm Floor	First Full Power
S3.8.3.5.2	Reactor Pressure Vessel Support Brackets	First Full Power
S3.8.3.5.3	Reactor Shield Wall	First Full Power
S3.8.3.5.4	Vent Wall	First Full Power
S3.8.3.5.5	Gravity Driven Cooling System Pool	First Full Power
S3.8.3.5.6	Miscellaneous Platforms	First Full Power
S3.8.3.6.1	Diaphragm Floor	First Full Power
S3.8.3.6.2	Reactor Pressure Vessel Support Brackets	First Full Power
S3.8.3.6.3	Reactor Shield Wall	First Full Power
S3.8.3.6.4	Vent Wall	First Full Power
S3.8.3.6.5	Gravity Driven Cooling System Pool	First Full Power
S3.8.3.6.6	Miscellaneous Platforms	First Full Power
S3.8.4	Other Seismic Category I Structures	First Full Power
S3.8.4.1.1	Reactor Building Structure	First Full Power
S3.8.4.1.2	Control Building	First Full Power
S3.8.4.1.3	Fuel Building	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
S3.8.4.1.4	Firewater Service Complex	First Full Power
S3.8.4.1.5	Radwaste Building	First Full Power
S3.8.4.2.1	Reactor Building	First Full Power
S3.8.4.2.2	Control Building	First Full Power
S3.8.4.2.3	Fuel Building	First Full Power
S3.8.4.2.4	Radwaste Building	First Full Power
S3.8.4.2.5	Welding of Pool Liners	First Full Power
S3.8.4.3.1.2	Load Combinations for Concrete Members	First Full Power
S3.8.4.3.1.3	Load Combinations for Steel Members	First Full Power
S3.8.4.3.2	Control Building	First Full Power
S3.8.4.3.3	Fuel Building	First Full Power
S3.8.4.3.4	Radwaste Building	First Full Power
S3.8.4.3.5	Firewater Service Complex	First Full Power
S3.8.4.5.1	Reactor Building	First Full Power
S3.8.4.5.2	Control Building	First Full Power
S3.8.4.5.3	Fuel Building	First Full Power
S3.8.4.5.4	Radwaste Building	First Full Power
S3.8.4.5.5	Firewater Service Complex	First Full Power
S3.8.4.6.1	Concrete	First Full Power
S3.8.4.6.2	Reinforcing Steel	First Full Power
S3.8.4.6.3	Splices of Reinforcing Steel	First Full Power
S3.8.4.6.4	Quality Control	First Full Power
S3.8.5.1	Description of the Foundations	First Full Power
S3.8.5.2	Applicable Codes, Standards and Specifications	First Full Power
S3.8.5.3	Loads and Load Combinations	First Full Power
S3.8.5.5	Structural Acceptance Criteria	First Full Power
S3.8.5.6	Materials, Quality Control, and Special Construction Techniques	First Full Power
S3.8.6.1	Foundation Waterproofing	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
S3.8.6.2	Site-Specific Physical Properties and Foundation Settlement	First Full Power
Table 3.8-1	Key Dimensions of Concrete Containment	First Full Power
Table 3.8-2	Load Combinations, Load Factors and Acceptance Criteria for the Reinforced Concrete Containment	First Full Power
Table 3.8-3	Major Allowable Stresses in Concrete and Reinforcing Steel	First Full Power
Table 3.8-4	Load Combination, Load Factors and Acceptance Criteria for Steel Containment Components of the RCCV	First Full Power
Table 3.8-5	Welding Activities and Weld Examination Requirements for Containment Vessel	First Full Power
Table 3.8-6, Ref 13, 15, 19, 20, 21, 22, 23	Codes, Standards, Specifications, and Regulations Used in the Design and Construction of Seismic Category I Internal Structures of the Containment	First Full Power
Table 3.8-7	Load Combination, Load Factors and Acceptance Criteria for Steel Structures Inside the Containment	First Full Power
Table 3.8-8	Key Dimensions of RB, CB, FB, RW and FWSC	First Full Power
Table 3.8-9 Ref 1, 2, 3, 4, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 40	Codes, Standards, Specifications, and Regulatory Guides Used in the Design and Construction of Seismic Category I Structures	First Full Power
Table 3.8-10	Temperatures During Operating Conditions (RB)	First Full Power
Table 3.8-11	Temperatures During Operating Conditions (CB)	First Full Power
Table 3.8-12	Temperatures During Operating Conditions (FB)	First Full Power
Table 3.8-13	Key Dimensions of Foundations	First Full Power
Table 3.8-14	Load Combinations and Factor of Safety for Foundation Design	First Full Power
Table 3.8-15	Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Reinforced Concrete Structures	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
Table 3.8-16	Load Combinations, Load Factors and Acceptance Criteria for the Safety-Related Steel Structures	First Full Power
Table 3.8-18	Temperatures During Operating Conditions (FWSC)	First Full Power
Figure 3.8-1	Configuration of Concrete Containment	First Full Power
S3.9.2.3	Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions	First Full Power
S3.9.3.1	Loading Combinations, Design Transients and Stress Limits	First Full Power
S3.9.3.1.1	Plant Conditions – Correlation of Plant Condition with Event Probability	First Full Power
S3.9.3.1.2	Inspections/Testing Following the Reactor Coolant System Exceeding Service Level B Pressure Limit	First Full Power
S3.9.3.2	Reactor Pressure Vessel Assembly	First Full Power
S3.9.3.3	Main Steam System Piping	First Full Power
S3.9.3.4	Other Components: • Main Steamline Isolation, Safety Relief, and Depressurization Valves • ASME Class 1, 2 and 3 Piping	First Full Power
S3.9.3.5	Valve Operability Assurance	First Full Power
S3.9.3.6	Design and Installation of Pressure Relief Devices: • Main Steam Safety Relief Valves • Other Safety Relief and Vacuum Breaker Valves	First Full Power
S3.9.3.7	Component Supports	First Full Power
S3.9.3.7.1	Piping Supports	First Full Power
S3.9.3.7.1 Item (3) b.	Inspection, Testing, Repair and/or Replacement of Snubbers	First Full Power
S3.9.3.7.1 Item (3) c. iii.	Snubber Design and Testing	First Full Power
S3.9.3.7.1 Item (e)	Snubber Preservice and Inservice Examination and Testing	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
S3.9.3.7.2	Reactor Pressure Vessel Sliding Supports	First Full Power
S3.9.3.9.1	Threaded Fasteners – ASME B&PV Code Class 1, 2 and 3 – Material Selection	First Full Power
S3.9.3.9.2	Threaded Fasteners – ASME B&PV Code Class 1, 2 and 3 – Special Materials Fabrication Processes and Special Controls	First Full Power
S3.9.3.9.3	Threaded Fasteners – ASME B&PV Code Class 1, 2 and 3 – Preservice and Inservice Inspection Requirements	First Full Power
Table 3.9-2	Load Combinations and Acceptance Criteria for Safety-Related, ASME B&PV Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures	First Full Power
Table 3.9-9	Load Combinations and Acceptance Criteria for Class 1 Piping Systems	First Full Power
Table 3.9-10	Snubber Loads	First Full Power
Table 3.9-11	Strut Loads	First Full Power
Table 3.9-12	Linear Type (Anchor and Guide) Main Steam Piping Support	First Full Power
S3.10.1.1	Selection of Qualification Method	First Full Power
Ref 3.11-6	LTR NEDE-33516P, ESBWR Qualification Plan Requirements for a 72-Hour Duty Cycle Battery	First Full Power
S3A.2	RB/FB complex, CB and FWSC shape, dimensions and embedment depths	First Full Power
Table 3A.2-1	Standard ESBWR Building Dimensions	First Full Power
S3A.3.1	Generic Site Conditions	First Full Power
S3A.3.2	North Anna ESP Site Conditions	First Full Power
Table 3A.3-1	Generic Site Properties for SSI Analysis	First Full Power
Table 3A.3-2	North Anna Site-specific Properties for SSI Analysis	First Full Power
Table 3A.3-3	Layered Site Cases	First Full Power
S3A.4.1	Input motion for SSI analysis	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
S3A.5, S3A.5.1, S3A.5.2	Soil-Structure Interaction Analysis Method	First Full Power
Table 3A.5-1	Soil Spring and Damping Coefficient for RB/FB complex	First Full Power
Table 3A.5-2	Soil Spring and Damping Coefficient for CB	First Full Power
Table 3A.5-3	Soil Spring and Damping Coefficient for FWSC	First Full Power
Figure 3A.5-1	Method for Frequency-Independent Soil Properties	First Full Power
S3A.6	Soil-Structure Interaction Analysis Cases	First Full Power
Table 3A.6-1	Seismic SSI Analysis Cases	First Full Power
S3A.7, S3A.7.1, S3A.7.2, S3A.7.3	Analysis Models	First Full Power
Figure 3A.7-1	RB/FB Stick Model	First Full Power
Figure 3A.7-2	RCCV Stick Model	First Full Power
Figure 3A.7-3	Pedestal Stick Model	First Full Power
Figure 3A.7-4	RB/FB Complex Seismic Model	First Full Power
Figure 3A.7-5	Control Building Stick Model	First Full Power
Figure 3A.7-6	Control Building Seismic Model	First Full Power
Figure 3A.7-7	FWSC Seismic Model	First Full Power
Figure 3A.7-8	SASSI2000 Plate Elements for RB/FB Basemat	First Full Power
Figure 3A.7-9	SASSI2000 Plate Elements for RB/FB Exterior Walls	First Full Power
Figure 3A.7-10	Overview of RB/FB SASSI2000 Model	First Full Power
Figure 3A.7-11	SASSI2000 Plate Elements for CB Basemat	First Full Power
Figure 3A.7-12	SASSI2000 Plate Elements for CB Exterior Walls	First Full Power
Figure 3A.7-13	Overview of CB SASSI2000 Model	First Full Power
Figure 3A.7-14	SASSI2000 Plate Elements for FWSC Basemat	First Full Power
Figure 3A.7-15	Overview of FWSC SASSI2000 Model	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
S3A.9, S3A.9.1, S3A.9.2, S3A.9.3	Site Envelope Seismic Responses	First Full Power
Table 3A.9-1a to 3A.9-1h	Enveloping Seismic Loads	First Full Power
Table 3A.9-2a to 3A.9-2e	Enveloping Seismic Loads for LOCA Flooding	First Full Power
Table 3A.9-3a to 3A.9-3i	Enveloping Maximum Vertical Acceleration	First Full Power
Table 3A.9-4a to 3A.9-4e	Enveloping Maximum Vertical Acceleration for LOCA Flooding	First Full Power
Figure 3A.9-1a to 3A.9-3l	Enveloping Floor Response Spectra	First Full Power
Appendix 3B	Containment Hydrodynamic Load Definitions	First Full Power
Table 3D.1-1	Computer Program User Details	First Full Power
Appendix 3F	Response of Structures to Containment Loads	First Full Power
Appendix 3G	Design Details and Evaluation Results of Seismic Category I Structures	First Full Power
Ref 3H.4-8	LTR NEDE-33536P/NEDO-33536, Control Building and Reactor Building Environmental Temperature Analysis for ESBWR	First Full Power
Appendix 3I	Designated NEDE-24326-1-P Material Which May Not Change Without Prior NRC Approval	First Full Power
Chapter 4		
Ref 4.2-4	LTR NEDE-33240P/NEDO-33240, GE14E Fuel Assembly Mechanical Design Report	None
Ref 4.2-5	LTR NEDC-33242P/NEDO-33242, GE14E for ESBWR Fuel Rod Thermal-Mechanical Design Report	None
Ref 4.2-8	LTR NEDE-33244P/NEDO-33244, ESBWR Marathon Control Rod Mechanical Design Report	None
Ref 4.2-9	LTR NEDE-33243P/NEDO-33243, ESBWR Marathon Control Rod Nuclear Design Report	None

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Location	Short Description of Tier 2* Information	Expiration
Ref 4.3-8, 4.4-21, 4A-2, 4D-27	LTR NEDC-33239P/NEDO-33239, GE14 for ESBWR Nuclear Design Report	None
Ref 4.3-10, 4.4-20, 4A-1, 4D-23	LTR NEDC-33326P/NEDO-33326, GE14E for ESBWR Initial Core Nuclear Design Report	First Full Power
Ref 4.4-12	LTR NEDC-33237P/NEDO-33237, GE14 for ESBWR Critical Power Correlation, Uncertainty, and OLMCPR Development	None
Ref 4.4-22	LTR NEDC-33456P/NEDO-33456, Full-Scale Pressure Drop Testing for a Simulated GE14E Fuel Bundle	First Full Power
S4B.1	Fuel Licensing Acceptance Criteria – General	None
S4B.3	Fuel Licensing Acceptance Criteria – Nuclear	None
S4B.6	Fuel Licensing Acceptance Criteria – Critical Power	None
Table 4B-1	Fuel Rod Thermal-Mechanical Design Criteria	None
S4C.1	Control Rod Licensing Acceptance Criteria - General	None
Ref 4D-19	LTR NEDE-33217P/NEDO-33217, ESBWR Man-Machine Interface System and Human Factors Engineering Implementation Plan	First Full Power
Chapter 5		
S5.2.1.1	10 CFR 50.55a compliance for seismic design of piping	First Full Power
S5.2.4.2	Accessibility requirements to support ASME B&PV Code Section XI examinations	First Full Power
Chapter 6		
S6.6.2	Accessibility requirements to support ASME B&PV Code Section XI examinations	First Full Power
Chapter 7		
Ref 7.1-8, 7B.3-3	LTR NEDE-33295P/NEDO-33295, ESBWR Cyber Security Program Plan	First Full Power
Ref 7.1-9, 7.2-1, 7.3-2, 7.4-2, 7.5-2, 7.8-4	LTR NEDE-33304P/NEDO-33304P, GEH ESBWR Setpoint Methodology	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
Ref 7.1-10, 7.2-4, 7.3-4, 7.8-3, 7B.3-2	LTR NEDE-33245P/NEDO-33245, ESBWR – Software Quality Assurance Program Manual	First Full Power
Ref 7.1-12, 7.2-3, 7.3-3, 7B.3-1	LTR NEDE-33226P/NEDO-33226, ESBWR – Software Management Program Manual	First Full Power
S7.2.1.3.5, S7.2.2.3.5, S7.3.5.3.5	BTP HICB-14 discussions about Software Management Program Manual and Software Quality Assurance Program Manual LTRs	First Full Power
S7.8.2.1	Software Quality Assurance Program Manual LTR	First Full Power
S7B.1	Software Development	First Full Power
Ref 7B.3-4	LTR NEDE-33217P/NEDO-33217, ESBWR Man-Machine Interface System and Human Factors Engineering Implementation Plan	First Full Power
Chapter 8		
	None	N/A
Chapter 9		
S9.1.2.4	Mechanical and structural design of spent fuel racks	None
Ref 9.1-2	LTR NEDC-33374P/NEDO-33374, Criticality Analysis for ESBWR Fuel Racks	None
Chapter 10		
	None	N/A
Chapter 11		
	None	N/A
Chapter 12		
	None	N/A
Chapter 13		
Ref 13.3-1, 13.5-1	LTR NEDE-33217P/NEDO-33217, ESBWR Man-Machine Interface System and Human Factors Engineering Implementation Plan	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
Chapter 14		
S14.2.8.2.7	First of a kind testing for reactor stability	Successful completion of testing for First ESBWR
S14.2.8.2.35, S14.2.8.2.35.1, S14.2.8.2.35.2, S14.2.8.2.35.3, S14.2.8.2.35.4, S14.2.8.2.35.5	First of a Kind Tests	Successful completion of testing for First ESBWR
S14.3A.2	Design Acceptance Criteria ITAAC for Piping Design	First Full Power
S14.3A.3	Digital Instrumentation and Control Design Acceptance Criteria ITAAC Closure	First Full Power
S14.3A.4	Human Factors Engineering Design Acceptance Criteria ITAAC Closure	First Full Power
Chapter 15		
Ref 15.0-6, 15.2-2, 15.3-4, 15.5-6	LTR NEDC-33239P/NEDO-33239, GE14 for ESBWR Nuclear Design Report	None
Ref 15.0-7, 15.2-3, 15.3-5, 15.5-3	LTR NEDC-33326P/NEDO-33326, GE14E for ESBWR Initial Core Nuclear Design Report	First Full Power
Chapter 16 and 16B		
	None	N/A
Chapter 17		
S17.1.3	Software Quality Assurance Program Manual LTR, Software design verification and validation	First Full Power
Ref 17.1-2	LTR NEDE-33245P/NEDO-33245, ESBWR – Software Quality Assurance Program Manual	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
Chapter 18		
Ref 18.1-1, 18.2-1, 18.3-1, 18.4-1, 18.5-1, 18.6-1, 18.7-1, 18.8-1, 18.9-1, 18.10-1, 18.11-1, 18.12-1, 18.13-1	LTR NEDE-33217P/NEDO-33217, ESBWR Man-Machine Interface System and Human Factors Engineering Implementation Plan	First Full Power
Ref 18.3-2	LTR NEDO-33262, ESBWR Human Factors Engineering Operating Experience Review Implementation Plan	First Full Power
Ref 18.4-2	LTR NEDO-33219, ESBWR Human Factors Engineering Functional Requirements Analysis Implementation Plan	First Full Power
Ref 18.4-3	LTR NEDE-33220P/NEDO-33220, ESBWR Human Factors Engineering Allocation of Function Implementation Plan	First Full Power
Ref 18.5-2	LTR NEDE-33221P/NEDO-33221, ESBWR Human Factors Engineering Task Analysis Implementation Plan	First Full Power
Ref 18.6-2	LTR NEDO-33266, ESBWR Human Factors Engineering Staffing and Qualifications Implementation Plan	First Full Power
Ref 18.7-2	LTR NEDO-33267, ESBWR Human Factors Engineering Human Reliability Analysis Implementation Plan	First Full Power
Ref 18.8-2	LTR NEDE-33268P/NEDO-33268, ESBWR Human Factors Engineering Human-System Interface Design Implementation Plan	First Full Power
Ref 18.11-2	LTR NEDE-33276P/NEDO-33276, ESBWR Human Factors Engineering Verification and Validation Implementation Plan	First Full Power
Ref 18.12-2	LTR NEDO-33278, ESBWR Human Factors Engineering Design Implementation Plan	First Full Power

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Location	Short Description of Tier 2* Information	Expiration
Ref 18.13-2	LTR NEDO-33277, ESBWR Human Factors Engineering Human Performance Monitoring Implementation Plan	First Full Power
Chapter 19		
	None	N/A