1.9 Conformance with Regulatory Criteria

This section provides a guide to U.S. EPR conformance with regulatory criteria. These criteria include Regulatory Guides (RG), Standard Review Plans (SRP), Generic Issues, (including Three Mile Island (TMI) requirements), Operational Experience (Generic Communications) and Advanced and Evolutionary Light-Water Reactor design issues. Conformance is assessed to regulatory criteria in effect six months before the anticipated docket date of the U.S. EPR design certification application.

Table 1.9-1—U.S. EPR Conformance Table Legend, defines the codes used to indicate conformance determinations in the "U.S. EPR Assessment" columns of Table 1.9-2—U.S. EPR Conformance with Regulatory Guides, Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933), and Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087). Multiple assessment codes from Table 1.9-1 are identified, as necessary, for individual regulatory criteria in the conformance demonstration tables (Table 1.9-2 through Table 1.9-4) to address U.S. EPR conformance for these cases:

- NRC guidance expectations to address specific conformance with a regulatory criterion in multiple Final Safety Analysis Report (FSAR) sections.
- Exceptions to specific, limited portions of a listed regulatory criterion based on specific applicability to the U.S. EPR design certification.
- Exceptions or clarifications necessary to address potential guidance conflicts and the relative pertinence and applicability of the overall guidance to the U.S. EPR design certification.

ANP-10292, U.S. EPR Conformance with Standard Review Plan (NUREG-0800) (Reference 1), provides the results of a review of the U.S. EPR conformance with the acceptance criteria for each section of the SRP, NUREG-0800 (March 2007 for Chapters 1-18 and June 2007 for Chapter 19) (Reference 2).

A COL applicant that references the U.S. EPR design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.

1.9.1 Conformance with Regulatory Guides

Table 1.9-2 provides a listing of the regulatory positions in each RG from Divisions 1, 4, 5 and 8, as they apply to the U.S. EPR design certification. The applicability of each active Division 1 RG to the U.S. EPR design certification is addressed in Table 1.9-2; with the exception that those Division 1 RGs either withdrawn by the NRC staff or not yet issued are not included in Table 1.9-2. For Divisions 4, 5 and 8, only those RGs with potential applicability to nuclear power plant designs are addressed in Table 1.9-2. The RGs in Divisions 2, 3, 6, 7, 9 and 10 provide guidance for facilities other than power reactors; therefore, they do not apply to the U.S. EPR design certification.

For each RG in Table 1.9-2, a brief U.S. EPR conformance assessment notation, including annotation of any exceptions, is provided. The table listing also provides a cross-reference from each RG to the FSAR sections that address it. The RGs not applicable to the U.S. EPR design certification are so noted.

1.9.2 Conformance with the Standard Review Plan

The SRP conformance assessment in accordance with 10 CFR 52.47(a)(9) is provided in Reference 1. Compliance is evaluated against technically relevant portions of Reference 2 current six months prior to the application for design certification. The SRP conformance table in Reference 1 contains a listing of SRP sections, Branch Technical Positions (BTP) and the associated SRP section or BTP revision. For those SRP sections technically relevant to the U.S. EPR design certification, the SRP conformance table also lists individual SRP acceptance criteria requirements and specific acceptance criteria from each SRP section. For each SRP and BTP criterion listed in the SRP conformance table, a brief U.S. EPR conformance assessment notation, including annotation of any exceptions, is provided along with a reference to the FSAR section where the information is described. The SRP and BTP criteria that are not applicable to the U.S. EPR design certification are noted as such. Because BTPs provide specific or clarifying guidance for individual SRP specific acceptance criteria, conformance with BTP criteria are addressed in the SRP conformance table by referencing the U.S. EPR conformance assessment notation(s) of the governing SRP specific acceptance criteria.

1.9.3 Generic Issues

In accordance with 10 CFR 52.47(a)(8), conformance must be assessed against technically relevant TMI requirements identified in 10 CFR 50.34(f), except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Plant characteristics and plant programs that address relevant TMI requirements are described in the appropriate FSAR sections.

In accordance with 10 CFR 52.47(a)(21), proposed resolutions must be identified for any technically relevant unresolved safety issues (USI) and medium- to high-priority generic safety issues (GSI) identified in the version of NUREG-0933 (Reference 3) that is current six months prior to the application for design certification. Resolution and closure of generic issues is managed via the NRC Generic Issues Program. NRC letter SECY-07-0110, July 6, 2007 (Reference 4) provides the most recent supplemental status report of the Generic Issues Program prior to the FSAR submittal. As such,

Appendix B of NUREG-0933, Rev. 21 (Reference 3) (including the Main Report and Supplements 1- 31) and NRC letter SECY-07-0110 (Reference 4), were used to identify those generic issues applicable to the U.S. EPR design certification.

Table 1.9-3 identifies the applicable TMI requirements and generic issues, along with an abbreviated summary description of the NRC position for each table entry. Table 1.9-3 also provides a brief U.S. EPR conformance assessment notation, including annotation of any exceptions, and a reference to the FSAR section(s) addressing the issue. Those 10 CFR 50.34(f) TMI requirements specifically applicable to another nuclear power plant design concept, such as a boiling water reactor (BWR), nuclear power plant designs of another vendor or passive plant designs, are not included in Table 1.9-3, because they are not technically relevant to the U.S. EPR design. Those NUREG-0933 generic issues (Reference 3) determined as non-applicable were eliminated from consideration in Table 1.9-3 based on these:

- Resolved: Issue has been completely resolved and removed from the latest Generic Issues Program list of Active and Regulatory Office Implementation Generic Issues (Reference 4).
- BWR, Ice Condenser Containment or Other: Issue applies to another nuclear power plant design concept or to the design of a nuclear facility other than a nuclear power plant.

1.9.4 Operational Experience (Generic Communications)

10 CFR 52.47(a)(22) requires that applicants for design certification of new plant designs include a description of how operational experience has been incorporated into the design process. Operational experience insights are incorporated into applicable SRP sections as they are updated. Operational experience from NRC Bulletins and Generic Letters not incorporated into the most recent applicable SRP six months before the application docket date are incorporated into the design unless stated otherwise. The U.S. EPR design is an evolution of nuclear power plant designs that have been operated in the U.S., as addressed by 10 CFR 52.41(b)(1); hence NRC guidance for technically relevant operational experience issues is addressed in the appropriate FSAR sections.

The most recent SRP updates relative to the U.S EPR design certification occurred in March 2007 (for SRP Chapters 1-18) and June 2007 (for SRP Chapter 19). In the time period from the mentioned SRP updates to June 14, 2007, no NRC Bulletins and Generic Letters were issued. Therefore, the conformance assessment of the U.S. EPR design certification relative to operational experience is satisfied per the assessment provided for SRP guidance in Reference 1.

1.9.5 Advanced and Evolutionary Light-Water Reactor Design Issues

Guidance in SRP Section 1.0 recommends that this section address the applicable licensing and policy issues developed by the NRC and documented in SECY-93-087 (Reference 5) and the associated Staff Requirements Memoranda (SRM).

Table 1.9-4 lists the design issues identified in Reference 5 and provides a brief U.S. EPR conformance assessment notation, including annotation of any exceptions, for each issue. Table 1.9-4 also provides a cross-reference from the SECY issues to the FSAR sections that address them.

1.9.6 References

1. ANP-10292, "U.S. EPR Conformance with SRP Acceptance Criteria," AREVA NP Inc., Rev. 1, May 2009.

2. NUREG-0800," Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007, as amended by the subsequent revision of Chapter 19, June 2007.

3. NUREG-0933, "A Prioritization of Generic safety Issues," U.S. Nuclear Regulatory Commission September, Revision 21, 2007.

4. NRC letter SECY-07-0110, "Generic Issues Program list of Active and Regulatory Office Implementation Generic Issues," Secretary of the Commission, Office of the U.S. Nuclear Regulatory Commission, July 6, 2007.

5. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," Secretary of the Commission, Office of the U.S. Nuclear Regulatory Commission, April, 2, 1993, as supplemented by the associated Staff Requirements Memoranda dated July 21, 1993.

Table 1.9-1—U.S. EPR Conformance Table Legend
Sheet 1 of 2

Assessment Code	Description
Y	The U.S. EPR design conforms to relevant aspects of the associated NRC guidance as stipulated within the specific context of the cited guidance statement.
EXCEPTION- (“Regulatory Position”)	The U.S. EPR design employs an alternative approach relative to the NRC guidance in the cited regulatory position statement. The affected regulatory position statement is provided in the accompanying parentheses – e.g., “(SRP-SAC-03).” The exception is described and justified in the text of the noted FSAR section.
EXEMPTION (“Regulatory Position”)	The U.S. EPR design intends to employ an alternative approach relative to the NRC rulemaking in the cited regulatory position statement. The affected regulatory position statement is provided in the accompanying parentheses - e.g., “(SRP-SAC-03).” The basis exemption is described and justified in the text of the noted FSAR section.
ITAAC	Guidance applies to the evaluation of the need for design certification-related inspections, test, and acceptance criteria (ITAAC) and the development of necessary and appropriate ITAAC. ITAAC evaluations and any resultant ITAAC are incorporated into Tier1 of the U.S. EPR FSAR.
N/A-BWR	Guidance is only applicable to BWR nuclear power plant designs and is not applicable to the U.S. EPR design certification.
N/A-CLASS	Guidance is only applicable to structures, systems, and components (SSC) carrying a particular quality or seismic classification, as noted in the guidance statement. Based on the design configuration of the U.S EPR, the noted classification does not apply; hence, the guidance is not applicable to the U.S. EPR design certification.
N/A-COL	Guidance addresses concerns not addressed with the context of a design certification application and must be addressed by a COL applicant referencing the U.S. EPR design certification.
N/A-CP/OL	Guidance is only applicable to construction permit (CP) or operating license (OL) applications or to holders of a CP or OL filed under 10 CFR 50 and is not applicable to the U.S. EPR design certification.
N/A-ESP	Guidance is only applicable to applicants submitting an early site permit (ESP) application and is not applicable to the U.S. EPR design certification.
N/A-ICE	Guidance is only applicable to nuclear power plant designs employing an ice condenser containment feature and is not applicable to the U.S. EPR design certification.
N/A-INFO	Cited NRC guidance document content only provides information or clarification, does not contain specific nuclear power plant design certification applicant conformance expectations, and therefore, is not applicable to the U.S. EPR design certification.
N/A-OPT (Refer to “Selected Option”)	Guidance provides multiple options for satisfying acceptance criteria, and another of the available options has been implemented. A reference to the implemented option is provided in the accompanying parentheses – e.g., “(Refer to SRP 15.8, 15.8-AC-01).”

**Table 1.9-1—U.S. EPR Conformance Table Legend
Sheet 2 of 2**

Assessment Code	Description
N/A-OTHER	Guidance is applicable to a nuclear power plant feature or concept not employed by the U.S. EPR design and is not applicable to the U.S. EPR design certification.
N/A-PAS	Guidance is only applicable to passive nuclear power plant designs and is not applicable to the U.S. EPR design certification.
N/A-SUP (Refer to "Regulatory Reference")	Guidance is only applicable to previous nuclear power plant license applications, is replaced by superseding guidance for applications on or after this date, and is not applicable to the U.S. EPR design certification. The superseding guidance reference for design certification applicants is provided in the accompanying parentheses – e.g., "(Refer to RG 1.145)."
N/A-VEN	Guidance applicability is specifically limited to a unique technical issue identified with the nuclear power plant design of another vendor and is not applicable to the U.S. EPR design certification.

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
Sheet 1 of 19

RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
Division 1 Regulatory Guides			
1.1, 11/1970	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	Y	6.3.2.2
1.3, R2	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	N/A-BWR	N/A
1.4, R2	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	N/A-SUP (Refer to RG 1.145 and RG 1.183)	N/A
1.5, 03/1971	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	N/A-BWR	N/A
1.6, 03/1971	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	Y	Table 8.1-1
			8.3.1.2.4
			8.3.2.2.3
1.7, R4	Control of Combustible Gas Concentrations in Containment	Y	3.8.1.3.1
			3.8.2.3.1
			6.2.5.3.3
1.8, R3	Qualification and Training of Personnel for Nuclear Power Plants	N/A-COL	N/A
1.9, R4	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	Y	Table 8.1-1
			8.3
			8.4
1.11, 03/1971	Instrument Lines Penetrating Primary Reactor Containment	Y	3.6.2
			6.2.4
1.12, R2	Nuclear Power Plant Instrumentation for Earthquakes	Y	3.7.4
1.13, R2	Spent Fuel Storage Facility Design Basis	Y	3.5
			9.1.2
			9.1.3
			9.1.5.2.2
			9.4.2.1
1.14, R1	Reactor Coolant Pump Flywheel Integrity	Y	5.4.1

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.16, R4	Reporting of Operating Information -- Appendix A Technical Specifications	N/A-COL	N/A
1.20, R3	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	Y	3.9.2.3 and 3.9.2.4
			14.2.12
		EXCEPTION (Vibration testing of SG internals and condensate system)	3.9.2.3 and 3.9.2.4
			14.2.12
1.21, R1	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	Y	11.5.1 and 14.2.12
1.22, 02/1972	Periodic Testing of Protection System Actuation Functions	Y	7.1.3.4.1
1.23, R1	Meteorological Monitoring Programs for Nuclear Power Plants	Y	2.3
1.24, 03/1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	N/A-SUP (Refer to RG 1.145)	N/A
1.25, 03/1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	N/A-SUP (Refer to RG 1.145 and RG 1.183)	N/A
1.26, R4	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	Y	3.2.2
1.27, R2	Ultimate Heat Sink for Nuclear Power Plants	Y	3.5
			9.2.5
1.28, R3	Quality Assurance Program Requirements (Design and Construction)	Y	14.2.6
		Y (Per AREVA Topical Report ANP-10266A	17.5

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.29, R4	Seismic Design Classification	Y	3.2.1
1.30, 08/1972	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	N/A-COL	N/A
1.31, R3	Control of Ferrite Content in Stainless Steel Weld Metal	Y	3.6.3
			5.2.3
			6.1.1
1.32,R3	Criteria for Power Systems for Nuclear Power Plants	Y	6.3
			Table 8.1-1
			8.2
			8.3.2.2.3
1.33,R2	Quality Assurance Program Requirements (Operation)	N/A-COL	N/A
1.34, 12/1972	Control of Electroslag Weld Properties	Y	5.2.3.4
1.35, R3	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments	N/A-OTHER	N/A
1.35.1, 07/1990	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	Y	3.8.1.2.5 3.8.1.7.2
1.36, 02/1973	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	Y	5.2.3.4.3
			6.1.1
1.37, R1	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	Y	3.13
			5.2.3
			5.3.1
			5.4.2
			6.1.1
			17.5
			10.3
1.38, R2	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	N/A-COL	N/A
1.39, R2	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	N/A-COL	N/A
1.40, 03/1973	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	Y	3.11

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.41, 03/1973	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments	Y	14.2
1.43, 05/1973	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	Y	5.2.3
1.44, 05/1973	Control of the Use of Sensitized Stainless Steel	Y	3.6.3.3.4
			5.2.3
			6.1.1
1.45, R1, 5/2008	Reactor Coolant Pressure Boundary Leakage Detection Systems	Y	3.6.3
			5.2.5
1.47, 05/1973	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Y	7.1.3.4.2
			7.5.2.2.4
			Table 8.1-1
			8.3.2.2.4
1.50, 05/1973	Control of Preheat Temperature for Welding of Low-Alloy Steel	Y	5.2.3
			6.1.1
1.52, R3	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	Y	6.2.3.2
			6.4.2.2
			6.5
			9.4.1.1
			9.4.5.1
			9.4.7.2.1
			12.3.3.3
			12.3.6.5.6
			14.2
1.53, R2	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Y	7.1.3.4.3
			8.1.4
			8.3.2.2.3
			15.2
			15.3
1.54, R1	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants	Y	6.1.2.4
		EXCEPTION (Misc. codes)	6.1.2.4

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.56, R1	Maintenance of Water Purity in Boiling Water Reactors	N/A-BWR	N/A
1.57, R1	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	Y	3.8.2.3.2
1.59, R2 (with Errata 07/30/ 1980)	Design Basis Floods for Nuclear Power Plants	Y	3.4
1.60, R1	Design Response Spectra for Seismic Design of Nuclear Power Plants	Y	3.7.1
1.61, R1	Damping Values for Seismic Design of Nuclear Power Plants	Y	3.7.1.2
			Table 3.7.1-1
			3.8.3.2.5
			3.8.3.4.4
			3.8.3.2.5
			3.8.4.4.1
			Appendix 3D Attach E
1.62, 10/1973	Manual Initiation of Protective Actions	Y	7.1.3.4.4
1.63, R3	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	Y	3.11.2.3
			8.1
			8.3.1
1.65, 10/1973	Materials and Inspections for Reactor Vessel Closure Studs	Y	3.13
			5.3
1.68, R3	Initial Test Programs for Water-Cooled Nuclear Power Plants	Y	3.9.2
			4.4.5
			14.2
1.68.1, R1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	N/A-BWR	N/A
1.68.2, R1	Initial Startup Test Program To Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	Y	14.2
1.68.3, 04/1982	Preoperational Testing of Instrument and Control Air Systems	Y	14.2

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.69, 12/1973	Concrete Radiation Shields for Nuclear Power Plants	Y	3.8.3.2
			3.8.4.2.5
			12.3
1.70, R3	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	N/A-SUP (Refer to RG 1.206)	N/A
1.71, R1	Welder Qualification for Areas of Limited Accessibility	Y	5.2.3
			6.1.1
			10.3
1.72, R2	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	N/A-COL	N/A
1.73, 01/1974	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	Y	3.11.2.3
1.75, R3	Physical Independence of Electric Systems	Y	7.1.3.4.5
			Table 8.1-1
			8.3.2.2.3
			9.5.1.2
1.76, R1	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	Y	3.3.2
			3.5
1.77, 05/1974	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	Y	4.2
			15.0.3
1.78, R1	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	Y	2.3.4
			6.4
			9.4.1.1
			14.2
1.79, R1	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	Y	14.2
1.81, R1	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	N/A-OTHER N/A-COL	N/A
1.82, R3	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	Y	6.3

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.84, R33	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	Y	3.8.1
			3.8.2
			4.5.2
			5.2.1
			5.4.2.4.1
			10.3
1.86, 06/1974	Termination of Operating Licenses for Nuclear Reactors	N/A-COL	N/A
1.87, 06/1975	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors	N/A-OTHER	N/A
1.89, R1	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	Y	3.11
			Appendix 3D Attach C
1.90, R1	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	Y	3.8.1.2.5 3.8.1.7.2
1.91, R1	Evaluations of Explosions Postulated To Occur on Transportation Routes Near Nuclear Power Plants	N/A-COL	N/A
1.92, R2	Combining Modal Responses and Spatial Components in Seismic Response Analysis	Y	3.7.2
			3.7.3
			Appendix 3D Attach E
1.93, 12/1974	Availability of Electric Power Sources	Y	16.B3.8
1.94, R1	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	Y	3.8.1.2.5
1.96, R1	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	N/A-BWR	N/A

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.97, R4	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants	Y	3.10
			3.11.2
			7.1.3.4.6
			7.5
			11.5
			12.3
			16.B3.3
			18.7
1.98, 03/1976	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	N/A-BWR	N/A
1.99, R2	Radiation Embrittlement of Reactor Vessel Materials	Y	5.3.1
			5.3.2
1.100, R3	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	Y	3.10
			3.11
			3.9.3, 3.9.6, Appendix 3D Attach E
1.101, R5	Emergency Planning and Preparedness for Nuclear Power Reactors	N/A-COL	N/A
1.102, R1	Flood Protection for Nuclear Power Plants	Y	3.4
1.105, R3	Setpoints for Safety-Related Instrumentation	Y	15.1
			15.2
			15.3
			15.4
		Y	18.7
		EXCEPTION (As addressed in AREVA Topical Report ANP-10275P-A)	7.1.3.4.7
1.106, R1	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Y	8.3.1.1

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.107, R1	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	Y	3.8.1
1.109, R1	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I	Y	11.2
			11.3
1.110, 03/1976	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	Y	11.2
			11.3
1.111, R1	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	N/A-COL	2.3.5
1.112, R1	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	Y	11.1
			12.2
1.113, R1	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	N/A-COL	N/A
1.114, R2	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	N/A-COL	N/A
1.115, R1	Protection Against Low-Trajectory Turbine Missiles	Y	3.5
			3.8.4
			9.1.2
			10.3
1.116, R0-R	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	Y	14.2
1.117, R1	Tornado Design Classification	Y	3.5
			10.3
1.118, R3	Periodic Testing of Electric Power and Protection Systems	Y	7.1.3.4.8
			8.1
			8.3
			14.2
1.121, 08/1976	Bases for Plugging Degraded PWR Steam Generator Tubes	Y	5.4.2

Table 1.9-2—U.S. EPR Conformance with Regulatory Guides
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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.122, R1	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	Y	3.7.2 3.7.3.1
1.124, R2	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	Y	3.9.3 9.1
1.125, R1	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	N/A-COL	N/A
1.126, R1	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	Y (Per AREVA Topical Report ANP-10231P)	4.2
1.127, R1	Inspection of Water-Control Structures Associated with Nuclear Power Plants	N/A-COL	N/A
1.128, R2	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Y	Table 8.1-1 8.3.2.2.3 9.4.6.1
1.129, R2	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Y	Table 8.1-1 8.3.2.2.3
1.130, R2	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	Y	3.9.3
1.131, 08/1977	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	Y	3.11.2.1
1.132, R2	Site Investigations for Foundations of Nuclear Power Plants	N/A-COL	N/A
1.133, R1	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	Y	4.4.6.6
1.134, R3	Medical Evaluation of Licensed Personnel at Nuclear Power Plants	N/A-COL	N/A
1.135, 09/1977	Normal Water Level and Discharge at Nuclear Power Plants	N/A-COL	N/A

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.136, R3	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	Y	3.8.1.2.5
			3.8.1.3
			3.8.2.2.5
			3.8.3.2.5
		EXCEPTION (2001 Ed. ASME Code)	3.8.1.2.5
			3.8.1.3
			3.8.2.2.5
			3.8.3.2.5
1.137, R1	Fuel-Oil Systems for Standby Diesel Generators	Y	9.5.4
1.138, R2	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	N/A-COL	N/A
1.139, 05/1978	Guidance for Residual Heat Removal	Y	14.2
1.140, R2	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants	Y	9.4.1.1
			9.4.2.1
			9.4.3.1
			9.4.5.1
			9.4.7.2.1
			9.4.8
			12.3.3.3
			12.3.6.5.6
			14.2
1.141, 04/1978	Containment Isolation Provisions for Fluid Systems	Y	6.2.4

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.142, R2	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	Y	3.5.3.2
			3.5.3.3
			3.8.3.2.5
			3.8.3.3.1
			3.8.4.2.5
			3.8.5.2
		EXCEPTION (ACI-349 2001 edition used)	3.5.3.2
			3.5.3.3
			3.8.3.2.5
			3.8.3.3.1
			3.8.4.2.5
			3.8.5.2
1.143, R2	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	Y	3.2.1
			3.7.2
			3.10
			10.4.8
			11.2
			11.3
			11.4
1.145, R1 (reissued 02/1983)	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	Y	2.3.4
1.147, R14	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	Y	5.2.1.2
1.148, 03/1981	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	Y	3.10.1.1
1.149, R3	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations	N/A-COL	N/A
1.150, R1	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations	Y	5.2.4
1.151, 07/1983	Instrument Sensing Lines	Y	3.2.1
			7.1.3.4.9

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.152, R2	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.10
1.153, R1	Criteria for Safety Systems	Y	Table 8.1-1
			8.3.2.2.3
1.154, 01/1987	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	N/A-COL	N/A
1.155, 08/1988	Station Blackout	Y	6.2.4
			7.4
			Table 8.1-1
			8.2.2.5
			8.3.1
			8.3.2.2.7
			8.4
			9.4.1
			9.4.5
			10.3
			19.2.2
1.156, 11/1987	Environmental Qualification of Connection Assemblies for Nuclear Power Plants	Y	3.11
1.157, 05/1989	Best-Estimate Calculations of Emergency Core Cooling System Performance	Y	15.6.5
1.158, 02/1989	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	Y	3.11
			8.1
			8.3.2.1.1.2
1.159, R1	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	N/A-COL	N/A
1.160, R2	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	N/A-COL	N/A
1.161, 06/1995	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	N/A-COL	N/A
1.162, 02/1996	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	N/A-COL	N/A
1.163, 09/1995	Performance-Based Containment Leak-Test Program	Y	6.2.6

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.165, 03/1997	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion	Y	2.5.2
1.166, 03/1997	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions	Y	3.7.4.4
1.167, 03/1997	Restart of a Nuclear Power Plant Shut Down by a Seismic Event	N/A-COL	N/A
1.168, R1	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.11
1.169, 09/1997	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.12
1.170, 09/1997	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.13
1.171, 09/1997	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.14
1.172, 09/1997	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.15
1.173, 09/1997	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y	7.1.3.4.16
1.174, R1	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	N/A-COL	N/A
1.175, 08/1998	An Approach for Plant-Specific, Risk-Informed Decision making: Inservice Testing	N/A-COL	N/A
1.176, 08/1998	An Approach for Plant-Specific, Risk-Informed Decision making: Graded Quality Assurance	N/A-COL	N/A
1.177, 08/1998	An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications	N/A-COL	N/A
1.178, R1	An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping	N/A-COL	N/A
1.179, 01/1999	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	N/A-COL	N/A

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.180, R1	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	Y	3.11
			7.1.3.4.17
1.181, 09/1999	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	N/A-COL	N/A
1.182, 05/2000	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants	N/A-COL	N/A
1.183, 07/2000	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	Y	11.1.2.2
			12.3.5
			15.0.3
1.184, 07/2000	Decommissioning of Nuclear Power Reactors	N/A-COL	N/A
1.185, 07/2000	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	N/A-COL	N/A
1.186, 12/2000	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	N/A-OTHER	N/A
1.187, 11/2000	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	N/A-COL	N/A
1.188, R1	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses	N/A-COL	N/A
1.189, R01	Fire Protection for Nuclear Power Plants	Y	3.2
			7.1.3.4.18
			7.4
			5.4.1.2.2
			9.5.1
		EXCEPTION (Misc. Regulatory Positions)	9.5.1
1.190, 03/2001	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	Y	5.3
1.191, 05/2001	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	N/A-COL	N/A
1.192, 06/2003	Operation and Maintenance Code Case Acceptability, ASME OM Code	Y	3.9.6
			5.2.1
1.193, R1	ASME Code Cases Not Approved for Use	Y	3.8.2

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.194, 06/2003	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	Y	2.3.4
1.195, 05/2003	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	Y	15.4.8
1.196, R1	Control Room Habitability at Light-Water Nuclear Power Reactors	Y	6.4
1.197, 06/2003	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	Y	6.4.5
1.198, 11/2003	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	N/A-COL	N/A
1.199, 11/2003	Anchoring Components and Structural Supports in Concrete	Y	3.8.1.2.5
			3.8.1.4.10
			3.8.3.2.5
			3.8.4.2.5
			3.8.5.2
		EXCEPTION (Appendix D to ACI-349 2006 edition used)	3.8.1.2.5
			3.8.1.4.10
			3.8.3.2.5
			3.8.4.2.5
			3.8.5.2
1.200, R1	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	N/A-OTHER	N/A
1.201, R1	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance	N/A-OTHER	N/A
1.202, 02/2005	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	N/A-COL	N/A
1.203, 12/2005	Transient and Accident Analysis Methods	N/A-OTHER	N/A
1.204, 11/2005	Guidelines for Lightning Protection of Nuclear Power Plants	Y	7.1.3.4.19
			8.1.4.3
			8.3.1.2

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
1.205, 05/2006	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	N/A-COL	N/A
1.206, 06/2007	Combined License Applications for Nuclear Power Plants (LWR Edition)	Y	1.1.6.1
1.207, 03/2007	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors	Y	3.12.5
1.208, 03/2007	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	Y	2.5 3.7.1
1.209, 03/2007	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	Y	3.11.2 7.1.3.4.20
1.216, 08/2010	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	Y	3.8.1.2.5 3.8.2.2.5
Division 4 Regulatory Guides			
4.1, R1	Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants	N/A-COL	N/A
4.2, R2	Preparation of Environmental Reports for Nuclear Power Stations	N/A-COL	N/A
4.2S1, 09/2002	Supplement 1 to Regulatory Guide 4.2, Preparation of Supplemental Environmental Reports for Applications To Renew Nuclear Power Plant Operating Licenses	N/A-COL	N/A
4.4, 05/1974	Reporting Procedure for Mathematical Models Selected To Predict Heated Effluent Dispersion in Natural Water Bodies	N/A-COL	N/A
4.7, R2	General Site Suitability Criteria for Nuclear Power Stations	N/A-COL	N/A
4.8, 12/1975	Environmental Technical Specifications for Nuclear Power Plants	N/A-COL	N/A
4.11, R1	Terrestrial Environmental Studies for Nuclear Power Stations	N/A-COL	N/A
4.15, R2	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) -- Effluent Streams and the Environment	Y	11.5.1 and 14.2.12

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
Division 5 Regulatory Guides			
5.7, R1	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	N/A-COL	N/A
5.12, 11/1973	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	N/A-COL	N/A
5.44, R3	Perimeter Intrusion Alarm Systems	N/A-COL	N/A
5.54, 03/1978	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants	N/A-COL	N/A
5.65, 09/1986	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	N/A-COL	N/A
5.66, 06/1991	Access Authorization Program for Nuclear Power Plants	N/A-COL	N/A
5.68, 08/1994	Protection Against Malevolent Use of Vehicles at Nuclear Power Plants	N/A-COL	N/A
Division 8 Regulatory Guides			
8.2, 02/1973	Guide for Administrative Practices in Radiation Monitoring	N/A-COL	N/A
8.4, 02/1973	Direct-Reading and Indirect-Reading Pocket Dosimeters	N/A-COL	N/A
8.5, R1	Criticality and Other Interior Evacuation Signals	N/A-COL	N/A
8.6, 05/1973	Standard Test Procedure for Geiger-Muller Counters	N/A-COL	N/A
8.7, R2	Instructions for Recording and Reporting Occupational Radiation Exposure Data	N/A-COL	N/A
8.8, R3	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable	Y	11.4 12.1.1 12.3
8.9, R1	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	N/A-COL	N/A
8.10, R1-R	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	N/A-COL	N/A
8.13, R3	Instruction Concerning Prenatal Radiation Exposure	N/A-COL	N/A
8.15, R1	Acceptable Programs for Respiratory Protection	N/A-COL	N/A

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RG / Rev	Description	U.S. EPR Assessment	FSAR Section(s)
8.19, R1	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates	Y	12.3.5
8.25, R1	Air Sampling in the Workplace	N/A-COL	N/A
8.27, 03/1981	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants	N/A-COL	N/A
8.28, 08/1981	Audible-Alarm Dosimeters	N/A-COL	N/A
8.29, R1	Instruction Concerning Risks from Occupational Radiation Exposure	N/A-COL	N/A
8.34, 07/1992	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses	N/A-COL	N/A
8.35, 06/1992	Planned Special Exposures	N/A-COL	N/A
8.36, 07/1992	Radiation Dose to the Embryo/Fetus	N/A-COL	N/A
8.38, R1	Control of Access to High and Very High Radiation Areas of Nuclear Plants	N/A-COL	N/A

Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)
Sheet 1 of 5

Issue	Description	U.S. EPR Assessment	FSAR Section(s)
TMI Requirements (10 CFR 50.34(f))			
(1)(i)	Probabilistic risk assessment (PRA)	Y	6.2.5 19.1
(1)(ii)	Evaluation of Auxiliary Feedwater System	Y	10.4.9
(1)(iii)	Reactor coolant pump (RCP) Seal Damage	Y	9.2 15.6
(1)(iv)	Evaluation of a small break loss of coolant accident (SBLOCA) caused by a stuck-open power-operated relief valve (PORV)	Y (Note: no PORV)	5.4.10 15.6.5
(2 (i)	Provide simulator model to simulate SBLOCA	N/A-COL	N/A
(2)(ii)	Improve plant procedures during construction	N/A-COL	N/A
(2)(iii)	Control Room design that reflects Human Factors	Y	18.1 18.2
(2)(iv)	Provide plant safety parameter display	Y	18.7
(2)(v)	Provide automatic indication of safety systems status	Y	7.5.2.1.1 7.1.3.1.4 Table 8.1-1 18.7
(2) (vi)	Provide capability for reactor coolant system (RCS) high point venting	Y	5.4.12
(2)(vii)	Perform radiation, shielding evaluations and incorporate equipment qualification (EQ) into the design as appropriate	Y	3.11 12.3.2
(2)(viii)	Provide capability for post-accident sampling	Y	9.3.2
(2)(x)	Provide test program for RCS pressure relief valves	Y	3.9.6 5.2.2 14.2
(2)(xi)	Provided indication of RCS pressure relief valve position	Y	5.2.2 7.1.3.1.5 7.5.2.1.1
(2)(xii)	Provide emergency feedwater (EFW) initiation and indication	Y	7.1.3.1.6 7.5.2.1.1 10.4.9

Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
(2)(xiii)	Provide pressurizer (PZR) heater power supplies	Y	Table 8.1-1
			8.3
(2)(xiv)	Provide isolation systems	Y	6.2.4
			7.1.3.1.7
(2)(xv)	Provide capability for a containment purging/ venting design that minimizes containment purge time	Y	6.2.4
			9.4.7
			11.3
(2)(xvii)	Provide sampling and instrumentation for displaying post-accident containment parameters in the control room		
	a. Containment pressure	Y	7.1.3.1.8
			7.5.2.1.1
			18.7
	b. Containment water level	Y	7.1.3.1.8
			7.5.2.1.1
			18.7
	c. Containment hydrogen concentration	Y	7.1.3.1.8
			7.5.2.1.1
			6.2.5
	d. Containment radiation intensity	Y	7.1.3.1.8
			7.5.2.1.1
			9.3.2
			11.5
			12.3
			18.7
	e. Noble gas effluents	Y	7.1.3.1.8
			7.5.2.1.1
			9.3.2
			11.5
			12.3
			18.7

Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
(2)(xviii)	Provide instrumentation and unambiguous control room indication for inadequate core cooling conditions	Y	7.1.3.1.9 7.5.2.1.1
(2)(xix)	Provide core damage instrumentation and control room indication	Y	7.1.3.1.10 7.5.2.1.1
(2)(xx)	Provide emergency power supplies for PZR components	Y	7.1.3.1.11 7.5.2.1.1 Table 8.1-1 8.3
(2)(xxv)	Provide an onsite Technical Support Center	Y	13.3
	Provide an onsite Operational Support Center	N/A-COL	N/A
(2)(xxvi)	Provide for post-accident leakage control and detection outside containment	Y	5.4.7 9.3.2
(2)(xxvii)	Provide for in-plant radiation monitoring	Y	12.3.4 11.5
(2)(xxviii)	Evaluate potential radioactivity and radiation pathways for post accident control room habitability problems	Y	6.4 9.4.1
(3)(i)	Provide administrative procedures for evaluating operating experience and lessons learned	Y	18.2
(3)(ii)	Identify safety-related systems, structures, and components (SSC) on quality assurance (QA) list	Y	3.2 17.5
(3)(iii)	Establish a QA program	Y	17.5
(3)(iv)	Provide dedicated containment ventilation penetration	Y	19.2.3.3.8
(3)(vii)	Provide a management plan for design and construction	N/A-COL	13.5
Generic Issues (NUREG-0933)			
Issue 24	Automatic ECCS Switchover to Recirculation (Medium-Priority GSI)	N/A-OTHER (Alternative design solution)	6.3 15.6.5
Issue 43	Reliability of Air Systems (High-Priority GSI)	Y	9.3.1

Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
Issue 51	Proposed Requirements for Improving the Reliability of Open-Cycle Service Water Systems (Medium-Priority GSI)	Y	9.2.1
Issue 70	Power-Operated Relief Valves (PORV) and Block Valve Reliability (Medium-Priority GSI)	Y	3.9.6 5.2.2 5.4.12 5.4.13
Issue 89	Stiff Pipe Clamps (Medium-Priority GSI)	Y	3.9.3
Issue 93	Steam Binding of Auxiliary Feedwater Pumps (High-Priority GSI)	Y	10.4.9
Issue 94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors (High-Priority GSI)	Y	5.2.2 5.3.2
Issue 99	RCS/RHR Suction Line Valve Interlock on PWRs (High-Priority GSI)	Y	5.4.7
Issue 105	Interfacing Systems LOCA at Light-Water Reactors (LWRs) (High-Priority GSI)	Y	3.12 5.4.7 6.8 9.3.4 19.2
Issue 128	Electrical Power System Reliability (High-Priority GSI)	Y	8.3 16.1
Issue 143	Availability of Chilled-Water Systems and Room Cooling (High-Priority GSI)	Y	9.2.8
Issue 153	Loss of Essential Service Water in LWRs (High-Priority GSI)	Y	9.2.1
Issue 156.6.1	Pipe Break Effects on Systems and Components (High-Priority GSI)	Y	3.5 3.6 3.8 3.9

Table 1.9-3—U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
Issue 163	Multiple Steam Generator Tube Leakage (High-Priority GSI)	Y	15.6
			19.2
Issue 185	Control of Recriticality Following Small-Break LOCAs in PWRs (High-Priority GSI)	Y (Per AREVA Topical Report ANP-10288P)	15.6.5
Issue 186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants (USI)	N/A-COL	9.1.5
Issue 191	Assessment of Debris Accumulation on PWR Sump Performance (High-Priority GSI)	Y	6.3
Issue 199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States for Existing Plants (USI)	N/A-COL	2.5
		N/A-COL	3.7

Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
I.A	Use of a Physically-Based Source Term: Position on the use of a realistic source term in design that deviates from the siting requirements in 10 CFR 100.	Y	11.1
			15.0.3
I.B	Anticipated Transient Without SCRAM (ATWS): Position on the current practices and design features to achieve a high degree of protection against an ATWS.	Y	15.8
I.C	Mid-Loop Operation: Position on design features necessary to ensure a high degree of reliability of RHR systems in PWR.	Y	5.4.7
			15.4
			19.1
			19.2
I.D	Station Blackout (SBO): Position on methods to mitigate the effects of a loss-of-offsite-power concurrent with failure of the emergency diesel generators.	Y	8.4
I.E	Fire Protection: Positions on design configuration and features the fire protection system and other management schemes to ensure safe shutdown of the reactor.	Y	9.5.1
I.F	Intersystem LOCA: Position on acceptable design practices and preventative measures to minimize the probability of an ISLOCA.	Y (Per AREVA Topical Report ANP-10264)	3.12
			5.4.7
			6.8
			9.3.4
			19.2
I.G	Hydrogen Control: Position on acceptable requirements to measure and to mitigate against the effects of hydrogen produced due to a water reaction with zirconium fuel cladding.	Y	6.2.5
I.H	Core Debris Coolability: Acceptability criteria for cooling area and quenching ability regarding corium interaction with concrete.	Y	19.2
I.I	High-Pressure Core Melt Ejection: Position on acceptable design features to prevent the event of a high-pressure core melt ejection.	Y	19.2

Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
I.J	Containment Performance: Position on acceptable conditional containment failure probabilities or other analyses to ensure a high degree of protection from the containment.	Y	19.2
I.K	Dedicated Containment Vent Penetration: Position for a dedicated vent penetration to preclude containment failure resulting from a containment overpressurization event.	Y	19.2.3.3.8
I.L	Equipment Survivability: Position on the applicability of environmental qualification and quality assurance requirements related to plant features provided only for severe-accident protection.	Y	19.2
I.M	Elimination of Operating-Basis Earthquake: Position on the applicability of the OBE in design and the possibility of decoupling the OBE and SSE in the design of safety systems.	Y	3.7 3.7.2 3.7.3 19.1
I.N	In-Service Testing of Pumps and Valves: Position on periodic testing to confirm operability of safety-related pumps and valves.	Y	3.9.6 8.3 14.2
II.A	Industry Codes and Standards: Position on use of recently developed or modified design codes and industry standards in ALWR designs that have not been reviewed for acceptability by the NRC.	Y	3.1
II.B	Electrical Distribution: Position on acceptable practices relating to the electrical distribution of safety- and non-safety loads.	Y	8.2
II.C	Seismic Hazard Curves and Design Parameters: Position on use of proposed generic bounding seismic hazard curves and performance of seismic PRA.	Y	19.1
II.D	Leak-Before-Break: Position on use of leak-before-break concept.	Y	3.6.2 3.6.3
II. E	Classification of Main Steam Lines in BWRs: Position on the staff's defined approach for seismic classification of the main steam line in both evolutionary and passive BWRs.	N/A-BWR	N/A

**Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
II.F	Tornado Design Basis: Position on the maximum tornado wind speed to be used for a design basis tornado.	Y	3.3
II.G	Containment Bypass: Position on ALWR design against containment bypass. Specifically, failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling heat exchanger tubes in large pools of water outside containment.	Y	6.2.1 6.3 15.0.3 19.2
II.H	Containment Leak Rate Testing: Position on testing duration for Type C leak rate testing (prior to rule change).	Y	6.2.6
II.I	Post-Accident Sampling System: Position on the required capability to analyze dissolved hydrogen, oxygen, and chloride in accordance with applicable regulations.	Y	9.3.2
II.J	Level of Detail: Position on a design certification submittal with depth of detail similar to that in an FSAR.	Y	All FSAR sections
II.K	Prototyping: No guidance provided; information only	N/A-INFO	N/A
II.L	ITAAC: Position on providing ITAAC to demonstrate that a nuclear power plant referencing a certified design is built and operates consistent with the design certification.	ITAAC	Tier 1
II.M	Reliability Assurance Program: Position on providing a description of purpose, scope, objectives, and implementation of a design reliability assurance program.	Y	17.4
II.N	Site-Specific PRA and Analysis of External Events: Position on the inclusion of external event analysis beyond the design basis that needs to be addressed as part of the plant PRA during the design certification review.	Y	19.1
II.O	Severe Accident Mitigation Design Alternatives (SAMDA): Position on the consideration of SAMDA as part of the final design approval/design certification of an advanced reactor.	Y	19.2

Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
II.P	Generic Rulemaking Related to Design Certification: No guidance provided; information only.	N/A-INFO	N/A
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems: Position on the use of defense-in-depth and diversity of instrumentation and control systems as part of the final design approval/design certification of an advanced reactor.	Y (Per AREVA Technical Report ANP-10304)	7.1.3.3.1 7.8
II.R	SG Tube Ruptures: Position on requiring that analysis of multiple SGTRs of 2 to 5 SG tubes be included in the application for design certification of passive ALWRs.	Y	15.6 19.2
II.S	PRA Beyond Design Certification: Position on requiring conversion of the design certification PRA into a plant-specific PRA	N/A-COL	19.0
II.T	Control Room Annunciator (Alarm) Reliability: Position on recommending that additional requirements for ALWR alarm systems are necessary to minimize the problems experienced by operating nuclear power plants, such	Y (Per AREVA Technical Report ANP-10304)	7.1.3.3.2
III.A	Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Designs: Position on the proposed staff approach for resolving the regulatory treatment of the active non-safety systems in passive ALWRs.	N/A-PAS	N/A
III.B	Definition of Passive Failure: Position on the staff redefining some passive failures of components as active failures (i.e., check valves) to cause valves to be evaluated in a much more stringent manner than in previous licensing review	N/A-PAS	N/A
III.C	SBWR Stability	N/A-PAS N/A-BWR	N/A
III.D	Safe Shutdown Requirements: Position on using non-safety grade active cooling systems to bring a reactor to cold shutdown since non-safety RHR systems to not comply with the guidance of 1.139 or branch technical position 5-1	N/A-PAS	N/A

Table 1.9-4—U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)
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Issue	Description	U.S. EPR Assessment	FSAR Section(s)
III.E	Control Room Habitability Position on appropriate analytical methods (i.e., dose limits and accident duration) to be used in determining the acceptability criteria for control room habitably in accordance with regulatory standards.	Y	15.0.3
III.F	Radionuclide Attenuation: Position on fission product removal processes inside containment by natural effects and holdup by the secondary building and piping systems in addition to commission position on containment spray systems for passive ALWRs.	Y	6.5.3 15.0.3
III.G	Simplification of Offsite Emergency Planning: Position on simplifying off-site emergency planning of passive designs due to the estimated low probability of core damage of such designs.	N/A-PAS	N/A
III.H	Role of the Passive Plant Control Room Operator: Commission position on sufficient man-in-the-loop testing and evaluation be performed and that a fully functional integrated control room prototype is necessary for passive plant control room designs to demonstrate that functions and tasks are integrated properly into the man/machine interface decisions.	Y	18.2 18.3 18.7