ANP-10292
Revision 0

U.S. EPR Conformance with Standard Review Plan (NUREG-0800) Technical Report

December 2007

AREVA NP Inc.

Abstract for ANP-10292

U.S. EPR Conformance with Standard Review Plan For Design Certification of the U.S. EPR

U.S. Evolutionary Pressurized Reactor (EPR) design certification requires demonstration of conformance with NUREG-0800. This report was prepared to meet the requirements of the aforementioned standard. This report is submitted as a supplement to the U.S. EPR Final Safety Analysis Report (FSAR), Tier 2, Chapter 1; Section 1.9.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
0	All	Initial Issuance

Contents

List of Tables

Nomenclature

Acronym	Definition
AAC	Alternate Alternating Current
ABB-CE	Asea Brown Boveri-Combustion Engineering
ABWR	Advanced Boiling Water Reactor
ACR	Acceptance Criteria Requirement
AC	Alternating Current
AFWS	Auxiliary Feedwater System
ALARA	As Low As Is Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI/AISC	American National Standards Institute/ American Institute of Steel Construction
AOO	Anticipated Operational Occurrence
ASCE/SEI	American Society of Civil Engineers/ Structural Engineering Institute
ASD	Allowable Stress Design
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transients Without Scram
BE	Best Estimate
BOP	Balance of Plant
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CAS	Central Alarm System
CAV	Cumulative Absolute Velocity
CDM	Certified Design Material
CEUS	Central and Eastern United States
CFR	Code of Federal Regulations
CHFR	Critical Heat Flux Ratio
CILRT	Containment Integrated Leakage Rate Test
CIV	Containment Isolation valves
COL	Combined License
COMS	Cold Overpressure Mitigation System
CP	Construction Permit
CPR	Critical Power Ratio

Acronym	Definition
CRAVS	Control Room Area Ventilation System
CRDS	Control Rod Drive System
CRF	Carryout Rate Fraction
CSDRS	Certified Seismic Design Response Spectra
C_v	Charpy V-notch
CVCS	Chemical and Volume Control System
DAC	Derived Air Concentration (SRP Chapter 12) Design Acceptance Criteria (SRP Chapter 14)
DC	Design Certification
DFSS	Diesel Fuel Storage Stucture
DLF	Dynamic Load Factor
DNB	Departure Of Nucleate Boiling
DNBR	Departure Of Nucleate Boiling Ration
DNS	Direct Numerical Simulation
DWT	Drop Weight Test
EAB	Exclusion Area Boundary
ECCS	Emergency Core cooling System
EDECAIES	Emergency Diesel Engine Combustion Air Intake and Exhaust System
EDECWS	Emergency Diesel Engine Cooling Water System
EDEFSS	Emergency Diesel Engine Fuel Oil Storage and Transfer System
EDELS	Emergency Diesel Engine Lubrication System
EDESS	Emergency Diesel Engine Starting System
EDG	Emergency Diesel Generator
EDSFI	Electrical Distribution System Functional Inspection
EFDS	Equipment and Floor Drainage System
EPA	Environmental Protection Agency
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EPZ	Emergency Planning Zone
EQ	Environmental Qualification
ERDS	Emergency Response Data System
ERF	Emergency Response Facility
ESP	Early Site Permit
ETE	Evacuation Time Estimate
FATT	Fracture Appearance Transition Temperature

Acronym	Definition
FEMA	Federal Emergency Management Agency
FIRS	Foundation Input Response Spectra
FRF	Frequency Response Function
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report
GDC	General Design Criteria
GL	Generic Letter
GMRS	Ground Motion Response Spectrum
GRS	Gaseous Radwaste System
GWMS	Gaseous Waste Management System
HA	Human Action
HED	Human Engineering Discrepancy
HEPA	High-Efficiency Particulate Air
HFE	Human Factors Engineering
HP	High Pressure
HPCI	High Pressure Coolant Injection
HRA	Human Reliability Analysis
HSI	Human System Interface
HVAC	Heating, Ventilation And Air Conditioning
I&C	Instrumentation and Controls
IEEE	Institute of Electrical and Electronics Engineers
IESNA	Illuminating Engineering Society of North America
IGSCC	Intergranular Stress Corrosion Cracking
ISI	Inservice Inspection
ISM	Independent Support Motion
ITAAC	Inspections, Tests, Analyses And Acceptance Criteria
LB	Lower Bound
LBB	Leak-Before-Break
LCO	Limiting Condition for Operation
LCS	Local Control Station
LES	Large Eddy Simulation
LHGR	Linear Heat Generation Rate
LLRT	Local Leakage Rate Test
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LP	Low Pressure

Acronym	Definition
LPZ	Low Population Zone
LTOP	Low-Temperature Overpressure Protection
LWMS	Liquid Waste Management System
LWR	Light Water Reactor
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSIVLCS	Main Steam Isolation Valve
MSL	Main Steam Line
MTC	Moderator Temperature Coefficient
NDRC	National Defense Research Council
NDT	Nil-Ductility Transition
NOAA	National Oceanic and Atmospheric Administration
non-DBA	Non-Design Basis Accident
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
NSSS	Nuclear Steam Supply System
NWS	National Weather Service
OBE	Operating Basis Earthquake
OER	Operating Experience Review
OL	Operating License
ORE	Occupational Radiation Exposure
OSC	Operational Support Center
PA	Protected Area
PCI	Pellet/Cladding Interaction
PCMI	Pellet/Cladding Mechanical Interaction
PERMISS	Process and Effluent Radiological Monitoring Instrumentation And Sampling System
PGA	Peak Ground Acceleration
PMWP	Probable Maximum Winter Precipitation
PORV	Power-Operated Relief Valve
POV	Power-Operated Valve
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report

Acronym	Definition
PSD	Power Spectral Density
PSS	Process Sampling System
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAP	Reliability Assurance Program
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RIA	Reactivity Initiated Accident
RPV	Reactor Pressure Vessel
RRS	Required Response Spectrum
RSP	Remote Shutdown Panel
RSS	Remote Shutdown Station
RTNSS	Regulatory Treatment Of Non-Safety System
RV	Reactor Vessel
SAC	Specific Standard Review Plan Acceptance Criteria
SAFDL	Specified Acceptable Fuel Design Limits
SAR	Safety Analysis Report
SAS	Secondary Alarm System
SBLOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SER	Safety Evaluation Report
SMT	Scale Model Test
SOV	Solenoid-Operated Valve
SPDS	Safety Parameter Display System
SQR	Seismic Qualification Report
SRI	Stanford Research Institute
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SRSS	Square Root of the Sum or Squares
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
STS	Standard Technical Specifications

Acronym	Definition
SUFWS	Safety Related Start Up Feedwater System
SWMS	Solid Waste Management System
TBS	Turbine Bypass System
TGSS	Turbine Gland Sealing System
TMI	Three Mile Island
TRS	Test Response Spectrum
TS	Technical Specifications
TSC	Technical Support Center
U.S. EPR	United States Evolutionary Power Reactor
UB	Upper Bound
UHRS	Uniform hazard response spectrum
UHS	Ultimate Heat Sink
URS	Uniform Response Spectrum
USGS	United States Geologic Survey
USM	Uniform Support Motion
V&V	Verification and Validation
ZPA	Zero Period Acceleration
χ/Q Value	Atmospheric Dispersion Factor

1.0 CONFORMANCE WITH REGULATORY CRITERIA

This technical report provides a guide to U.S. EPR conformance with Standard Review Plan (SRP) (NUREG-0800) regulatory criteria. Conformance is assessed to regulatory criteria in effect six months before the anticipated docket date of the U.S. EPR design certification application.

Table 1–1, U.S. EPR Conformance Table Legend, defines the codes used to indicate conformance determination notations in the “U.S. EPR Assessment” column of Table 1-2, U.S. EPR Conformance with Standard Review Plan (NUREG-0800). Multiple assessment codes from Table 1-1 are identified, as necessary, for individual regulatory criteria in Table 1–2 to address U.S. EPR conformance for these cases:

- NRC guidance expectations to address specific conformance with a regulatory criterion in multiple 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.

A combined license (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.

1.1 *Conformance with the Standard Review Plan*

The SRP conformance assessment in accordance with 10 CFR 52.47(a)(9) is provided in Table 1–2. Compliance is evaluated against technically relevant portions of NUREG-0800 current six months prior to the application for design certification. Table 1-2 contains a listing of SRP sections, Branch Technical Positions (BTPs) and the associated SRP section or BTP revision. For those SRP sections technically relevant to the U.S. EPR design certification, Table 1–2 also lists individual SRP acceptance criteria requirements and specific acceptance criteria from each SRP section. For each SRP and BTP criterion listed in Table 1–2, 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 so noted. Because BTPs provide specific or clarifying guidance for individual SRP specific acceptance criteria, conformance with BTP criteria are addressed in Table 1–2 by referencing the U.S. EPR conformance assessment notation(s) of the governing specific SRP acceptance criteria.

Table 1–1—U.S. EPR Conformance Table Legend

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, tests, analyses 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 boiling water reactor (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 combined license (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.

Table 1–1—U.S. EPR Conformance Table Legend

Assessment Code	Description
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)."
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.

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Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 1 Introduction and General Description of Plant			
SRP Criterion	**Description** **(AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 1.0	**Introduction and Interfaces (03/2007)**		
1.0-AC-01	**10 CFR 50.33**, **10 CFR 50.34**, **10 CFR 52.16**, **10 CFR 52.17**, **10 CFR 52.46**, **10 CFR 52.47**, **10 CFR 52.77** and **10 CFR 52.79** as they relate to general introductory matters.	Y	1.1
1.0-AC-02	Interfaces with standard designs		
	A. **10 CFR 52.47(a)(24)** requires the DC application to contain a representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR and to permit assessment of the adequacy of the interface requirements in paragraph **(a)(25)** of **10 CFR 52.47**.	Y	1.8 14.3.4
	B. **10 CFR 52.47(a)(25)** requires the DC FSAR to contain the interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the FSAR.	Y	1.8 14.3.4
	C. **10 CFR 52.47(a)(26)** requires the DC FSAR to contain justification that compliance with the interface requirements of paragraph **(a)(25)** of **10 CFR 52.47** is verifiable through inspections, tests, or analyses. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by paragraph **(b)(2)** of **10 CFR 52.47**.	Y	14.3 Tier 1, Chp 4
	D. 10 **CFR 52.79(d)(2)** requires that for a COL referencing a standard design certification, the FSAR demonstrate that the interface requirements established for the design under **10 CFR 52.47** have	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 1 Introduction and General Description of Plant			
SRP Criterion	**Description** **(AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	been met.		
1.0-AC-03	**10 CFR 50.34(h)**, **10 CFR 52.17(a)(1)(xii)**, **10 CFR 52.47(a)(9)**, and **10 CFR 52.79(a)(41)** as they relate to an evaluation of the application against the applicable NRC review guidance in effect 6 months before the docket date of the application.	Y	1.9.2
1.0-AC-04	**10 CFR 52.47(a)(21)** and **10 CFR 52.79(a)(20)** as they relate to proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design.	Y	1.9.3
1.0-AC-05	**10 CFR 50.34(f)**, **10 CFR 52.47(a)(8)** and **10 CFR 52.79(a)(17)** as they relate to compliance with technically relevant positions of the Three Mile Island requirements.	Y	1.9.3
1.0-AC-06	**10 CFR 52.47(a)(22)** and **10 CFR 52.79(a)(37)** as they relate to the information necessary to demonstrate how operating experience insights have been incorporated into the plant design.	Y	1.9.4
1.0-AC-07	**10 CFR 50.43(e)** as it relates to requirements for approval of applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions.	N/A-OTHER	1.1
1.0-AC-08	**10 CFR 52.79(a)(31)** regarding nuclear power plants to be operated on	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 1 Introduction and General Description of Plant			
SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	multi-unit sites, as it relates to an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multi-unit sites.		
1.0-SAC-01	There are no specific SRP acceptance criteria associated with these general requirements.	N/A-INFO	N/A
1.0-SAC-02	For the regulatory considerations, acceptance is based on addressing the regulatory requirements as discussed within this FSAR section or within the referenced FSAR section. The SRP acceptance criteria associated with the referenced section will be reviewed within the context of that review.	N/A-INFO	N/A
1.0-SAC-03	For performance of new safety features, the information is sufficient to provide a reasonable assurance that (1) these new safety features will perform as predicted in the applicant's FSAR, (2) the effects of system interactions are acceptable, and (3) the applicant provides sufficient data to validate analytical codes. The design qualification testing requirements may be met with either separate effects or integral system tests; prototype tests; or a combination of tests, analyses, and operating experience.	N/A-OTHER	1.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2
Site Characteristics

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 2.0	**Site Characteristics and Site Parameters (03/2007)**		
2.0-AC-01	**10 CFR 52.17** describes the technical contents of an ESP application.	N/A-ESP	N/A
2.0-AC-02	**10 CFR 52.47(a)(1**) requires a DC applicant to provide site parameters postulated for the design.	Y	2.0
2.0-AC-03	**10 CFR 52.79(a)(1)(i) - (vi)** provides the site-related contents of a COL application.	N/A-COL	2.1
2.0-AC-04	**10 CFR 52.79(b)** for a COL referencing an ESP as it relates to information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the ESP.	N/A-ESP	N/A
2.0-AC-05	**10 CFR 52.79(d)(1)** for a COL referencing a DC as it relates to information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the DC.	N/A-COL	2.0
2.0-AC-06	**10 CFR Part 100** as it relates to the siting factors and criteria for determining an acceptable site.	N/A-COL	2.0
2.0-SAC-01	For ESP, DC, and COL applications, the acceptance criteria associated with specific site characteristics/parameters and site-related design characteristics/parameters are contained in the related **SRP Chapter 2** or other referenced SRP sections.	Y	2.0
2.0-SAC-02	For a COL application referencing an ESP, acceptance is based on the applicant's demonstration that the design of the facility falls within the site characteristics and site-related design parameters specified in the ESP. If the final safety analysis report does not demonstrate that the design of the facility falls within the site characteristics and design parameters, the application shall include a request for a variance from the ESP that complies with the requirements of **10 CFR 52.39** and **10 CFR 52.93.**	N/A-ESP N/A-COL	N/A N/A
2.0-SAC-03	For a COL application referencing a DC, acceptance is based on the applicant's demonstration that the characteristics of the site fall within the site parameters of the	N/A-COL	2.0

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2
Site Characteristics

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	certified design. If the actual site characteristics do not fall within the certified standard design site parameters, the COL applicant provides sufficient justification (e.g., by request for exemption or amendment from the DC) that the proposed facility is acceptable at the proposed site.		
2.0-SAC-04	For a COL application referencing an ESP and a DC, acceptance is based on the applicant's demonstration that the site characteristics and site-related design parameters specified in the ESP fall within the site parameters and design characteristics specified in the DC. If the actual site characteristics do not fall within the certified standard design site parameters, the COL applicant provides sufficient justification (e.g., by request for exemption or amendment from the DC, or request for a variance from the ESP) that the proposed facility is acceptable at the proposed site.	N/A-ESP N/A-COL	N/A N/A
2.0-SAC-05	For a COL application referencing neither an ESP nor a DC, acceptance is based on the applicant's identification of the complete set of site characteristics and site-related design characteristics needed to enable the staff to reach a conclusion on all safety matters related to siting.	N/A-COL	N/A
SRP 2.1.1	**Site Location and Description (R3, 03/2007)**	N/A COL	2.1.1
SRP 2.1.2	**Exclusion Area Authority and Control (R3, 03/2007)**	N/A-COL	2.1.2
SRP 2.1.3	**Population Distribution (R3, 03/2007)**	N/A-COL	2.1.3
SRP 2.2.1 - 2.2.2	**Identification of Potential Hazards in Site Vicinity (R3, 03/2007)**	N/A-COL	2.2.1 2.2.2
SRP 2.2.3	**Evaluation of Potential Accidents (R3, 03/2007)**	N/A-COL	2.2.3
SRP 2.3.1	**Regional Climatology (R3, 03/2007)**		
2.3.1-AC-01	**10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2**, "Design Bases for Protection Against Natural Phenomena," as it relates to consideration of the most severe	Y	2.3.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2
Site Characteristics

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.		
2.3.1-AC-02	**10 CFR Part 50, Appendix A, GDC 4**, "Environmental and Dynamic Effects Design Bases," as it relates to information on tornadoes that could generate missiles.	Y	2.3.1
2.3.1-AC-03	**10 CFR 52.47(a)(1)** requires a DC applicant to provide site parameters postulated for the design.	Y	2.0
			2.3.1
2.3.1-AC-04	**10 CFR 52.79(a)(iii)** requires a COL applicant to identify the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and time in which the historical data have been accumulated.	N/A-COL	2.3.1
2.3.1-AC-05	**10 CFR Part 100, §100.10(c)(2), §100.20(c)(2), and §100.21(d)** with respect to the consideration given to the regional meteorological characteristics of the site.	N/A-COL	2.3.1
2.3.1-AC-06	For **ESP** applications, GDC are not applicable. The **GDC** 2 requirement to identify climatic site characteristics that consider the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated is specifically identified in **10 CFR 52.17(a)(1)(vi)**.	N/A-ESP	N/A
2.3.1-SAC-01	The description of the general climate of the region should be based on standard climatic summaries compiled by NOAA (e.g., References 5, 6). Consideration of the relationships between regional synoptic-scale atmospheric processes and local (site) meteorological conditions should be based on appropriate meteorological data (e.g., References 6, 7).	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
2.3.1-SAC-02	Data on severe weather phenomena should be based on standard meteorological records from nearby representative National Weather Service (NWS), military, or other stations recognized as standard installations that have long periods of data on record (e.g., References. 6, 7, 8). The applicability of these data to represent site conditions during the expected period of reactor operation should be substantiated.	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1
2.3.1-SAC-03	The tornado parameters should be based on **Regulatory Guide 1.76** (Reference 9). Alternatively, an applicant may specify any tornado parameters that are appropriately justified, provided that a technical evaluation of site-specific data is conducted. Any deviations from **Regulatory Guide 1.76** should be identified by the applicant.	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1
2.3.1-SAC-04	The basic (straight-line) 100-year return period 3-second gust wind speed should be based on appropriate standards, with suitable corrections for local conditions (e.g., References 10, 11).	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1
2.3.1-SAC-05	In accordance with **Regulatory Guide 1.27** (Reference 12), the UHS meteorological data that would result in the maximum evaporation and drift loss of water and minimum water cooling should be based on long-period regional records that represent site conditions. If applicable, the potential for water freezing in the UHS water storage facility should also be analyzed. The maximum accumulated degree-days below freezing recorded in the site region during the winter (or during the worst-case freezing spell in warmer climates)	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific	2.3.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	may be a reasonable conservative site characteristic for evaluating the potential for water freezing in a UHS water storage facility. Suitable information should be compiled from at least 30 years of meteorological data found in databases for nearby representative locations (e.g., References 13, 14, 15). The bases and procedures used to select critical meteorological data should be provided and justified.	Climatology)	
2.3.1-SAC-06	Consistent with the staff's branch position on winter precipitation loads (Reference 16), the winter precipitation loads to be included in the combination of normal live loads to be considered in the design of a nuclear power plant that might be constructed on the proposed site should be based on the weight of the 100-year snowpack or snowfall, whichever is greater, recorded at ground level. Likewise, the winter precipitation loads to be included in the combination of extreme live loads to be considered in the design of a nuclear power plant that might be constructed on the proposed site should be based on the weight of the 100-year snowpack at ground level plus the weight of the 48-hour PMWP at ground level for the month corresponding to the selected snowpack. Depending on the location of the site, the 48-hour PMWP may not necessarily be in the form of frozen precipitation. A CP, OL, or COL applicant may choose and justify an alternative method for defining the extreme winter precipitation load by demonstrating that the 48-hour PMWP could neither fall nor remain on the top of the snowpack and/or building roofs. The weight of the 100-year return period snowpack should be based on data recorded at nearby representative climatic stations (e.g., Reference 17) or obtained from appropriate standards with suitable corrections for local conditions (e.g., References 10, 11). For the purposes of determining the extreme winter precipitation load, the 48-hour PMWP is defined as the theoretically greatest depth of precipitation for a 48-hour period that is physically possible over a 25.9-square-kilometer (10-square-mile) area at a particular geographical location during those months with the historically highest snowpacks. The weight of the 48-hour PMWP should be determined in accordance with reports published by NOAA's Hydrometeorological Design Studies Center (e.g.,	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	References 18–22).		
2.3.1-SAC-07	Ambient temperature and humidity statistics should be derived from data recorded at nearby representative climatic stations (e.g., Reference 23) or obtained from appropriate standards with suitable corrections for local conditions (e.g., Reference 10). Reference 23 provides a method for estimating 100-year return period extreme temperature values as a function of annual extreme temperature values.	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1
2.3.1-SAC-08	High air pollution potential information should be based on U.S. Environmental Protection Agency (EPA) studies (e.g., References 24, 25).	Y (Design Parameters)	2.3.1
		N/A-COL (Site-Specific Climatology)	2.3.1
2.3.1-SAC-09	All other meteorological and air quality conditions identified by the applicant as climate site characteristics for ESP applications or used as design and operating bases for CP, OL, or COL applications should be documented and substantiated.	N/A-ESP	N/A
		N/A-COL	N/A
SRP 2.3.2	**Local Meteorology (R3, 03/2007)**	N/A-COL	2.3.2
SRP 2.3.3	**Onsite Meteorological Measurements Programs (R3, 03/2007)**	N/A-COL	2.3.3
SRP 2.3.4	**Short-term Atmospheric Dispersion Estimates for Accident Releases (R3, 03/2007)**		
2.3.4-AC-01	For CP and OL applications on or after January 10, 1997, **10 CFR 50.34(a)(1)(ii)(d)** with respect to an assessment of the plant design features intended to mitigate the radiological consequences of accidents, which includes consideration of site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.	N/A-CP/OL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
2.3.4-AC-02	For CP, OL, DC, and COL applications, **10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC 19)**, "Control Room," with respect to the meteorological considerations used to evaluate the personnel exposures inside the control room during radiological and airborne hazardous material accident conditions. GDC are not applicable for ESP applications.	Y	2.3.4
2.3.4-AC-03	For ESP applications, **10 CFR 52.17(a)(1)(ix)** with respect to a safety assessment of the site, including consideration of major SSCs of the facility and site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.	N/A-ESP	N/A
2.3.4-AC-04	For DC applications, **10 CFR 52.47(a)(2)(iv)** with respect to an assessment of the plant design features intended to mitigate the radiological consequences of accidents, which includes consideration of postulated site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.	Y	2.3.4
2.3.4-AC-05	For COL applications, **10 CFR 52.79(a)(1)(vi)** with respect to a safety assessment of the site, including consideration of major SSCs of the facility and site meteorology, to evaluate the offsite radiological consequences at the EAB and LPZ.	N/A-COL	2.3.4
2.3.4-AC-06	For reactor applications before January 10, 1997, **10 CFR 100.11(a)**, with respect to the meteorological considerations used in the evaluation to determine an acceptable EAB and LPZ.	N/A-CP/OL	N/A
2.3.4-AC-07	For reactor applications on or after January 10, 1997, **10 CFR 100.21(c)(2)**, with respect to the atmospheric dispersion characteristics used in the evaluation of EAB and LPZ radiological dose consequences for postulated accidents.	Y	2.3.4
2.3.4-SAC-01	A description of the atmospheric dispersion models used to calculate χ/Q values for accidental releases of radioactive and hazardous materials to the atmosphere. The models should be documented in detail and substantiated within the limits of the model so that the staff can evaluate their appropriateness of use with regards to release	Y	2.3.4

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CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	characteristics, plant configuration, plume density, meteorological conditions, and site topography.		
2.3.4-SAC-02	Meteorological data used for the evaluation (as input to the dispersion models) which represent annual cycles of hourly values of wind direction, wind speed, and atmospheric stability for each mode of accidental release. Any dispersion estimates should be calculated from the most representative meteorological data available for the site. Guidance on appropriate onsite meteorological data is provided in **Regulatory Guide 1.23**. This information is also reviewed in **SRP Section 2.3.3**.	Y	2.3.4
2.3.4-SAC-03	A discussion of atmospheric diffusion parameters, such as lateral and vertical plume spread (σ_y and σ_z) as a function of distance, topography, and atmospheric conditions, should be related to measured meteorological data. The methodology for establishing these relationships should be appropriate for estimating the consequences of accidents within the range of distances which are of interest with respect to site characteristics and established regulatory criteria.	Y	2.3.4
2.3.4-SAC-04	Hourly cumulative frequency distributions of χ/Q values from the effluent release point(s) to the EAB and LPZ should be constructed to describe the probabilities of these χ/Q values being exceeded. All cumulative frequency distributions of χ/Q values should be presented for appropriate distances (e.g., the EAB distance and the outer boundary of the LPZ) and time periods as specified in **Section 2.3.4.2 of Regulatory Guide 1.70**, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and **Section 2.3.4.2 of RG 1.206**, “Combined License Applications for Nuclear Power Plants (LWR Edition).” The methods for generating these distributions should be adequately described. Guidance for calculating EAB and LPZ atmospheric dispersion factors is provided in **Regulatory Guide 1.145**.	Y	2.3.4
2.3.4-SAC-05	Atmospheric dispersion factors used for the assessment of consequences related to atmospheric radioactive releases to the control room for design basis, other accidents,	Y	2.3.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and for onsite and offsite releases of hazardous airborne materials should be provided. Guidance for calculating control room χ/Q values for radiological releases and hazardous material releases is provided in **Regulatory Guide 1.194** and **Regulatory Guide 1.78**, respectively.		
2.3.4-SAC-06	For control room habitability analysis, a site plan drawn to scale should be included showing true North and potential atmospheric accident release pathways, control room intake, and unfiltered in leakage pathways.	Y	2.3.4
SRP 2.3.5	**Long-Term Atmospheric Dispersion Estimates for Routine Releases (R3, 03/2007)**	N/A-COL	2.3.5
SRP 2.4.1	**Hydrologic Description (R3, 03/2007)**	N/A-COL	2.4.1
SRP 2.4.2	**Floods (R4, 03/2007)**	N/A-COL	2.4.2
SRP 2.4.3	**Probable Maximum Flood (PMF) on Streams and Rivers (R4, 03/2007)**	N/A-COL	2.4.3
SRP 2.4.4	**Potential Dam Failures (R3, 03/2007)**	N/A-COL	2.4.4
SRP 2.4.5	**Probable Maximum Surge and Seiche Flooding (R3, 03/2007)**	N/A-COL	2.4.5
SRP 2.4.6	**Probable Maximum Tsunami Flooding (R3, 03/2007)**	N/A-COL	2.4.6
SRP 2.4.7	**Ice Effects (R3, 03/2007)**	N/A-COL	2.4.7 (Refer to 9.2.5 relative to UHS)
SRP 2.4.8	**Cooling Water Canals and Reservoirs (R3, 03/2007)**	N/A-COL	2.4.8 (Refer to 9.2.5 relative to UHS)
SRP 2.4.9	**Channel Diversions (R3, 03/2007)**	N/A-COL	2.4.9

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 2.4.10	**Flooding Protection Requirements (R3, 03/2007)**	N/A-COL	2.4.10
SRP 2.4.11	**Low Water Considerations (R3, 03/2007)**	N/A-COL	2.4.11
SRP 2.4.12	**Groundwater (R3, 03/2007)**	N/A-COL	2.4.12
SRP 2.4.13	**Accidental Releases of Liquid Effluents in Ground and Surface Waters (R3, 03/2007)**	N/A-COL	2.4.13
SRP 2.4.14	**Technical Specifications and Emergency Operation Requirements (R3, 03/2007)**	N/A-COL	2.4.14
SRP 2.5.1	**Basic Geologic and Seismic Information (R4, 03/2007)**	N/A-COL	2.5.1
SRP 2.5.2	**Vibratory Ground Motion (R4, 3/2007)**	N/A-COL	2.5.2
SRP 2.5.3	**Surface Faulting (R4, 03/2007)**	N/A-COL	2.5.3
SRP 2.5.4	**Stability of Subsurface Materials and Foundations (R3, 03/2007)**		
2.5.4-AC-01	**10 CFR 50.55a -** Codes and Standards requires that structures, systems, and components be designed, fabricated, erected, constructed, tested and inspected in accordance with the requirement of applicable codes and standards commensurate with the importance of the safety function to be performed.	Y	2.5.4
2.5.4-AC-02	**10 CFR Part 50, Appendix A:**		
	a) **General Design Criterion 1 (GDC 1),** "Quality Standards and Records," requires that structures, systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. It also requires that appropriate records of the	Y (for Design, Fabricate, Erect, & Test of SSCs)	2.5.4

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	design, fabrication, erection, and testing of structures, systems, and components important to safety be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	N/A-COL (for Maintain Records)	2.5.4
	b) **General Design Criterion 2 (GDC 2)**, "Design Bases for Protection Against Natural Phenomena," as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.	N/A-COL	2.5.4
	c) **General Design Criterion 44 (GDC 44),** "Cooling Water," requires that a system be provided with the safety function of transferring the combined heat load from structures, systems, and components important to safety to an ultimate heat sink under normal operating and accidental conditions	N/A-COL	2.5.4
	d) For ESP applications, GDC are not applicable. However, the **GDC 2** requirement to identify site characteristics that consider the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated is specifically identified in **10 CFR 52.17(a)(1)(vi)(vi).**	N/A-ESP	N/A
2.5.4-AC-03	**10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants"**, establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.	Y	3.2
2.5.4-AC-04	**10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants,"** as it applies to the design of nuclear power plant structures, systems and	Y	2.5.4

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	components important to safety to withstand the effects of earthquakes.		
2.5.4-AC-05	**10 CFR Part 100, "Reactor Site Criteria,"** provides the criteria which guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.	N/A-COL	2.1 2.5.4
2.5.4-AC-06	**10 CFR 100.23, "Geologic and Seismic Criteria,"** provides the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants.	N/A-COL	2.5.4
2.5.4-SAC-01	<u>Geologic Features</u>. In meeting the requirements of **10 CFR Parts 50 and 100**, the section defining geologic features is acceptable if the discussions, maps, and profiles of the site stratigraphy, lithology, structural geology, geologic history, and engineering geology are complete and are supported by site investigations sufficiently detailed to obtain an unambiguous representation of the geology. The information must be presented in this subsection or cross-referenced to the appropriate subsection in Section 2.5.1 of the SAR. Geologic features are evaluated by conducting an independent literature search and comparing these results with the information included in the applicant's SAR. References used in reviewing this subsection include published or unpublished reports, maps, geophysical data, construction records, etc., by the USGS, other Federal agencies, State agencies, and private companies. In conjunction with the literature search, the staff and its advisors review the geological investigations conducted by the applicant. Using the references listed at the end of this section and other sources, the following questions are considered in detail: 1. Are the exploratory techniques used by the site investigator representative of the present state-of-the-art? Do the samples represent the in situ soil conditions? 2. Do the applicant's investigations provide adequate coverage of the site area and in	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

<table>
<tr><th colspan="4">CHAPTER 2
Site Characteristics</th></tr>
<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>sufficient detail to define the specific subsurface conditions with a high degree of confidence?
3. Have all areas or zones of actual or potential surface or subsurface subsidence, uplift or collapse, deformation, alternation, solution cavities or structural weakness, unrelieved stresses in bedrock, or rocks or soils that might be unstable because of their physical or chemical properties been identified and adequately evaluated?</td><td></td><td></td></tr>
<tr><td>2.5.4-SAC-02</td><td><u>Properties of Subsurface Materials</u>.
In meeting the requirements of 10 CFR Parts 50 and 100, the description of properties of underlying materials is considered acceptable if state-of-the-art methods are used to determine the static and dynamic engineering properties of all foundation soils and rocks in the site area. These methods are described, for example, in geotechnical journals published by the American Society of Civil Engineers (Refs. 14, 22, and 23), applicable standards published by the American Society for Testing and Materials (Ref. 15), publications of the Institution of Civil Engineers (Ref. 15), and various research reports prepared by universities (Ref. 17). The properties of foundation material must be supported by field (Refs. 19 and 20) and laboratory (Ref. 21) test records.
Normally, a complete field investigation and sampling program must be performed to define the occurrence and properties of underlying materials at a given site (Ref. 18). Summary tables must be provided which catalog the important test results; test results should be plotted when appropriate. Also, a detailed discussion of laboratory sample preparation must be given when applicable. For critical laboratory tests, full details must be given, e.g., how saturation of the sample was determined and maintained during testing, transported and how the pore pressures were monitored during the experiment.
The applicant should provide a detailed and quantitative discussion of the criteria used to determine that the samples were properly taken, and tested in sufficient number to define all the critical soil parameters for the site, together with their potential variability. For sites</td><td>N/A-COL</td><td>2.5.4</td></tr>
</table>

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	that are underlain by saturated soils and sensitive clays, it should be shown that all zones which could become unstable due to liquefaction or strain-softening phenomena have been adequately sampled and tested. The relative density of the soils at the site should be determined. The applicant must also show that the consolidation behavior of the soils as well as their static and dynamic strength have been adequately defined. The discussion should explain how the developed data is used in the safety analyses, how the test data is analyzed to generate appropriate design parameters and present a table indicating the value of the parameters used in the analyses. Properties of underlying materials are evaluated to determine whether or not the investigations performed (including laboratory and field testing) were sufficient to justify the soil and rock properties used in the foundation analyses. To determine whether sufficient investigations were performed, the staff carefully reviews the criteria developed and used by the applicant in laying out the boring, sampling and testing program and evaluates the effectiveness of the program in defining the specific foundation conditions at the site to ensure that all critical conditions have been adequately sampled and tested. If suitable criteria have not been developed and used by the applicant, the staff develops appropriate criteria, using **Regulatory Guide 1.132** and the data given in the SAR, and determines if sufficient investigation and testing have been carried out. If criteria are given, the staff reviews them to determine if they are appropriate and have been implemented. If it is the staff's judgment that the applicant's investigations or testing are inappropriate or insufficient, additional investigations will be required. The final conclusion is based on professional judgment, considering the complexity of the site subsurface conditions. As part of the review, the staff must ascertain, often with the help of consultants, that state-of-the-art laboratory and field techniques and equipment are employed in determining the material properties.		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
2.5.4-SAC-03	Foundation Interfaces. In meeting the requirements of **10 CFR Parts 50 and 100**, the discussion of the relationship of foundations and underlying materials is acceptable if it includes (1) a plot plan or plans showing the locations of all site explorations, such as borings, trenches, seismic lines, piezometers, geologic profiles, and excavations with the locations of the safety-related facilities superimposed thereon; (2) profiles illustrating the detailed relationship of the foundations of all seismic Category I and other safety-related facilities to the subsurface materials; (3) logs of core borings and test pits; and (4) logs and maps of exploratory trenches in the application for an early site permit or COL. A supplemental report providing geologic maps and photographs of the excavations for the facilities of the nuclear power plant should be provided when available. Plot plans and profiles are reviewed by comparing the subsurface materials with the proposed locations (horizontal and vertical) of foundations and walls of all seismic Category I facilities. The profiles and plot plans are cross-checked in detail with the results of all subsurface investigations conducted at the site to ascertain that sufficient exploration has been carried out and to determine whether or not the interpretations made by the investigators are valid and the foundation design assumptions contain adequate margins of safety.	N/A-COL	2.5.4
2.5.4-SAC-04	Geophysical Surveys. In meeting the requirements of **10 CFR 100.23**, the presentation of the dynamic characteristics of soil or rock is acceptable if geophysical investigations have been performed at the site and the results obtained there from are presented in detail. Completeness of the presentation is judged by whether or not the exploratory techniques used by the applicant yield unambiguous and useful information, whether they represent state-of-the-art exploration methods (Ref. 10), and whether the applicant's interpretations are supported by adequate field records in the SAR. Multiple measurements of dynamic	N/A-COL	2.5.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	properties should be incorporated to capture uncertainty in the primary parameters controlling site response behavior. See also Subsection 2.5.2.3. Staff evaluation consists of a detailed review of all geophysical explorations conducted at the site, including seismic refraction, reflection, and in-hole surveys and magnetic and gravity surveys. Consultant expertise regarding specific techniques may be drawn upon in this review. Logs of core borings, trenches, and test pits are reviewed and compared with data from the seismic surveys and other geophysical explorations. Results must be consistent or additional investigations are required, or the applicant must use the most conservative values. The staff will visit the site to examine the walls and floors of excavations at an appropriate time after licensing to confirm conditions as mapped in the open excavations with interpretations and assumptions derived during the investigation program.		
2.5.4-SAC-05	<u>Excavation and Backfill</u>. In meeting the requirements of **10 CFR Part 50**, the presentation of the data concerning excavation, backfill, and earthwork analyses is acceptable if: 1. The sources and quantities of backfill and borrow are identified and are shown to have been adequately investigated by borings, pits, and laboratory property and strength testing (dynamic and static) and these data are included, interpreted, and summarized. 2. The extent (horizontally and vertically) of all Category I excavations, fills, and slopes are clearly shown on plot plans and profiles. 3. Compaction specifications and embankment and foundation designs are justified by field and laboratory tests and analyses to ensure stability and reliable performance. 4. The impact of compaction methods are incorporated into the structural design of the plant facilities. 5. Quality control methods are discussed and the quality assurance program described and referenced.	N/A-COL	2.5.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	6. Control of groundwater during excavation to preclude degradation of foundation materials and properties is described and referenced. Excavations, backfill, and earthwork are evaluated by the staff as follows: 1. The investigations for borrow material, including boring and test pit logs, and compaction test data are reviewed and judged as to their adequacy. 2. Laboratory dynamic and static records of tests performed on samples compacted to the design specifications are reviewed to ascertain that state-of-the-art criteria are met. 3. Analyses and interpretations are reviewed to ensure that static and dynamic stability requirements are met. 4. Excavation and compaction specifications and quality control procedures are reviewed to ascertain conformance to state-of-the-art conservative standards.		
2.5.4-SAC-06	Ground Water Conditions. In meeting the requirements of **10 CFR Parts 50 and 100**, the analysis of groundwater conditions is acceptable if the following are included in this subsection or cross-referenced to the appropriate subsections in SRP Section 2.4 of the SAR: 1. Discussion of critical cases of groundwater conditions relative to the foundation settlement and stability of the safety-related facilities of the nuclear power plant. 2. Plans for dewatering during construction and the impact of the dewatering on temporary and permanent structures. 3. Analysis and interpretation of seepage and potential piping conditions during construction. 4. Records of field and laboratory permeability tests as well as dewatering induced settlements. 5. History of groundwater fluctuations as determined by periodic monitoring of 16 local wells and piezometers. Flood conditions should also be considered. Groundwater conditions as they affect foundation stability are evaluated by studying the	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	applicant's records of the historic fluctuations of groundwater at the site as obtained by monitoring local wells and springs and by analysis of piezometer and permeability data from tests conducted at the site. The applicant's dewatering plans during and following construction are also reviewed. Adequacy of these plans is evaluated by comparing with the results of the groundwater investigations and by professional judgment of groundwater and soil conditions at the site. The impact of these dewatering plans on temporary and permanent structures are evaluated.		
2.5.4-SAC-07	Response of Soil and Rock to Dynamic Loading. In meeting the requirements of **10 CFR Parts 50 and 100**, descriptions of the response of soil and rock to dynamic loading are acceptable if: 1. An investigation has been conducted and discussed to determine the effects of prior earthquakes on the soils and rocks in the vicinity of the site. Evidence of liquefaction and sand cone formation should be included (Ref. 12). 2. Field seismic surveys (surface refraction and reflection and in-hole and cross-hole seismic explorations) have been accomplished and the data presented and interpreted to develop bounding P and S wave velocity profiles (Ref. 10). 3. Dynamic tests have been performed in the laboratory on undisturbed samples of the foundation soil and rock sufficient to develop strain-dependent modulus reduction and hysterietic damping properties of the soils and the results included. The section should be cross-referenced with Subsection 2.5.2.5 (Ref. 11). The soil-structure interaction analysis should be described in SRP Sections 3.7.1 and 3.7.2 and cross-referenced to this subsection. Response of soil and rock to dynamic loading and soil-structure interaction is evaluated by a detailed study of the results of the investigations and analyses performed. Specifically, the effects of past earthquakes on site soils or rocks (a requirement in SRP Section 2.5.2) are determined. The data from core borings, from geophysical	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	investigations, and from dynamic laboratory tests such as sonic and resonant column, torsional shear and cyclic triaxial tests on undisturbed samples are evaluated. The object of the staff review is to ascertain that reasonably conservative dynamic soil and rock characteristic, together with their potential variability, are used in the design and analyses and that all the significant soil and rock strata have been considered in the analyses. In some cases, independent analyses and interpretations are carried out as outlined in SRP Section 2.5.2, or as required to verify the liquefaction analysis discussed in Subsection 2.5.4.8.		
2.5.4-SAC-08	<u>Liquefaction Potential</u>. In meeting the requirements of **10 CFR Parts 50 and 100**, if the foundation materials at the site adjacent to and under Category I structures and facilities are saturated soils and the water table is above bedrock, then an analysis of the liquefaction potential at the site is required (Ref. 12). The need for a detailed analysis is determined by a study on a case-by-case basis of the site stratigraphy, critical soil parameters, and the location of safety-related foundations. Undisturbed samples obtained at the site and appropriate laboratory tests are required to show if the soils are likely to liquefy. Liquefaction potential assessments using both deterministic and probabilistic approaches are desirable. When the need for an in depth analysis is indicated, it may be based on cyclic triaxial test data obtained from undisturbed soil samples taken from the critical zones in the site area. The shear stresses induced in the soil by the postulated earthquake should be determined in a manner that is consistent with SRP Section 2.5.2. The criterion that should be used to determine when the soil samples tested "liquefied" should be taken as the onset of liquefaction (defined as the cycle when the pore pressure first equals the confining pressure). Test data showing the rate of pore pressure increase with number of pad cycles should be presented. If the behavior of the pore pressure is such that peak to peak axial strains greater than a few percent occur before liquefaction, then the applicant must include the effects of these strains in his assessment of the potential hazards that	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	complete or partial liquefaction could have on the stability and settlement of any Category I structures. Nonseismic liquefaction (such as that induced by erosion, floods, wind loads on structures and wave action) should be analyzed using state-of-the-art soil mechanics principles. Liquefaction potential is reviewed by a study of the results of geotechnical investigations including boring logs, laboratory classification test data and soil profiles to determine if any of the site soils could be susceptible to liquefaction. The results of in-situ tests such as the standard penetration tests and the density and strength data obtained from undisturbed samples obtained in exploration borings are examined and, when appropriate, related to the liquefaction potential of in situ soils. If it is determined that there may be liquefaction-susceptible soils beneath the site, the applicant's site exploration methods, laboratory test program, and analyses are reviewed for adequacy and reasonableness. The analysis submitted by the applicant is reviewed in detail and compared to an independent study performed by the staff employing both deterministic and probabilistic methods as appropriate. As a minimum, the staff study consists of: 1. A review of appropriate standard penetration test results, other in-situ test data and groundwater conditions to assess liquefaction potential. 2. A careful review of conventional laboratory and cyclic triaxial test data to ensure that appropriate samples were obtained and tested from critical, liquefiable zones. 3. Confirmation that an adequate number of samples were properly tested and that the test results account for the natural variation in different samples as well as define the cyclic resistance to liquefaction of the soils. 4. An assessment of the liquefaction potential using a conservative envelope of the test data submitted.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	5. A calculation of the stress induced by the earthquake that has been arrived at by an envelope of critical conditions calculated for the site based on variations in the properties of the soil strata. 6. Assurance that conservative ranges of relative density of granular soils or relative consistency of fine-grained soils are estimated. Estimates of the "safety factor" obtained from the applicant's analysis are compared to the safety margins estimated by the staff. (The applicant's plans to "eliminate" the liquefaction condition, usually by excavation and backfill, vibroflotation, or chemical grouting is evaluated as discussed in Subsections 2.5.4.5 and 2.5.4.12.) 7. An assessment of post-earthquake stability and settlements due to partial liquefaction using state-of-the-art techniques. 8. An assessment of nonseismic liquefaction based on state-of-the-art techniques.		
2.5.4-SAC-09	Earthquake Design Basis. In meeting the requirements of **10 CFR Part 50**, the earthquake design basis analysis is acceptable if a brief summary of the derivation of the site-specific Ground Motion Response Spectrum (GMRS) is presented and references are included to Subsection 2.5.2.6. The staff's evaluation of the amplification characteristics of specific soils and rocks beneath the site as determined by procedures discussed in that section and in Subsections 2.5.4.2, 2.5.4.4, and 2.5.4.7 are summarized and cross-referenced herein. The review of Subsection 2.5.4.9 concentrates on determining its consistency or inconsistency with other subsections. Cross-referencing with other sections is expected.	N/A-COL	2.5.4
2.5.4-SAC-10	Static Stability. In meeting the requirements of **10 CFR Parts 50 and 100**, the discussions of static analyses are acceptable if the stability of all safety-related facilities has been analyzed from a static stability standpoint including bearing capacity (Ref. 22), rebound,	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	settlement, and differential settlements (Ref. 23) under deadloads of fills and plant facilities, and lateral loading conditions. The bearing capacity estimates must include consideration of settlements associated with the strength estimates. Field and laboratory test procedures and results must be included to document soil and rock properties used in the analyses. The applicant must show that the methods of analysis used are appropriate for the local soil conditions and the function of the facility. Static analyses of the bearing capacity and settlement of the supporting soils under the loads of fills, embankments, and foundations are evaluated by conventional, state-of-the-art methods (Ref. 18). In general, the evaluation procedure includes: 1. Determining whether or not the soil and rock properties used in the analyses represent the actual site conditions beneath the planned locations of plant facilities. The site investigation, sampling, and laboratory test programs must be adequate for this evaluation. 2. Determining whether or not the methods of analysis are appropriate for the planned earthworks, foundations, and soil conditions at the site. 3. Determining whether or not the bearing capacity, settlement, differential settlement, and tilt estimates indicate conservative and tolerable behavior of the planned plant foundations when these values are compared to design criteria and quality assurance specifications. 4. Evaluation of particularly complex cases on the basis of accepted principles and techniques as supplemented by case histories and confirmatory measurement and analysis programs.		
2.5.4-SAC-11	Design Criteria. In meeting the requirements of **10 CFR Part 50**, the discussion of criteria and design methods is acceptable if the criteria used for the design, the design methods employed, and the factors of safety obtained in the design analyses are described and a list of	N/A-COL	2.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	references presented. An explanation and verification of the computer analyses used and source references should be included. Site exploration, sampling, testing, and interpretation are judged with respect to completeness, care and technique, meaningful documentation, performance records for similar projects, published guidelines, and state-of-the-art practice. Design safety features, the applicant's proposed confirmatory tests and measurements, and monitoring of performance for planned safety-related foundations and earthworks are reviewed and evaluated on a case-by-case basis.		
2.5.4-SAC-12	Techniques to Improve Subsurface Conditions. In meeting the requirements of **10 CFR Part 50**, the discussion of techniques to improve subsurface conditions is acceptable if plans, summaries of specifications, and methods of quality control are described for all techniques to be used to improve foundation conditions (such as grouting, vibroflotation, dental work, rock bolting, or anchors). Planned techniques to improve subsurface conditions are evaluated by reviewing the applicant's specifications and techniques for performance and quality control for such activities as grouting, excavation and backfill, vibroflotation, rock bolting, and anchoring.	N/A-COL	2.5.4
SRP 2.5.5	**Stability of Slopes (R3, 03/2007)**		
2.5.5-AC-01	**10 CFR 50.55a, “Codes and Standards.”** This rule requires that structures, systems, and components shall be designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirement of applicable codes and standards commensurate with the importance of the safety function to be performed.	Y	2.5
2.5.5-AC-02	**10 CFR Part 50, Appendix A:**		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(a) **General Design Criterion 1 (GDC 1), "Quality Standards and Records,"** requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. It also requires that appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	Y (for Design, Fabricate, Erect, & Test of SSCs)	2.5.5
		N/A-COL (for Maintain Records)	2.5.5
	(b) **General Design Criterion 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena,"** as it relates to consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.	N/A-COL	2.5.5
	(c) **General Design Criterion 44 (GDC 44), "Cooling Water,"** requires that a system be provided with the safety function of transferring the combined heat load from structures, systems, and components important to safety to an ultimate heat sink under normal operating and accidental conditions.	N/A-COL	2.5.5
	(d) For ESP applications, GDC are not applicable. However, the **GDC 2** requirement to identify site characteristics that consider the most severe of the natural phenomena that have been historically reported for the site and surrounding area and with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated is specifically identified in **10 CFR 52.17(a)(1)(vi)**.	N/A-ESP	N/A
2.5.5-AC-03	**10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants,"** establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components of	Y	2.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2
Site Characteristics

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.		
2.5.5-AC-04	**10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants,"** as it applies to the design of nuclear power plant structures, systems, and components important to safety to withstand the effects of earthquakes.	Y	2.5
2.5.5-AC-05	**10 CFR Part 100, "Reactor Site Criteria,"** provides the criteria which guide the evaluation of the suitability of proposed sites for nuclear power and testing reactors.	N/A-COL	2.5.2
2.5.5-AC-06	**10 CFR 100.23, "Geologic and Seismic Criteria,"** provides the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and identify geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants.	N/A-COL	2.5.1
2.5.5-SAC-01	<u>Slope Characteristics</u>. In meeting the requirements of **10 CFR Parts 50 and 100**, the discussion of slope characteristics is acceptable if the subsection includes: 1. Cross sections and profiles of the slope in sufficient quantity and detail to represent the slope and foundation conditions. 2. A summary and description of static and dynamic properties of the soil and rock comprised by seismic Category I embankment dams and their foundations, natural and cut slopes, and all soil or rock slopes whose stability would directly or indirectly affect safety-related and Category I facilities. The text should include a complete discussion of procedures used to estimate, from the available field and laboratory data, conservative soil properties and profiles to be used in the analysis. 3. A summary and description of groundwater, seepage, and high and low groundwater conditions. Plot plans, cross sections, and profiles of all safety-related slopes in relation to the	N/A-COL	2.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	topography and physical properties of the underlying materials are reviewed and compared with exploratory records to ascertain that the most critical conditions have been addressed and that the characteristics of all slopes have been defined. The soil and rock test data are reviewed to ensure that there is sufficient relevant test data to verify the soil strength characteristics assumed for the slopes, dikes, and dams under analysis. The evaluation is to some extent a matter of engineering judgment; however, if the safety factors resulting from the analysis are not appropriate to the hazards posed by a slope failure and other than clearly conservative soil properties and profiles were used, the applicant is required to obtain additional data to verify his assumptions, or to show that, even if the worst possible conditions are assumed, there is an adequate margin of safety. With respect to seismic analysis, this subsection and subsection 2.5.5.2 are reviewed concurrently because different methods of analysis may involve different approximations, assumptions, and soil properties. In addition to generic state-of-the-art literature, other potential sources of information are those containing design, construction, and performance records of natural slopes, excavation slopes, and dams that may have been constructed in the general vicinity of the nuclear power plant.		
2.5.5-SAC-02	Design Criteria and Analyses. In meeting the requirements of **10 CFR Parts 50 and 100**, the discussion of design criteria and analyses is acceptable if the criteria for the stability and design of all seismic Category I slopes are described and valid static and dynamic analyses have been presented to demonstrate that there is an adequate margin of safety. A number of different methods of analysis are available in the literature. To be acceptable, the static analyses should include calculations with different assumptions and methods of analysis to assess the following factors: 1. The uncertainties with regard to the shape of the slope, boundaries of the several	N/A-COL	2.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	types of soil within the slope and their properties, the forces acting on the slope, and pore pressures acting within the slope. 2. Failure surfaces corresponding to the lowest factor of safety. 3. The effect of the assumptions inherent in the method of analysis used. 4. Adverse conditions such as high water levels due to the probable maximum flood (PMF), sudden drawdown, or steady seepage at various levels. In general, safety factors related to the slope hazard are needed; however, actual values depend somewhat on the method of analysis, on the assumptions concerning the soil properties, on construction techniques, and on the range of material parameters. To be acceptable, the dynamic analyses must account for the effect of cyclic motion of the earthquake on soil strength properties as well as the potential effects of both horizontal and vertical components of shaking. Actual test data are needed for both the in situ soils as well as for any materials used in the construction of dams or embankments. As discussed above, the various parameters, such as geometry, soil strength, modeling method (location and number of elements (mesh) if a finite-element analysis is used), and hydrodynamic and pore pressure forces, should be varied to show that there is an adequate margin of safety. Where liquefaction is possible, major dam foundation slopes and embankments should be analyzed by state-of-the-art finite-element or finite difference methods of analysis. Where there are liquefiable soils, changes in pore pressure due to cyclic loading must be considered in the analysis to assess not only the potential for liquefaction but also the effect of pore pressure increase on the stress-strain characteristic of the soil and the post-earthquake stability of the slopes. The criteria, design techniques, and analyses are evaluated by the staff to ascertain that: 1. Appropriate state-of-the-art methods have been employed. 2. Conservative assumptions regarding soil and rock properties have been used in the design and analysis of slopes and embankments as discussed above in subsection		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	2.5.5.1. 3. Appropriately conservative margins of safety have been incorporated in the design. The criteria and design methods used by the applicant are reviewed to ascertain that state-of-the-art techniques are being employed. The design analyses are reviewed to be sure that the most conservative failure approach has been used and that all adverse conditions to which the slope might be subjected have been considered. Such conditions include ground motions, both horizontal and vertical, from the safe shutdown earthquake, settlement, cracking, flood or low-water steady-state seepage, sudden drawdown of an adjacent reservoir, or a reasonable assumption of the possible simultaneous occurrence of two natural events such as an earthquake and flood. The review is also concerned with determining whether or not the soil and rock characteristics derived from the investigations described in subsection 2.5.5.3 have been completely and conservatively incorporated into the design. When marginal factors of safety are indicated by the independent analyses performed by the staff and its consultants, additional substantiation and refinement is required or the applicant must use more conservative assumptions. No single method of analysis is entirely acceptable for all stability assessments; thus, no single method of analysis can be recommended. Relevant manuals issued by public agencies (such as the U.S. Navy Department, U.S. Army Corps of Engineers, and U.S. Bureau of Reclamation) are often used in reviews to ascertain whether the analyses performed by the applicant are reasonable (Refs. 14, 15, 16, and 17). Many of the important interaction effects cannot be included in current analyses and must be treated in some approximate fashion. Engineering judgment is an important factor in the staff's review of the analyses and in assessing the adequacy of the resulting safety factors. If the staff review indicates that questionable assumptions have been made by the applicant or some nonstandard or inappropriate method of analysis has been used, then the staff or its consultant may model the dam or slope in a manner which it feels is more		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	consistent with the data and perform an independent analysis employing both deterministic and probabilistic methods as appropriate.		
2.5.5-SAC-03	<u>Boring Logs</u>. In meeting the requirements of **10 CFR Parts 50 and 100**, the applicant should describe the borings and soil testing carried out for slope stability studies and dam and dike analyses. Because dams, dikes, and natural or cut slopes are often remote from the main plant area, results of additional exploration, tests, and analyses for these areas should also be presented in this subsection. A comprehensive program of site investigations including borings, sampling, geophysical surveys, test pits, trenches, and laboratory and field testing must be carried out by the applicant to define the physical characteristics of all soil and rock beneath safety-related and seismic Category I slopes, and borrow material that is to be used to construct safety-related dams, fills, and embankments (Refs. 10 and 11). The staff reviews these investigations to ascertain that the program has been adequate to define the in situ and earthwork soil and rock characteristics. The decision as to the adequacy of the investigation program is based on the methods discussed in SRP Section 2.5.4.	N/A-COL	2.5.5
2.5.5-SAC-04	<u>Compacted Fill</u>. In meeting the requirements of **10 CFR Part 50**, the applicant should describe the excavation, backfill, and borrow material planned for any dams, dikes, and embankment slopes. Planned construction procedures and control of earthworks should be described. To be acceptable, the information must be given as discussed in subsection 2.5.4.5. Some of this information could be presented in subsection 2.5.4.5. Because dams, dikes, and other earthworks are often remote from the main seismic Category I structures, it is necessary to complete this information in this subsection. Quality control techniques and requirements during and following construction must also be discussed and referenced to quality assurance sections of the SAR.	N/A-COL	2.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 2 Site Characteristics			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	The preliminary specifications and quality control techniques to be used during construction are reviewed by the staff to ascertain that all design conditions are likely to be met (Refs. 5 and 9). During this part of the review the following are among those subjects reviewed for adequacy: 1. Proposed construction dewatering plan to ensure that it will not result in damage either to the natural or engineered foundation materials or to temporary or permanent structural foundations. 2. The excavation plan to remove all unsuitable materials from beneath the foundations and the quality control procedures which establish suitable materials. 3. The techniques and equipment to be used in compacting foundation and embankment materials. 4. The quality control and testing program to provide a high level of assurance that: a. The selected borrow material is as good and as relatively homogeneous as anticipated from the investigation program. b. The compacted foundation soil meets design specifications. 5. The techniques for improving the stability of natural slopes such as drainage, grouting, rock bolting, and applying shotcrete and/or gunite. The plans for monitoring during and after construction to detect occurrences that could detrimentally affect the facility. Such monitoring includes periodic examination of slopes, survey of settlement monuments, and measurements of local wells and piezometers.		

CHAPTER 3
Design of Structures, Components, Equipment and Systems

SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 3.2.1	**Seismic Classification (R2, 03/2007)**		
3.2.1-AC-01	**GDC 1**, and the pertinent QA requirements of **10 CFR Part 50, Appendix B**, as they relate to applying QA requirements to activities affecting the safety-related functions of SSCs designated as Seismic Category I commensurate with their importance to safety.	Y	3.2
3.2.1-AC-02	**GDC 2**, as it relates to the requirements that SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions.	Y	3.2.1
3.2.1-AC-03	**GDC 61**, as it relates to the design of radioactive waste systems, and other systems that may contain radioactivity, to assure adequate safety under normal and postulated accident conditions.	Y	3.2.1
3.2.1-AC-04	**10 CFR Part 100, Appendix A** and **10 CFR Part 50, Appendix S**, as it relates to certain SSCs being designed to withstand the SSE and remain functional.	Y	3.2.1
3.2.1-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.2.1-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections,	N/A-COL	N/A

CHAPTER 3
Design of Structures, Components, Equipment and Systems

SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.2.1-SAC-01	To meet the requirements of **GDC 2**, **10 CFR Part 100, Appendix A**, and **10 CFR Part 50, Appendix S** regarding seismic design classification are met by using guidance provided in **RG 1.29** "Seismic Design Classification." This guide describes an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE. **RG 1.151** provides guidance with regard to seismic design requirements and classification of safety-related instrumentation sensing lines. **RG 1.143** provides guidance used to establish the seismic design requirements of radioactive waste management SSCs to meet the requirements of **GDC 2 and 61** as they relate to designing these SSCs to withstand earthquakes. The guide identifies several radioactive waste SSCs requiring some level of seismic design consideration. **RG 1.189** provides guidance used to establish the design requirements of fire protection to meet the requirements of **GDC 2** as it relates to designing these SSCs to withstand earthquakes. This guide identifies portions of fire protection SSCs requiring some level of seismic design consideration.	Y	3.2.1
SRP 3.2.2	**System Quality Group Classification (R2, 03/2007)**		
3.2.2-AC-01	**10 CFR Part 50, Appendix A**, **General Design Criterion (GDC) 1** and **10 CFR Part 50.55a**, as they relate to structures, systems, and components important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.	Y	3.2.2

CHAPTER 3
Design of Structures, Components, Equipment and Systems

SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
3.2.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.2.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.2.2-SAC-01	To meet the requirements of **GDC 1** and **10 CFR 50.55a**, the following regulatory guide is used: **RG 1.26**, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." This guide describes an acceptable method for determining quality standards for Quality Group B, C, and D water- and steam-containing components important to safety of water-cooled nuclear power plants.	Y	3.2.2
SRP 3.3.1	**Wind Loadings (R3, 03/2007)**		
3.3.1-AC-01	**10 CFR 50**, **Appendix A**, **GDC 2** requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to	Y	3.3.1

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	perform their safety functions as it relates to natural phenomena. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.		
3.3.1-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
3.3.1-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.3.1-SAC-01	The wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.	N/A-COL	3.3.1
3.3.1-SAC-02	The acceptance criteria for the design wind speed, its recurrence interval, the speed variation with height, the applicable gust factors, and the bases for determining these site-related parameters, are stated in SRP Sections 2.3.1 and 2.3.2. The approved values of these parameters should serve as basic input to the	N/A-COL	3.3.1

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	review and evaluation of the structural design procedures.		
3.3.1-SAC-03	The procedures used to transform the wind speed unto an equivalent pressure to be applied to structures and parts, or portions of structures, as delineated in American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-05, "Minimum Design Loads for Buildings and Other Structures," are acceptable. In particular, the procedures used are acceptable if found in accordance with the following: A. For a design wind speed, V, the velocity pressure, q_z, evaluated at height, z, is given by: $q_z = 0.00256\ K_z K_{dt} K_d V^2 I$ (lb/ft^2) where: K_z = velocity pressure exposure coefficient evaluated at height, z, as defined in ASCE/SEI 7-05, Table 6-3, but not less than 0.87 K_{dt} = topographic factor equal to 1.0 Kd = wind directionality factor equal to 1.0 V = design wind speed in miles per hour (mi/h) as stated in SRP Section 2.3.1 I = importance factor equal to 1.15 B. For each wind direction considered, the upwind exposure category should be based on ground surface roughness that is determined from natural topography, vegetation, and constructed facilities. Surface roughness C is defined as open terrain with scattered obstructions having heights generally less than 30 ft. This category includes flat open country, grasslands, and all water surfaces in hurricane prone regions. Because most nuclear power plants	Y	3.3.1.1

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	are located in relatively open country, Kz values in Table 6-3 should be selected from the Exposure C column. The definition of Exposure C is provided in ASCE/SEI 7-05, Section 6.5.6.3. C. Design wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable. i. Section 6.5.12 – Design Wind Loads on Enclosed and Partially Enclosed Buildings ii. Section 6.5.13 - Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs iii. Section 6.5.14 - Design Wind Loads on Solid Freestanding Walls and Signs iv. Section 6.5.15 – Design Wind Loads on Other Structures		
SRP 3.3.2	**Tornado Loadings (R3, 03/2007)**		
3.3.2-AC-01	**10 CFR 50, Appendix A, GDC 2** requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions as it relates to natural phenomena. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.	Y	3.3.2
3.3.2-AC-02	**10 CFR 52.47(b)(1),** which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act,	ITAAC	Tier 1

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	and the NRC's regulations;		
3.3.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.3.2-SAC-01	The tornado wind and associated missiles generated by the tornado wind used in the design shall be the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.	N/A-COL	3.3.2
3.3.2-SAC-02	The acceptance criteria for tornado parameters including maximum wind speed, translational speed, rotational speed, and atmospheric pressure change, and the bases for determining these parameters are defined in SRP Sections 2.3.1 and 2.3.2. Acceptance criteria for the spectrum of tornado-generated missiles and their characteristics, as well as the bases for determining these parameters, are defined in SRP Section 3.5.1.4. These parameters should serve as basic input to the review and evaluation for structural design.	N/A-COL	3.3.2
3.3.2-SAC-03	The acceptance criteria for procedures used to transform tornado parameters into equivalent loads on structures are as follows: A. Tornado Characteristics and Effects Tornados are characterized, in **Table 1 of Regulatory Guide (RG) 1.76** for the contiguous United States into three geographical regions and by (1) maximum	Y	3.3.2

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	wind speed, (2) translational speed, (3) maximum rotational speed, (4) radius of maximum rotational speed, (5) pressure drop, and (6) rate of pressure drop for each of the three regions. Tornado effects are subdivided into three groups: i. Tornado wind effects caused by the direct action of air flow on structures, ii. Atmospheric pressure change effects caused by the differential pressure between the interior and exterior of a structure during the passage of a tornado, and iii. Tornado-generated missile impact effects. Tornado effects considered in design should include combinations of tornado wind effects, atmospheric pressure change effects, and tornado-generated missile impact effects. B. Tornado Wind Effects Procedures delineated in American Society of Civil Engineers/ Structural Engineering Institute (ASCE/SEI) 7-05, “Minimum Design Loads for Buildings and Other Structures” are acceptable for transforming tornado wind speed into pressure-induced forces applied to structures. In particular, the following shall apply: i. The maximum velocity pressure, q_z, should be based on the applicable maximum tornado wind speed, V, using the following equation from ASCE/SEI 7-05, Section 6.5.10: $q_z = 0.00256\ K_z K_{dt} K_d V^2 I\ (lb/ft^2)$ where: K_z = velocity pressure exposure coefficient equal to 0.87 K_{dt} = topographic factor equal to 1.0 K_d = wind directionality factor equal to 1.0		

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	V = maximum tornado wind speed (mi/h) I = importance factor equal to 1.15 The maximum tornado wind speed, V, is the resultant of the maximum rotational speed and the translational speed of the tornado. ii. Wind speed is assumed not to vary with the height above ground. iii. Design tornado wind loads should be determined in accordance with the following sections in ASCE/SEI 7-05, as applicable. (1) 6.5.12 Design Loads on Enclosed and Partially Enclosed Buildings (2) 6.5.13 Design Wind Loads on Open Buildings with Monoslope, Pitched, or Troughed Roofs (3) 6.5.14 Design Wind Loads on Solid Freestanding Walls and Solid Signs (4) 6.5.15 Design Wind Loads on Other Structures C. Atmospheric Pressure Change Effects **RG 1.76** provides guidance for determining the pressure drop and the rate of pressure drop caused by the passage of a tornado. “Wind Effects on Structures: Fundamentals and Applications to Design,” (Third Edition, John Wiley and Sons, Inc., New York, 1996.) by E. Simiu and R. H. Scanlan, provides methods for determining loads on structures due to atmospheric pressure changes during the passage of a tornado. For a structure that is completely open subjected to a tornado, the internal and external pressures on the structure equalize rapidly during the passage of the tornado. Therefore, the atmospheric pressure change between the interior and the exterior of that structure approaches zero.		

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	For a structure that is enclosed (unvented structure), the internal pressure remains equal to the atmospheric pressure before the passage of a tornado. The atmospheric pressure outside the structure changes during the passage of a tornado, which creates pressure differences between the interior and the exterior of that structure, and these differential pressures produce outward acting loads on the roof and walls of the enclosed structure. For a structure that is partially enclosed (vented structure), the determination of loads on the structure due to atmospheric pressure changes during the passage of a tornado is more complicated. If venting is adopted as a way to reduce the atmospheric pressure change effect on a structure, the review will be performed on a case-by-case basis. D. Tornado-Generated Missile Impact Effects Tornado-generated missile characteristics and the design-basis tornado missile spectrum are provided in **RG 1.76**. The acceptance criteria for transforming tornado-generated missile impact into equivalent static loads on structures are delineated in SRP Section 3.5.3, subsection II. E. Combined Tornado Effects After tornado-generated wind effects, Ww, atmospheric pressure change effects, Wp, and missile impact effects, Wm, are determined, the combination thereof should then be established in a conservative manner for structures. An acceptable method of combining these effects and establishing the total tornado load on a structure is as follows: $W_t = W_p$ Eq. 1 $W_t = W_w + 0.5\ W_p + W_m$ Eq. 2		

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	where: W_t = total tornado load W_w = load from tornado wind effect W_p = load from tornado atmospheric pressure change effect W_m = load from tornado missile impact effect		
3.3.2-SAC-04	The information provided to demonstrate that failure of any structure or component not designed for tornado loads will not affect the capability of other SSCs to perform necessary safety functions, is acceptable if found in accordance with either of the following: A. The postulated failure or collapse of structures and components not designed for tornado loads, including missiles, can be shown not to result in any structural or other damage to safety-related structures, systems, or components. B. Safety-related structures are designed to resist the effects of the postulated structural failure, collapse, or generation of missiles from structures and components not designed for tornado loads.	Y	3.3.2.3
SRP 3.4.1	**Internal Flood Protection for Onsite Equipment Failures (R3, 03/2007)**		
3.4.1-AC-01	The requirements of **10 CFR Part 50, Appendix A, GDC 2** relate to the SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Meeting the requirements of **GDC 2** includes evaluating the effects of flooding from full circumferential failures of non-seismic, moderate-energy piping, which is not considered in SRP Section	Y	3.4.1

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	3.6.2.		
3.4.1-AC-02	The requirements of **10 CFR Part 50, Appendix A, GDC 4** relate to the SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents.	Y	3.4.1
3.4.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.4.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.4.1-SAC-01	Guidance acceptable for meeting the seismic design and classification requirements of **GDC 2** is found in **Regulatory Guide (RG) 1.29, Position C.1** for safety-related SSCs **and Position C.2** for nonsafety-related SSCs.	Y	3.4.1
3.4.1-SAC-02	The requirements of **GDC 4** are met if SSCs important to safety are designed to accommodate the effects of discharged fluid resulting form high and moderate energy line breaks that are postulated in SRP sections 3.6.1 and 3.6.2.	Y	3.4.1

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SRP 3.4.2	**Analysis Procedures (R3, 03/2007)**		
3.4.2-AC-01	**GDC 2** requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions as it relates to natural phenomena. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.	Y	3.4.2
3.4.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.4.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.4.2-SAC-01	The highest flood and groundwater levels and the associated static and dynamic effects, if any, used in the design shall be the most severe ones that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have	Y	3.4.2

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	been accumulated.		
3.4.2-SAC-02	In most situations, the highest flood level is below the proposed plant grade and only its hydrostatic effects need be considered. Unless the hydrostatic head associated with the highest flood and groundwater levels is relieved by utilizing a drainage or a pumping system around the foundations of a structures, hydrostatic pressure has to be considered as a structural load on basement walls and the foundation slab of a structure. In consideration of any uplifting or floating of a structure, the total buoyancy force may be based on the highest flood level or the highest groundwater level excluding wave action. However, wave action should be included in the calculation for lateral and overturning movements of a structure.	Y	3.4.2
3.4.2-SAC-03	Where the flood level is above the proposed plant grade, the dynamic loads of wave action should be considered. Procedures for determining such dynamic loads are acceptable if they are in accordance with or equivalent to those delineated in the U.S. Army Coastal Engineering Research Center, “Shore Protection Manual” (Vol. I, June 2002, reprinted from 1973 edition and Vol. II, June 2002, reprinted from 1973 edition) or in EM 1110-2-1100, Coastal Engineering Manual, Part II, Chapter 1, “Water Wave Mechanics,” U.S. Army Corps of Engineers, April 30, 2002 as applicable.	N/A-OTHER (Flood level for reference design is below grade)	N/A
SRP 3.5.1.1	**Internally Generated Missiles (Outside Containment) (R3, 03/2007)**		
3.5.1.1-AC-01	**10 CFR 50, Appendix A, GDC 4** as it relates to the design of the SSCs important to safety if the design affords protection from the internally generated missile that may result from equipment failure.	Y	3.5.1.1
3.5.1.1-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary	ITAAC	Tier 1

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	and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.5.1.1-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.5.1.1-SAC-01	The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. Its statistical significance is determined by calculating the probability of missile occurrence. If this probability is less than 10^{-7} per year, the missile is not considered statistically significant. If the probability of occurrence is greater than 10^{-7} per year, the probability of impact on a significant target is determined. If the product of these two probabilities is less than 10^{-7} per year, the missile is not considered statistically significant. If the product is greater than 10^{-7} per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10^{-7} per year, the missile is not considered statistically significant. If the combined probability is greater than 10^{-7} per year, missile protection of SSCs important to safety, and of nonsafety-related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.	Y	3.5.1.1

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3.5.1.1-SAC-02	Missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing local shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important in safety. **RG 1.117** provides guidance on the SSCs that should be protected. Where barriers are used as a method of protection of SSCs from internal missiles, the design of the barriers is acceptable if it meets the guidance of **RG 1.115 position C.3**. Components within one train of a system with redundant trains need not be protected from missiles originating from the same train.	Y	3.5
SRP 3.5.1.2	**Internally Generated Missiles (Inside Containment) (R3, 03/2007)**		
3.5.1.2-AC-01	**10 CFR 50, Appendix A, GDC 4** as it relates to the design of the SSCs important to safety if the design affords protection from the internally generated missile that may result from equipment failure.	Y	3.5.1.2
3.5.1.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.5.1.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed	N/A-COL	N/A

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	inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.5.1.2-SAC-01	The applicant's statistical significance of an identified missile can be evaluated by a probability analysis. Its statistical significance is determined by calculating the probability of missile occurrence. If this probability is less than 10^{-7} per year, the missile is not considered statistically significant. If the probability of occurrence is greater than 10^{-7} per year, the missile is not considered significant. If the probability of occurrence is greater than 10^{-7} per year, the probability that it will impact a significant target is determined. If the product of these two probabilities is less than 10^{-7} per year, the missile is not considered significant. If the product is greater than 10^{-7} per year, the probability of significant damage is determined. If the combined probability (product of all three) is less than 10^{-7} per year, the missile is not considered significant. If the combined probability is greater than 10^{-7} per year, missile protection of SSCs important to safety, and of nonsafety -related SSCs whose failure could affect an intended safety function of the safety related SSCs, should be provided by one or more of the six methods listed below.	Y	3.5.1.2
3.5.1.2-SAC-02	The missile protection for SSCs important to safety is adequate if provided by one or more of the following methods: (1) locating the system or component in a missile-proof structure, (2) separating redundant systems or components for the missile path or range, (3) providing shields and barriers for systems and components, (4) designing the equipment to withstand the impact of the most	Y	3.5.1.2

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	damaging missile, (5) providing design features to prevent the generation of missiles, or (6) orienting missile sources to prevent missiles from striking equipment important to safety.		
SRP 3.5.1.3	**Turbine Missiles (R3, 03/2007)**		
3.5.1.3-AC-01	The NRC acceptance criteria is based on meeting the relevant requirements of General Design Criteria **(GDC) 4**, as it relates to SSCs important to safety being appropriately protected against environmental and dynamic effects, including the effects of missiles, that may result from equipment failure. Failure of large steam turbines in the main turbine generator has the potential to eject high-energy missiles that can produce such damage. The staff's overall safety objective is to ensure that SSCs important to safety are adequately protected from the effects of turbine missiles. Accordingly, consideration should be given to safety-related systems (i.e., those SSCs necessary to perform required safety functions). The specific criteria necessary to meet the relevant requirements of **GDC 4** to reduce the probability of turbine missile generation are as follows: A. The integrity of the reactor coolant pressure boundary; B. The capability to shut down and maintain the reactor in a safe condition; and C. The capability to prevent accidents that could result in potential offsite exposures, which represent a significant fraction of the guideline exposures specified in **10 CFR Part 100**, "Reactor Site Criteria." Examples of safety-related systems that should be protected are described in the Appendix to Regulatory Guide **(RG) 1.117,** “Tornado Design Classification.”	Y	3.5.1.3
3.5.1.3-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary	ITAAC	Tier 1

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	and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
3.5.1.3-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.5.1.3-SAC-01	The probability of unacceptable damage resulting from turbine missiles, P4, is expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine rotor (or internal structure) fragments through the turbine casing, P1; (b) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components, P2; and (c) the probability of struck structures, systems, or components failing to perform their safety function, P3. Stated in mathematical terms, $P4 = P1 \times P2 \times P3$. In accordance with the guidance provided in **SRP Section 2.2.3** and **RG 1.115**, the probability of unacceptable damage from turbine missiles should be less than or equal to 1 in 10 million per year for an individual plant (i.e., P4 should be < 10^{-7} per year per plant). Although the calculation of strike probability, P2, is not difficult in principle (i.e., a	Y	3.5.1

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	straightforward ballistics analysis), in practice it requires numerous modeling approximations and simplifying assumptions to define the properties of missiles, interactions of missiles with barriers and obstacles, trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and identification and location of safety-related targets. Specific approximations and assumptions tend to have a significant effect on the resulting value of P2. Similarly, a reasonably accurate specification of the damage probability, P3, is complicated by difficulties associated with defining the missile impact energy required to render safety-related systems unavailable to perform their safety functions and with postulating sequences of events that would follow a missile-producing turbine failure. Because of the uncertainties associated with calculating P2 and P3, the staff concludes that such analyses are "order of magnitude" calculations only. On the basis of simple estimates for a variety of plant layouts, the strike and damage probability product can be reasonably assumed to fall in a range that depends on the gross features of turbine generator orientation. A. For favorably oriented turbine generators, the product of P2 and P3 tends to be in the range of 10^{-4} to 10^{-3} per year per plant. B. For unfavorably oriented turbine generators, the product of P2 and P3 tends to be in the range of 10^{-3} to 10^{-2} per year per plant. Favorably oriented turbine generators are located such that the containment and all, or almost all, safety-related SSCs outside containment are excluded from the low-trajectory hazard zone described in **RG 1.115.** Because of assumptions and modeling difficulties in the probabilistic calculations as described above, the staff does not encourage applicants to calculate P2, P3, or their product. Instead, the staff accepts a product of strike and damage		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	probabilities of 10^{-3} per year per plant for a favorably oriented turbine and 10^{-2} per year per plant for an unfavorably oriented turbine. The suggested values represent the staff's best estimate of the product of P2 and P3, based on the results of calculations performed at the NRC (**NUREG-1048**, Supplement No. 6, and **NUREG-0887**, Supplement No. 3) and elsewhere.		
3.5.1.3-SAC-02	Operating experience indicates that turbine rotor crack (**NUREG/CR-1884**; PNO-111-81-104; and NRC Memorandum from E. Jordan to W. Russell), turbine stop and control valve failures (J.J. Burns, Jr.; License Event Report No. 82-132, Docket No. 50-361; and NRC Memorandum from E. Jordan to W. Russell), blade failures, and rotor ruptures can result in the generation of high-energy missiles (D. Kalderon and NRC Memorandum from E. Jordan to W Russell). Analyses indicate that missile generation can be modeled and the probability of missile generation can be strongly influenced by a suitable program of periodic inservice testing and inspection. In general, two modes of turbine rotor failure can result in turbine missile generation: (a) rotor material failure at approximately the rated operating speed and (b) failure of the overspeed protection system. Failure of turbine rotors at or below the design speed (nominally, 120% of normal operating speed) can be caused by small flaws or cracks that grow to critical size during operation. Failure of the turbine rotors at destructive overspeed (about 180% to 190% of normal operating speed) can result from failure of the overspeed protection system. The material properties of the turbine casing are of interest because secondary missiles could be generated if the casing fails or, alternatively, the casing could serve to arrest and contain missiles. The missile generation probability at the design speed should be related to rotor	N/A-COL	3.5.1.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	design parameters, material properties, and the intervals of inservice examinations of disks. The missile generation probability at the destructive overspeed should be related to the speed sensing and tripping characteristics of the turbine governor and overspeed protection system, the design and arrangement of main steam control and stop valves, the reheat steam intercept, reheat stop valves, and the inservice testing and inspection intervals for system components and valves. In addition, the turbine casing material in its operational environment should be evaluated for fracture toughness properties. **SRP Section 10.2** provides additional guidance regarding inspection and testing of turbine generator components. Further information regarding turbine missile generation mechanisms and probabilities can be found in **NUREG-1048**, NRC Memorandum from E. Jordan to W. Russell, and Letter from C. Rossi (NRC) to J. Martin (Westinghouse Electric Corporation).		
3.5.1.3-SAC-03	The staff believes that maintaining an acceptably low missile generation probability, P1, by means of a suitable program of periodic testing and inspection is a reliable method for ensuring that the objective of precluding generation of turbine missiles (and hence the possibility of damage to safety-related structures, systems, and components by those missiles) can be met. The NRC safety objective for turbine missiles (i.e., P4 should be < 10^{-7} per year per plant) is best expressed in terms of either of two sets of criteria applied to missile generation probability, P1. All applicants are expected to commit to operating criteria (see Table 3.5.1.3-1) appropriate to the applicable turbine orientation. One set of criteria should be applied to favorably oriented turbines; the other should be applied to unfavorably oriented turbines. This approach places responsibility on the applicant for initially demonstrating, and	N/A-COL	3.5.1.3

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	thereafter maintaining, an NRC-specified turbine reliability. Accordingly, the applicant should commit to conduct appropriate inservice inspection and testing throughout the life of the plant. Accordingly, the applicant should demonstrate the capability to perform visual, surface, and volumetric (ultrasonic) examinations suitable for inservice inspection of turbine rotors and shafts and provide reports, as required, describing the applicant's methods for determining turbine missile generation probabilities (**NUREG-1048** Supplement No 6; Letter from C. Rossi (NRC) to J. Martin (Westinghouse Electric Corporation); and **NUREG-0887**) for NRC review and approval.		
3.5.1.3-SAC-04	Applicants obtaining turbines from manufacturers that have prepared NRC-approved reports to describe their methods and procedures for calculating turbine missile generation probabilities are expected to meet criteria appropriate to the orientation of the turbine (see **Table 3.5.1.3-1**). Turbine manufacturers should provide applicants with tables of missile generation probabilities versus time (inservice visual, surface, and volumetric rotor inspection interval for design speed failure and inservice valve testing interval for destructive overspeed failure) for each turbine. These probabilities should be used to establish inspection and test schedules that meet NRC safety objectives.	N/A-COL	3.5.1.3

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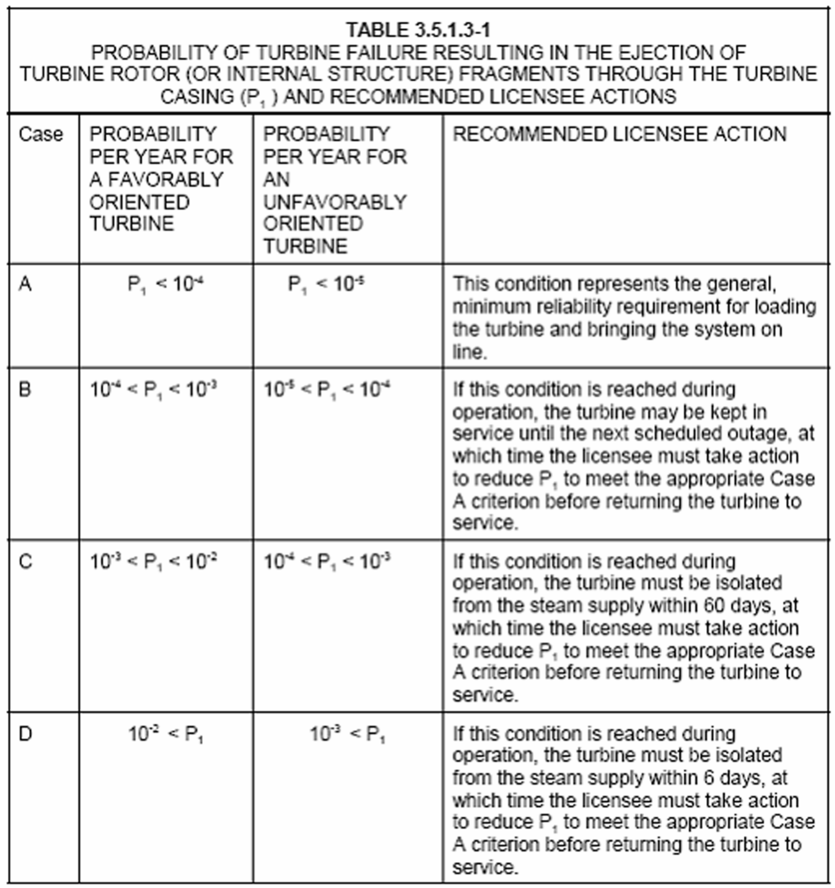
SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
			

TABLE 3.5.1.3-1
PROBABILITY OF TURBINE FAILURE RESULTING IN THE EJECTION OF TURBINE ROTOR (OR INTERNAL STRUCTURE) FRAGMENTS THROUGH THE TURBINE CASING (P_1) AND RECOMMENDED LICENSEE ACTIONS

Case	PROBABILITY PER YEAR FOR A FAVORABLY ORIENTED TURBINE	PROBABILITY PER YEAR FOR AN UNFAVORABLY ORIENTED TURBINE	RECOMMENDED LICENSEE ACTION
A	$P_1 < 10^{-4}$	$P_1 < 10^{-5}$	This condition represents the general, minimum reliability requirement for loading the turbine and bringing the system on line.
B	$10^{-4} < P_1 < 10^{-3}$	$10^{-5} < P_1 < 10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
C	$10^{-3} < P_1 < 10^{-2}$	$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine must be isolated from the steam supply within 60 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.
D	$10^{-2} < P_1$	$10^{-3} < P_1$	If this condition is reached during operation, the turbine must be isolated from the steam supply within 6 days, at which time the licensee must take action to reduce P_1 to meet the appropriate Case A criterion before returning the turbine to service.

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3.5.1.3-SAC-05	Applicants are expected to commit to the following program if turbines are obtained from manufacturers that have not submitted, or received NRC approval for, reports describing their methods and procedures for calculating turbine missile generation probabilities: A. An inservice inspection program should be used to detect rotor or disk flaws that could lead to brittle failure at or below design speed in the steam turbine rotor assembly. The turbine rotor design should facilitate inservice inspection of all high-stress regions, including disk bores and keyways, without removal of the disks from the shaft. The volumetric inservice inspection interval for the steam turbine rotor assembly should be established according to the following guidelines: i. The initial inspection of a new rotor or disk should be performed before any postulated crack is calculated to grow to more than one-half the critical crack depth. If the calculated inspection interval is less than the scheduled first fuel cycle, the licensee should seek the manufacturer's guidance on delaying the inspection until the first refueling outage. If the calculated inspection interval is longer than the first fuel cycle, the licensee should seek the manufacturer's guidance for scheduling the first inspection during a later refueling outage. ii. Disks that have been inspected and found free of cracks or that have been repaired to eliminate all indications of cracks should be reinspected using the criterion described in (1) above. Crack growth should be calculated from the time of the last inspection. iii. Disks operating with known and measured cracks should be reinspected before the elapse of one-half the time calculated for any crack to grow to	N/A-COL	3.5.1.3

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	one-half the critical depth. The guidance described in (1) above should be used to set the inspection date on the basis of the calculated inspection interval. iv. Under no circumstances should the volumetric inservice inspection interval for low-pressure (LP) disks exceed 3 years or two fuel cycles, whichever is longer. B. In accordance with the manufacturer's procedures, the turbine inservice inspection program should use visual, surface, and volumetric examinations to inspect turbine components such as couplings, coupling bolts, LP turbine shafts, blades and disks, and high-pressure (HP) rotors. Shafts and disks with crack(s) having depths at or near one-half the critical crack depth should be repaired or replaced. All cracked couplings and coupling bolts should be replaced. C. The inservice inspection and test program should be used for the governor and overspeed protection system to provide further assurance that flaws or component failures will be detected in the overspeed sensing and tripping subsystems, main steam control and stop valves, reheat steam intercept and stop valves, or extraction steam non-return valves — any of which could lead to an overspeed condition above that specified by the design overspeed. The inservice inspection program for operability of the governor and overspeed protection system should include, at a minimum, the following provisions: i. For typical turbine governor and overspeed protection systems, at intervals of approximately 3 years during refueling or maintenance shutdowns, at least one main steam control valve, one main steam stop valve, one reheat intercept valve, one reheat stop valve, and one of each type of steam		

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	extraction valve should be dismantled for examination. Visual and surface examinations of valve seats, disks, and stems should be conducted. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance. If any valve is shown to have flaws or excessive corrosion or improper clearances, the valve should be repaired or replaced. All other valves of that type should also be dismantled and inspected. ii. At least once a week during normal operation, main steam control and stop valves, reheat intercept and stop valves, and steam extraction nonreturn valves should be exercised by closing each valve and observing directly the valve motion as it moves smoothly to a fully closed position. iii. At least once a month during normal operation, each component of the electro-hydraulic governor system (which modulates control and intercept valves), as well as the primary and backup overspeed trip devices (both of which trip the main steam control and stop valves and the reheat intercept and stop valves), should be tested. The online test failure of any one of these subsystems mandates repair or replacement of failed components within 72 hours. Otherwise, the turbine should be isolated from the steam supply until repairs are completed. Refer to **SRP Section 10.2** for additional information regarding inspection and testing of turbine generator components. D. The design, inspection, and operating conditions should provide assurance that the probability of turbine missile generation will not exceed those described in **Table 3.5.1.3-1**.		
3.5.1.3-SAC-	An applicant may propose to install barriers or to take credit for existing structures	Y	3.5.1.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
06	or features as barriers. Such a decision could be based on the applicant's deterministic judgment that a SSCs is particularly vulnerable to destruction or unacceptable damage in the event of a turbine failure. The applicant should include specific details in the safety analysis report (SAR) supporting the need for such protection. If an applicant proposes to design or evaluate barriers to reduce or eliminate turbine missile hazards to equipment, the barriers should meet the acceptance criteria described in **SRP Section 3.5.3**. Additional design guidance is provided in “Fundamentals of Protective Design,” TM-5-885-1, Department of the Army, July 1965.		
SRP 3.5.1.4	**Missiles Generated by Tornados and Extreme Winds (R3, 03/2007)**		
3.5.1.4-AC-01	General Design Criterion **(GDC) 2**, “Design bases for protection against natural phenomena,” of Appendix A to 10 CFR Part 50, requires structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability to perform their safety functions.	Y	3.5.1.4
3.5.1.4-AC-02	**GDC 4,** “Environmental and dynamic effects design bases,” of Appendix A to 10 CFR Part 50, requires that SSCs important to safety be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.	Y	3.5.1.4
3.5.1.4-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design	ITAAC	Tier 1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
3.5.1.4-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.5.1.4-SAC-01	Regulatory Guide **(RG) 1.76** describes acceptable design-basis tornado-generated missile spectrum for the design of nuclear power plants.	Y	3.5.1.4
3.5.1.4-SAC-02	The method of identifying appropriate design-basis missiles generated by natural phenomena shall be consistent with the acceptance criteria defined for the evaluation of potential accidents from external sources in **SRP Section 2.2.3**. Other methodologies used by licensees and applicants with appropriate rationale may be acceptable on a case-by-case basis.	Y	3.5.1.4
SRP 3.5.1.5	**Site Proximity Missiles (Except Aircraft) (R4, 03/2007)**	N/A-COL	3.5.1.5
SRP 3.5.1.6	**Aircraft Hazards (R3, 03/2007)**	N/A-COL	3.5.1.6
SRP 3.5.2	**Structures, Systems, and Components to be Protected from Externally-Generated Missiles (R3, 03/2007)**		
3.5.2-AC-01	General Design Criterion **(GDC) 2**, “Design bases for protection against natural phenomena,” of Appendix A to **10 CFR Part 50**, requires structures, systems, and components (SSCs) important to safety shall be designed to withstand the effects of natural phenomena such as tornadoes and hurricanes without loss of capability	Y	3.5.2

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	to perform their safety functions.		
3.5.2-AC-02	**GDC 4**, “ Environmental and dynamic effects design bases,” of Appendix A to **10 CFR Part 50**, requires that SSCs important to safety be appropriately protected against the effects of missiles that may result from events and conditions outside the nuclear power unit.	Y	3.5.2
3.5.2-AC-03	**10 CFR 52.47(b)(1**), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.5.2-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.5.2-SAC-01	Acceptance is based on the design meeting the guidelines of Regulatory Guide **(RG) 1.13** as to the capability of spent fuel pool systems and structures to withstand the effects of externally-generated missiles and to prevent missiles from contacting stored fuel assemblies; **RG 1.27** as to the capability of the ultimate heat sink and connecting conduits to withstand the effects of externally-generated missiles; **RG 1.115** as to the protection of important safety-related SSCs from the	Y	3.5.2

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	effects of turbine missiles; and **RG 1.117** as to the protection of important safety-related SSCs from the effects of tornado missiles.		
SRP 3.5.3	**Barrier Design Procedures (R3, 03/2007)**		
3.5.3-AC-01	**GDC 2** requires that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, tsunami, floods, and seiches without loss of capability to perform their safety functions as it relates to natural phenomena. The design bases for these SSCs shall reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.	Y	3.5.2
3.5.3-AC-02	**GDC 4** requires that SSCs important to safety shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.	Y	3.5.2
3.5.3-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
3.5.3-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has	N/A-COL	N/A

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	been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.5.3-SAC-01	Specific criteria necessary to meet the relevant requirements of **GDC 2 and 4** are as follows: For Local Damage Prediction A. Concrete Sufficient thickness of concrete should be provided to prevent perforation, spalling, or scabbing of the barriers in the event of missile impact. Several empirical equations, such as the modified National Defense Research Council (NDRC) formula; proposed in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," by R.P. Kennedy, Nuclear Engineering and Design 1976 Pages 183-203 are available to estimate missile penetration into concrete. These equations should be used to determine the required barrier thicknesses. Thicknesses resulting from such calculations should not be less than those listed in Table 1, which specifies the minimum thicknesses necessary to protect against tornado missiles. Table 1, Minimum Acceptable Barrier Thickness Requirements, provides minimum concrete barrier thickness requirements for preventing local damage against tornado generated missiles for tornado spectrum shown in **Table 2 of Regulatory Guide (RG) 1.76.** Barrier thicknesses less than those listed in Table 1 may be used, provided that sufficient justification (including test data) is presented to support them. These justification will be reviewed on a case-by-case basis.	Y	3.5.3

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	Other types of missiles are specified in **SRP Sections 3.5.1.1 through 3.5.1.6**. For turbine missile barriers, penetration and scabbing predictions should be based on empirical equations such as the modified NDRC formula or the results of a valid test program.		

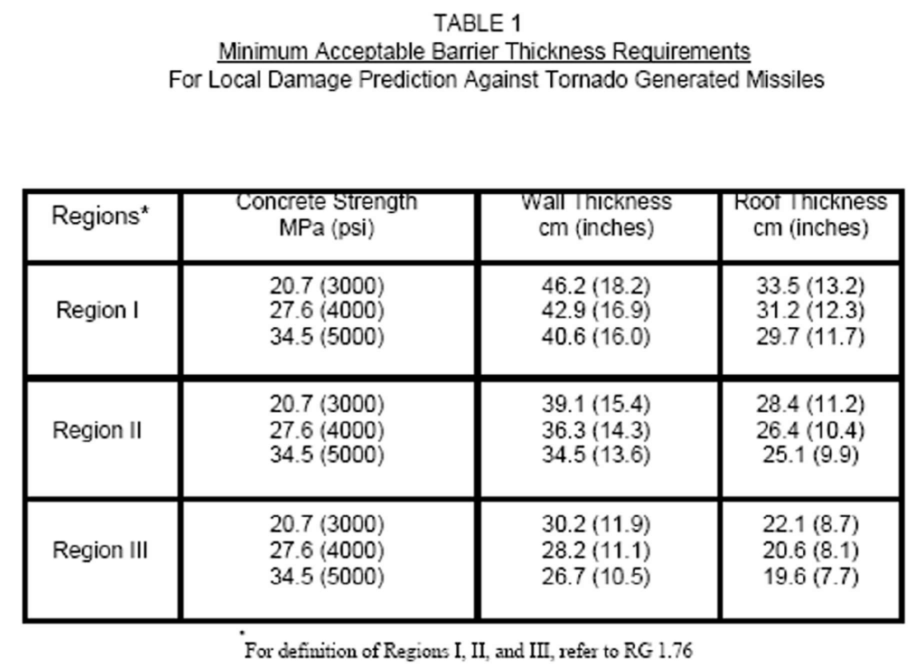

TABLE 1
Minimum Acceptable Barrier Thickness Requirements
For Local Damage Prediction Against Tornado Generated Missiles

Regions*	Concrete Strength MPa (psi)	Wall Thickness cm (inches)	Roof Thickness cm (inches)
Region I	20.7 (3000) 27.6 (4000) 34.5 (5000)	46.2 (18.2) 42.9 (16.9) 40.6 (16.0)	33.5 (13.2) 31.2 (12.3) 29.7 (11.7)
Region II	20.7 (3000) 27.6 (4000) 34.5 (5000)	39.1 (15.4) 36.3 (14.3) 34.5 (13.6)	28.4 (11.2) 26.4 (10.4) 25.1 (9.9)
Region III	20.7 (3000) 27.6 (4000) 34.5 (5000)	30.2 (11.9) 28.2 (11.1) 26.7 (10.5)	22.1 (8.7) 20.6 (8.1) 19.6 (7.7)

*For definition of Regions I, II, and III, refer to RG 1.76

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	B Steel The results of tests conducted by the Stanford Research Institute (SRI) on the penetration of missiles into steel plates are summarized in “U.S. Reactor Containment Technology” (ORNL/NSIC-5, Vol.1, Chapter 6, Oak Ridge National Laboratory, 1965) by W.B. Cottrell and A.W. Savolainen. The equations presented in aforementioned document are acceptable. Other equations such as the Ballistic Research Laboratory formula described in, “Reactor Safeguards,” by C. R. Russell, published by MacMillan, New York, 1962, may be used, provided the results are either comparable to those obtained by using the aforementioned “U.S. Reactor Containment Technology” method or are validated by penetration tests. C Composite Sections For composite or multi-element barriers, procedures for prediction of local damage are acceptable if the residual velocity of the missile perforating the first element is considered as the striking velocity for the next element. For determining this residual velocity, the equations presented in “Ballistic Perforation Dynamics,” Journal of Applied Mechanics, Transactions of the ASME, Vol. 30, Series E, No. 3, September 1963 by R. F. Recht and T. W. Ipson, are acceptable when the first barrier of a multi-element missile barrier is steel. When the first barrier is concrete, procedures used are reviewed on a case-by-case basis.		
3.5.3-SAC-02	For Overall Damage Prediction The response of a structure or barrier to missile impact depends largely on the location of impact (e.g., midspan of a slab or near a support), on the dynamic	Y	3.5.3.2

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	properties of the target and missile, and on the kinetic energy of the missile. In general, the assumption of plastic collisions is acceptable, where all of the missile's initial momentum is transferred to the target and only a portion of its kinetic energy is absorbed as strain energy within the target. However, where elastic impacts are expected, the additional momentum transferred to the target by missile rebound should be considered in the analyses. After it has been demonstrated that the missile will not penetrate the barrier, an equivalent static load concentrated at the impact area should then be determined, from which the structural response, in conjunction with other design loads, can be evaluated using conventional design methods. An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver, Inc., Revised November 1973 by R. A. Williamson and R. R. Alvy. Other procedures may be used, with adequate justification provided the results obtained are comparable to that of the above reference. Maximum allowable ductility ratios for steel and reinforced concrete barriers, in the above analysis, are given in American National Standard Institute/ American Institute of Steel Construction (ANSI/AISC) N690-1994 including supplement 2(2004), American National Standard Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities (1994) and in **RG 1.142**. respectively		
SRP 3.6.1	**Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment (R3, 03/2007)**		
3.6.1-AC-01	**10 CFR Part 50, Appendix A, GDC 2**, as it relates to protection against natural	Y	3.6.1

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	phenomena, such as seismically-induced failures of non-seismic piping. The application of 10 CFR Part 50, Appendix A, GDC 2 to this section is to incorporate environmental effects of full-circumferential ruptures of non-seismic moderate energy piping in areas where effects are not already bounded by failures of high energy piping. Acceptance is based on conformance to **BTP 3-3**.		
3.6.1-AC-02	**GDC 4**, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated pipe rupture. Acceptance is based on conformance to **BTP 3-3**.	Y	3.6.1
3.6.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.6.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.6.1-SAC-01	High and moderate energy fluid systems are separated from essential systems and components, as described in **Appendix B to BTP 3-3**.	Y	3.6.1
3.6.1-SAC-02	High and moderate energy fluid systems, or portions thereof, are enclosed as	Y	3.6.1

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	described in **item B.1.b of BTP 3-3**, which states "Fluid system piping or portions thereof not satisfying the provisions of item B.1.a should be enclosed within structures or compartments designed to protect nearby essential systems and components. Alternatively, essential systems and components may be enclosed within structures or compartments designed to withstand the effects of postulated piping failures in nearby fluid systems."		
3.6.1-SAC-03	For cases where neither physical separation nor protective enclosures are considered practical by the applicant, the reviewer will verify the following:	Y	3.6.1
	A. The reasons for which the applicant judged both physical separation and system enclosure to be impractical as means of protection are consistent with item **B.1.c. of BTP 3-3**, which states: "Plant arrangements or system features that do not satisfy the provisions of either item B.1.a or item B.1.b should be limited to those for which the above provisions are impractical because of the stage of design or construction of the plant; because the plant design is based upon that of an earlier plant accepted by the staff as a base plant under the Commission's standardization and replication policy; or for other substantive reasons such as particular design features of the fluid systems. Such cases may arise, for example, (1) at interconnections between fluid systems and essential systems and components, or (2) in fluid systems having dual functions (i.e., required to operate during normal plant conditions as well as to shut down the reactor). In these cases, redundant design features that are separated or otherwise protected from postulated piping failures, or additional protection, should be provided so that the effects of postulated piping failures are shown by the analyses and guidelines of Section B.3 to be acceptable.	Y	3.6.1

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	Additional protection may be provided by designing or testing essential systems and components to withstand the environmental effects associated with postulated piping failures.		
	B. Redundant design features or additional protections (assuming a single active failure in any required system) have been provided such that failure modes and effects analyses for all failure situations ensure the performance of safety features. These analyses are done under the criteria and assumptions of **item B.3. of BTP 3-3**.	Y	3.6.1
3.6.1-SAC-04	Design Features are in accordance with **item B.2 of BTP 3-3**.	Y	3.6.1
3.6.1-SAC-05	The effects of postulated failures on essential equipment and the ability of the plant to be safely shut down are analyzed in accordance with **item B.3. of BTP 3-3**.	Y	3.6.1
SRP 3.6.2	**Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping (R2, 03/2007)**		
3.6.2-AC-01	**GDC 4**, as it relates to SSCs important to safety being designed to accommodate the dynamic effects associated with postulated pipe rupture.	Y	3.6.2
		N/A-COL (For pipe break hazards analysis)	3.6.1 3.6.2
3.6.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design	ITAAC	Tier 1

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	certification, the provisions of the Atomic Energy Act, and the NRC's regulations		
3.6.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.6.2-SAC-01	Postulated Pipe Rupture Locations Inside Containment Acceptable criteria to define postulated pipe rupture locations and configurations inside containment are specified in **Branch Technical Position (BTP) 3-4**.	Y	3.6.2
		N/A-COL (For pipe break hazards analysis)	3.6.1 3.6.2
3.6.2-SAC-02	Postulated Pipe Rupture Locations Outside Containment Acceptable criteria to define postulated rupture locations and plant layout considerations for protection against postulated pipe ruptures outside containment are specified in **BTP 3-4**.	Y	3.6.2
		N/A-COL (For pipe break hazards analysis)	3.6.1 3.6.2
3.6.2-SAC-03	Methods of Analysis Detailed acceptance criteria covering pipe-whip dynamic analysis, including determination of the forcing functions of jet thrust and jet impingement, are included in **subsection III, "Review Procedures," of this SRP section**. The general bases and assumptions of the analysis are given in **BTP 3-4, subsection 2.C**.	Y	3.6.2
		N/A-COL (For pipe break hazards analysis)	3.6.1 3.6.2

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SRP 3.6.3	**Leak-Before-Break Evaluation Procedures (R1, 03/2007)**		
3.6.3-AC-01	**GDC 4** as it relates to the exclusion of dynamic effects of the pipe ruptures that are postulated in **SRP Section 3.6.2**. The design basis for the piping means those conditions specified in the SAR, as amended, and which may include regulations in 10 CFR Part 50, applicable sections of the SRP, Regulatory Guides, and industry standards such as the ASME Code.	Y	3.6.3
3.6.3-AC-02	LBB should only be applied to high energy, ASME Code Class 1 or 2 piping or the equivalent. Applications to other high energy piping will be considered based on an evaluation of the proposed design and inservice inspection requirements as compared to ASME Code Class 1 and 2 requirements.	Y	3.6.3
3.6.3-AC-03	Approval of the elimination of dynamic effects from postulated pipe ruptures is obtained individually for particular piping systems at specific nuclear power units. LBB is applicable only to an entire piping system or analyzable portion thereof. LBB cannot be applied to individual welded joints or other discrete locations. Analyzable portions are typically segments located between piping anchor points. When LBB technology is applied, all potential pipe rupture locations are examined. The examination is not limited to those postulated pipe rupture locations determined from **SRP Section 3.6.2**.	Y	3.6.3
3.6.3-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1

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3.6.3-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.6.3-SAC-01	Compliance with **GDC 4** requires that components important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. Safety-related components should be protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failure or events and conditions outside the nuclear power unit. Meeting the requirements of **GDC 4** provides assurance that SSCs important to safety will be protected from the dynamic effects of pipe rupture and capable of performing their intended safety function.	Y	3.6.3
3.6.3-SAC-02	LBB analyses should demonstrate that the probability of pipe rupture is extremely low under conditions consistent with the design basis for the piping. A deterministic evaluation of the piping system that demonstrates sufficient margins against failure, including verified design and fabrication and an adequate inservice inspection program, can be assumed to satisfy the extremely low probability criterion.	Y	3.6.3
SRP 3.7.1	**Seismic Design Parameters (R3, 03/2007)**		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
3.7.1-AC-01	**10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2 -** The design basis shall reflect appropriate consideration of the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.	N/A-COL	3.7.1.1.1
3.7.1-AC-02	**10 CFR Part 100, Subpart A** which is applicable to power reactor site applications before January 10, 1997, refers to Appendix A of this part for seismic criteria. **10 CFR Part 100**, **Appendix A** indicates that the SSE and the OBE shall be considered in the design of safety-related SSCs. 10 CFR Part 100, Appendix A further states that the design used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the SSE shall involve the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. **10 CFR Part 100, Subpart B** which is applicable to power reactor site applications on or after January 10, 1997, refers to **10 CFR 100.23** of this part for seismic criteria. Section 100.23 describes the criteria and nature of investigations required to obtain the geologic and seismic data necessary to determine the suitability of the proposed site and the plant design bases. 10 CFR 100.23 also refers to **10 CFR Part 50, Appendix S** for the definition of the minimum SSE ground motion for use in design.	N/A-COL	3.7.1.1.1
3.7.1-AC-03	**10 CFR Part 50, Appendix S is** applicable to applications for a design certification or combined license to **10 CFR Part 52** or a construction permit or operating license pursuant to 10 CFR Part 50 on or after January 10, 1997. For SSE ground	Y	3.7.1

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	motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion through design, testing, or qualification methods. The evaluation must take into account soil-structure interaction effects and the expected duration of the vibratory motion. If the OBE is set at one-third or less of the SSE, an explicit analysis or design is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. Appendix S also requires that the horizontal component of the SSE ground motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least. 0.1g.		
3.7.1-AC-04	**10 CFR 52.47(a)(1)** requires a DC applicant provide site parameters postulated for the design.	Y	3.7.1.1.1
3.7.1-AC-05	**10 CFR 52.47(b)(1), which requires** that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.7.1-AC-06	**10 CFR 52.79(b)** for a COL referencing an ESP as it relates to information sufficient to demonstrate that the design of the facility falls within the site characteristics and design parameters specified in the ESP.	N/A-ESP	N/A
3.7.1-AC-07	**10 CFR 52.79(d)(1)** for a combined license referencing a DC as it relates to information sufficient to demonstrate that the characteristics of the site fall within	N/A-COL	3.7.1.1.1

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	the site parameters specified in the design certification.		
3.7.1-AC-08	**10 CFR 52.80(a),** which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.7.1-SAC-01	Design Ground Motion A. Design Response Spectra. The site-specific GMRS reviewed under **SRP Section 2.5.2** are determined in the free-field on the ground surface. For sites with soil layers near the surface that will be completely excavated to expose competent material, the GMRS is specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop should be developed as a free surface motion, not as an in-column motion. Although the definition of competent material is not mandated by regulation, a number of reactor designs have specified a shear wave velocity of 1000 fps as the definition of competent material, which is considered acceptable. If noncompetent material is present, any excavation and/or backfilling should not alter the development or location of the site-specific GMRS. However, the soft soil or backfill material needs to be considered in the SSI or other analyses. According to **Appendix S to 10 CFR Part 50**, the minimum peak ground acceleration (PGA) for the horizontal component of the SSE at the	Y	3.7.1.1

<table>
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<tr><td></td><td>foundation level in the free-field should be 0.1g or higher. The response spectrum associated with this minimum PGA should be a smooth broad-band response spectra (e.g., RG 1.60, or other appropriate shaped spectra if justified) considered as an outcrop response spectra at the free-field foundation level. This response spectrum anchored at 0.1g will be referred in this SRP section as the minimum required response spectrum.
i. <u>Non-standard Plant Design</u>. For a non-standard plant design (e.g., COL application referencing only an ESP, or a COL application not referencing a CD and ESP), the design response spectra is developed from the site specific GMRS or from a broad band shaped spectra similar to RG 1.60 which also envelops the site-specific GMRS. Foundation level response spectra consistent with the design response spectra are determined for each seismic Category I structure. These foundation level spectra are compared to the minimum required spectrum to ensure they meet the 0.1g pga requirement in accordance with Appendix S to 10 CFR Part 50. If the foundation level spectra do not bound the minimum required response spectrum, then the design response spectra can be adjusted/modified in order to bound the minimum required spectrum. If the design response spectra are not modified, then the use of the two separate sets of spectra in the analysis and design of SSCs need to be reviewed for adequacy.
ii. <u>Certified Standard Plant Design (CD).</u> For a design certification (DC) application, the postulated seismic design response spectra need to bound the minimum required response spectrum anchored at 0.1g (as specified in Appendix S to 10 CFR Part 50). These design response</td><td></td><td></td></tr>
</table>

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	spectra are referred to as the CSDRS when the design is certified by the Commission under **10 CFR Part 52**. For a certified standard plant design (e.g., COL application that references a CD or a COL application that references a CD and ESP), a similar approach described above (under subsection II.1.a.i) is used to ensure that the CSDRS envelop the minimum required response spectrum. Foundation level response spectra consistent with the CSDRS are determined for each seismic Category I structure. These foundation level spectra are compared to the minimum required spectrum to ensure they meet the 0.1g pga requirement in accordance with **Appendix S to 10 CFR Part** 50. If the foundation level spectra do not bound the minimum required spectrum, then the CSDRS can be adjusted/modified in order to bound the minimum required spectrum If the CSDRS are not modified, then the use of the two separate sets of spectra in the analysis and design of SSCs need to be reviewed for adequacy. For evaluation of soil liquefaction and soil/rock stability of slopes that may affect plant safety, the use of the site-specific GMRS rather than the CSDRS is reviewed on a case-by-case basis. The free-field design response spectra (also referred to as the CSDRS for a CD) are usually developed for the 5-percent damping value. In the seismic analysis and design, the applicant needs to define the free-field design response spectra corresponding to all damping values to be used. For the case of **RG 1.60** response spectra, **Tables 1 and 2 of RG 1.60** provide amplification factors at four frequencies for calculating response spectra corresponding to different damping values. For the case of the free-field		

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	design response spectra that are different from **RG 1.60** response spectra, Appendix C to this SRP section provides procedures to calculate response spectra for different damping values other than 5 percent. To be acceptable, the seismic design response spectra should be specified for three mutually orthogonal directions - two horizontal and one vertical. Current practice is to assume that the design spectra (including maximum ground accelerations) in the two horizontal directions are the same. B. Design Time Histories. The SSE and OBE design ground motion time histories can be either real time histories or artificial time histories. To be acceptable, the design ground motion time histories should consist of three mutually orthogonal directions - two horizontal and one vertical. For both horizontal and vertical input motions, either a single time history or multiple time histories can be used. When time histories are used, each of the three ground motion time histories must be shown to be statistically independent from the others. Each pair of time histories are considered to be statistically independent if the absolute value of their correlation coefficient does not exceed 0.16. Simply shifting the starting time of a given time history can not be used to establish a different time history. Also, artificial time histories which are not based on seed recorded time histories should not be used. or linear structural analyses, the total duration of the artificial ground motion time histories should be long enough such that adequate representation of the Fourier components at low frequency is included in the time history. The corresponding stationary phase strong-motion duration should be consistent with the longest duration of strong motion from the earthquakes defined in **SRP Section 2.5.2** at low and high frequency and as presented in		

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	NUREG/CR-6728. The strong motion duration is defined as the time required for the Arias Intensity to rise from 5% to 75%. The uniformity of the growth of this Arias Intensity should be reviewed. The minimum acceptable strong motion duration should be six seconds. In addition to the duration, the ratios V/A and AD/V2 (A, V, D are peak ground acceleration, ground velocity, and ground displacement, respectively) should be consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the uniform hazard response spectra. These parameters should be consistent with the values determined for the low and high frequency events described in **Appendix D of RG 1.208**. For nonlinear structural analysis problems, multiple sets of ground motion time histories should be used to represent the design ground motion. Each set of ground motion time histories shall be selected from real recorded ground motions appropriate for the characteristic low and high frequency events. The amplitude of these ground motions may be scaled but the phasing of Fourier components must be maintained. The adequacy of this set of ground motions, including-duration estimates, is reviewed on a case-by-case basis. Option 1: Single Set of Time Histories. To be considered acceptable, the response spectra generated from the artificial time history to be used as input ground motion in the free-field should satisfy the enveloping requirements for either Approach 1 or Approach 2 below: i. Approach 1. For Approach 1, the spectrum from the artificial ground motion time history must envelop the free-field design response spectra		

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<tr><td></td><td>for all damping values used in the seismic response analysis. When spectral values (e.g., spectral accelerations) are calculated from the artificial time history, the frequency intervals at which spectral values are determined are to be sufficiently small. Table 3.7.1-1 (below) provides an acceptable set of frequencies at which the response spectra may be calculated.
<table>
<tr><th colspan="2">Table 3.7.1-1
Suggested Frequency Intervals for Calculation of Response Spectra</th></tr>
<tr><th>Frequency Range</th><th>Increment</th></tr>
<tr><td>(hertz)</td><td>(hertz)</td></tr>
<tr><td>0.2 - 3.0</td><td>0.10</td></tr>
<tr><td>3.0 - 3.6</td><td>0.15</td></tr>
<tr><td>3.6 - 5.0</td><td>0.20</td></tr>
<tr><td>5.0 - 8.0</td><td>0.25</td></tr>
<tr><td>8.0 - 15.0</td><td>0.50</td></tr>
<tr><td>15.0 - 18.0</td><td>1.0</td></tr>
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<tr><td>18.0 - 22.0</td><td>2.0</td></tr>
<tr><td>22.0 - highest frequency of interest</td><td>3.0</td></tr>
</table>
<p>Each calculated spectrum of the artificial time history is considered to envelop the design response spectrum when no more than five points fall below, and no more than 10 percent below, the design response spectrum.</p>
<p>Studies indicate that numerically generated artificial ground acceleration histories produce PSD functions having a quite different appearance from one individual function to another, even when all these time histories are generated so as to closely envelop the same design response spectra. For example, the use of the available techniques of generating acceleration time histories that satisfy enveloping RG 1.60 spectra usually results in PSD functions that fluctuate significantly and randomly as a function of frequency. It is also recognized that the more closely one tries to envelop the specified design response spectra, the more significantly and randomly do the spectral density functions tend to fluctuate and these fluctuations may lead to unconservative results for the response of SSCs. Therefore, when a single artificial ground motion time history is used in the design of seismic Category I SSCs, it must in general satisfy requirements for both enveloping design response spectra as well as adequately matching a target PSD function compatible with the design</p>
</td><td></td><td></td></tr>
</table>

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	response spectra. Therefore, in addition to the response spectra enveloping requirement, the use of a single time history should also be justified by demonstrating sufficient energy at the frequencies of interest through the generation of PSD function, which envelops the target PSD function throughout the frequency range of significance. When **RG 1.60** response spectra are used as design response spectra, the requirements for a compatible target PSD are contained in Appendix A to this SRP section. Target PSD functions other than those given in Appendix A can be used if justified. For design response spectra other than **RG 1.60** response spectra, a compatible target PSD should be generated. For generation of target PSD in such cases, the guidelines and procedures provided in Appendix B to this SRP section can be used. Procedures used to generate the target PSD will be reviewed on a case-by-case basis. The PSD requirements are included as secondary and minimum requirements to prevent potential deficiency of power over the frequency range of interest. It should be noted that the ground motion is still primarily defined by the design response spectrum. The use of PSD criteria alone can yield time histories that may not envelop the design response spectrum. ii. Approach 2. For Approach 2, the artificial ground motion time histories that are generated to match or envelop the design response spectra shall comply with Steps (a) through (d) below. The general objective is to generate a modified recorded or artificial accelerogram which achieves approximately mean based fit to the target response spectrum; that is, the average ratio of the spectral acceleration calculated from the accelerogram to the target, where the ratio is		

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	calculated frequency by frequency, is only slightly greater than "1." The aim is to achieve an accelerogram that does not have significant gaps in the Fourier amplitude spectrum, but which is not biased high with respect to the target. a) The time history shall have a sufficiently small time increment and sufficiently long duration. Records shall have a Nyquist frequency of at least 50 Hz, (e.g., a time increment of at most 0.010 seconds) and a total duration of at least 20 seconds. If frequencies higher than 50 Hz are of interest, the time increment of the record must be suitably reduced to provide a Nyquist frequency (Nf =1/(2Δt), where Δt = time increment) above the maximum frequency of interest. The total duration of the record can be increased by zero packing to satisfy these frequency criteria. b) Spectral acceleration at 5% damping shall be computed at a minimum of 100 points per frequency decade, uniformly spaced over the log frequency scale from 0.1 Hz to 50 Hz or the Nyquist frequency. The comparison of the response spectrum obtained from the artificial ground motion time history with the target response spectrum shall be made at each frequency computed in the frequency range of interest. c) The computed 5% damped response spectrum of the accelerogram shall not fall more than 10% below the target response spectrum at any one frequency. To prevent response spectra in large frequency windows from falling below the target response spectrum, the response spectra within a frequency window of no larger than ±10%		

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	centered on the frequency shall be allowed to fall below the target response spectrum. This corresponds to response spectra at no more than 9 adjacent frequency points defined in (b) above from falling below the target response spectrum. d) In lieu of the power spectrum density requirement of Approach 1, the computed 5% damped response spectrum of the artificial ground motion time history shall not exceed the target response spectrum at any frequency by more than 30% (a factor of 1.3) in the frequency range of interest. If the response spectrum for the accelerogram exceeds the target response spectrum by more than 30% at any frequency range, the power spectrum density of the accelerogram needs to be computed and shown to not have significant gaps in energy at any frequency over this frequency range. Artificial ground motion time histories defined as described above shall have characteristics consistent with characteristic values for the magnitude and distance of the appropriate controlling events defined for the uniform hazard response spectrum (UHRS). <u>Option 2: Multiple Sets of Time Histories</u>. As discussed in Section I.1.B and Section II.1.B of this SRP section, the use of multiple real or artificial time histories for analyses and design of SSCs is acceptable. For linear structural analyses, a minimum of four times histories should be used. For nonlinear structural analyses, the number of time histories must be greater than four and the technical basis for the appropriate number of time histories are reviewed on a case-by-case basis. This review also includes		

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	the adequacy of the characteristics of the multiple time histories. The response spectra calculated for each individual time history need not envelop the design response spectra. However, the multiple time histories are acceptable if the average calculated response spectra generated from these time histories envelop the design response spectra. An acceptable method to demonstrate the adequacy of a set of multiple time histories, in terms of enveloping requirements and having sufficient power over the frequency range of interest, is to follow the procedures described for Approach 2 presented in subsection II.1.B.ii of this SRP. When implementing Approach 2, the criteria in paragraphs (a) and (b) of this approach need to be satisfied for each of the time histories. The criteria in paragraphs (c) and (d) of this approach can be satisfied by utilizing the results for the average of the suite of multiple time histories.		
3.7.1-SAC-02	<u>Percentage of Critical Damping Values</u>. The specific percentage of critical damping values used in the analyses of Category I SSCs are considered to be acceptable if they are in accordance with **RG 1.61**. Damping values different from those listed in **RG 1.61** (e.g., higher damping values) may be used in a dynamic seismic analysis if test data are provided to support them. These damping values will be reviewed and accepted by the staff on a case-by-case basis. In addition, a demonstration of the correlation between stress levels and damping values will be required and reviewed for compliance with the applicable regulatory position in **RG 1.61**. If other methods for correlation of damping values with stress level are used, they will need to be reviewed and accepted on a case-by-case	Y	3.7.1.2

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	basis. The material soil damping for foundation soils must be based upon validated values or other pertinent laboratory data, considering variation in soil properties and strains within the soil, and must include an evaluation of dissipation from pore pressure effects as well as material damping for saturated site conditions. The maximum soil damping value acceptable to the staff is 15 percent.		
3.7.1-SAC-03	Supporting Media for Seismic Category I Structures. To be acceptable, the description of supporting media for each Category I structure must include foundation embedment depth, depth of soil over bedrock, soil layering characteristics, design groundwater elevation, dimensions of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, Poisson's ratios, and density as a function of depth. If the minimum shear wave velocity of the supporting foundation material is less than 1,000 fps, additional studies need to be performed which consider the average shear wave velocity, and its degree of variability addressing potential impact on soil-structure interaction, potential settlements and design of foundation elements.	Y	3.7.1.3
3.7.1-SAC-04	Review Considerations for DC and COL Applications A. COL Application Referencing an ESP and CD In addition to the criteria presented below, Figures 1 and 2 in Appendix D provide additional guidance in understanding the Part 52 process. i. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. For COL application referencing an ESP and CD, the GMRS are included in the ESP. ii. Confirm that the criterion for the minimum required response spectrum	N/A-COL	3.7.1.1

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	(in accordance with subsection II.1.A.ii) has been satisfied. iii. Confirm that COL action items contained in the CD have been met. This includes seismic design parameters such as soil layering assumptions used in the certified design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity. Technical justification for all deviations from the range of values used in the standard plant design must be provided for review. iv. Confirm that the ESP conditions have been met or review the COL applicant's approach to address any deviations. v. When the site-specific GMRS and the CSDRS, are calculated at the same elevation, confirm that the CSDRS envelop the GMRS. For this case the standard design is acceptable for that site, assuming no other issue is identified during the review process. If the CSDRS do not envelop the site-specific GMRS then proceed to step vii. vi. When the site-specific GMRS and the CSDRS are determined at different elevations, calculate the site-specific GMRS transferred to the base elevations of each seismic Category I foundation. These site-specific GMRS at the foundation levels are referred to as foundation input response spectra (FIRS) and are derived as free-field outcrop spectra; that is, only the effects of materials that are below the base elevation of the seismic Category I structure are included in the site response analysis. For each seismic Category I structure foundation, if the SDRS-consistent spectra at the foundation level envelop the site-specific FIRS at the foundation level, the standard design is acceptable for that site, assuming no other issue is identified during the review		

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	process. If not, then proceed to step vii. Perform an SSI analysis using the site-specific FIRS and an advanced seismic analytical technique (e.g., method that considers the effects of incoherent ground motion). When such analytical methods are utilized, the detailed technical justification shall be reviewed on a case-by-case basis. Further discussion on consideration of the effects of incoherent ground motion is provided in subsection II.4.C (under the heading Input Ground Motion, Specific Guidelines for SSI Analysis) in SRP Section 3.7.2. The in-structure responses in terms of floor response spectra, building member forces, and deformations at key locations in the structure shall be obtained. The key locations for calculating the in-structure responses, proposed by the licensee, need to be evaluated to ensure that they are sufficient to represent the various locations throughout the building. Locations should include responses at peripheral locations to detect rocking and torsion, and should include responses to check overturning, torsional, and sliding stability of the structures. The dynamic models and analysis techniques need to be sufficiently refined to be able to capture the response of the structures throughout the frequency range of interest, including the high frequency responses, typically expected in the central and eastern United States (CEUS) regions. The SSI analysis shall also consider the site-specific soil variability (i.e., best estimate, lower bound estimate, and upper bound estimate). Compare these responses at the key locations in the structure to the standard design in-structure responses. If the CSDRS responses envelop the in-structure responses from the FIRS, the		

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	standard design is acceptable assuming no other issue is identified during the review process. If the responses are not enveloped, additional analyses are required to demonstrate the acceptability of the design or the design might need to be modified. If further analyses are utilized, then the analyses must consider the potentially higher responses at all locations, not only those at the key locations described above. B. COL Application Referencing a CD. Follow the same steps described above under A – COL Application Referencing an ESP and CD, except that step iv. does not apply to this case. C. COL Application Referencing an ESP. In addition to the criteria presented below, Figure 3 in Appendix D provides additional guidance in understanding the process when a certified design is not used. i. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. For COL application referencing an ESP, the GMRS are included in the ESP. ii. Confirm that the ESP conditions have been met or review the COL applicant’s approach to address any deviations. iii. Follow the acceptance criteria described in subsection II.1.A (excluding subsection II.1.A.ii), of this SRP Section to develop the seismic design response spectra. The seismic SSI analysis would then follow the conventional approach for SSI analyses.		

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	D. <u>COL Application not Referencing an ESP and DC.</u> In addition to the criteria presented below, Figure 3 in Appendix D provides additional guidance in understanding the process when a certified design is not used. i. Site-specific GMRS are reviewed separately under SRP 2.5.2 for adequacy. ii. Follow the acceptance criteria described in subsection II.1.A (excluding subsection II.1.A.ii), of this SRP Section to develop the seismic design response spectra. The seismic SSI analysis would then follow the conventional approach for SSI analyses. iii.		
SRP 3.7.2	**Seismic System Analysis (R3, 03/2007)**		
3.7.2-AC-01	**10 CFR Part 50, General Design Criterion (GDC) 2** - The design basis shall reflect appropriate consideration of the most severe earthquakes that have been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been accumulated.	N/A-COL	3.7.2
3.7.2-AC-02	**10 CFR Part 100, Subpart A**, which is applicable to power reactor site applications before January 10, 1997, refers to Appendix A of this part for seismic criteria. **10 CFR Part 100**, **Appendix A** indicates that the SSE and the OBE shall be considered in the design of safety-related SSCs. **10 CFR Part 100, Appendix A** further states that the design used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the SSE shall involve the use of either a suitable dynamic analysis or a suitable	N/A-COL	3.7.2

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	qualification test to demonstrate that SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. **10 CFR Part 100,Subpart B** which is applicable to power reactor site applications on or after January 10, 1997, refers to **10 CFR 100.23** of this part for seismic criteria. Section100.23 describes the criteria and nature of investigations required to obtain the geologic and seismic data necessary to determine the suitability of the proposed site and the plant design bases. 10 CFR 100.23 also indicates that applications to engineering design are contained in **10 CFR part 50, Appendix S**.		
3.7.2-AC-03	**10 CFR Part 50, Appendix S** is applicable to applications for a design certification or combined license to 10 CFR Part 52 or a construction permit or operating license pursuant to 10 CFR Part 50 on or after January 10, 1997. For SSE ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion through design, testing, or qualification methods. The evaluation must take into account soil-structure interaction effects and the expected duration of the vibratory motion. If the OBE is set at one-third or less of the SSE, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied. Appendix S also requires that the horizontal component of the SSE ground motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.	Y	3.7.2
3.7.2-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary	ITAAC	Tier 1

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	and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
3.7.2-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations	N/A-COL	N/A
3.7.2-SAC-01	<u>Seismic Analysis Methods</u>. The seismic analysis of all seismic Category I SSCs should use either a suitable dynamic analysis method or an equivalent static load analysis method, if justified. The SRP acceptance criteria primarily address linear elastic analysis coupled with allowable stresses near elastic limits of the structures. However, for certain special cases (e.g., evaluation of as-built structures), reliance on limited inelastic/nonlinear behavior when appropriate is acceptable to the staff. Analysis methods incorporating inelastic/nonlinear considerations and the analysis results are reviewed on a case-by-case basis.	Y	3.7.2.1
	A. <u>Dynamic Analysis Method</u>. When calculating seismic responses of Category 1 structures, dynamic analysis (response spectrum analysis method or time history analysis method) should be performed. To be acceptable, dynamic analyses should consider the		

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	following:		
	i Use of appropriate methods of analysis (time history analysis method [time domain solution and frequency domain solution]; response spectrum analysis method), accounting for the effects of SSI, if applicable. In general, the response spectrum analysis method is not suitable for SSI analysis.		
	ii Seismic analysis should be performed for three orthogonal (two horizontal and one vertical) components of earthquake ground motion.		
	iii Consideration of the torsional, rocking, and translational responses of the structures and their foundations (including footings, basemats and buried walls).		
	iv Use of an adequate number of discrete mass degrees of freedom in dynamic modeling. The adequacy of the number of discrete mass degrees of freedom can be confirmed by (1) preliminary modal analysis, and (2) correlation between static analysis results using the dynamic model and static analysis results using a distributed mass representation.		
	(1) It is important to ensure that, for each excitation direction (2 horizontal and vertical), all modes with frequencies less than the ZPA (or PGA) frequency of the corresponding spectrum are adequately represented in the dynamic solution. Preliminary modal analysis should be performed to establish that a sufficient number of discrete mass degrees of freedom have been included in the dynamic model to (a) predict a sufficient number of modes, and (2) produce mode shapes		

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	that are reasonably smooth. If a mode shape exhibits rapid change in modal displacement between adjacent mass degrees of freedom, additional mass degrees of freedom should be added until reasonably smooth mode shapes are obtained for all modes to be included in the dynamic analysis.		
	(2) After completion of (1), simple 1g static analyses of the dynamic model should be performed for each of the three (3) excitation directions, and compared to the corresponding results obtained from static analyses that utilize a distributed mass representation. Lack of correlation, particularly in the vicinity of and at support locations, is indicative of an insufficient number of discrete mass degrees of freedom.		
	v When using either the response spectrum method or the modal superposition time history method, responses associated with high frequency modes (i.e., f > ZPA [or PGA] frequency) should be included in the total dynamic solution using the guidance and methods described in Regulatory Guide 1.92, Revision 2, Regulatory Positions C.1.4 and C.1.5.		
	vi Consideration of maximum relative displacements between adjacent supports of seismic Category I SSCs.		
	vii Inclusion of significant effects such as piping interactions, externally applied structural restraints, hydrodynamic (both mass and stiffness effects) loads, and nonlinear responses.		
	B. <u>Equivalent Static Load Method</u>. An equivalent static load method is acceptable if:		

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	i Justification is provided that the system can be realistically represented by a simple model and the method produces conservative results in terms of responses. Typical examples or published results for similar structures may be submitted in support of the use of the simplified method.		
	ii The simplified static analysis method accounts for the relative motion between all points of support.		
	iii To obtain an equivalent static load for an SSC that can be represented by a simple model, a factor of 1.5 is applied to the peak spectral acceleration of the applicable ground or floor response spectrum. A factor less than 1.5 may be used, if adequate justification is provided.		
3.7.2-SAC-02	<u>Natural Frequencies and Responses</u>. To be acceptable, the following information should be provided:	Y	3.7.2.2
	A. A summary of modal masses, effective masses, natural frequencies, mode shapes, modal and total responses for the Category I structures, including the containment structure, or a summary of the total responses if the method of direct integration is used.		
	B. The calculated time histories (two horizontal and one vertical), or other parameters of motion, or response spectra (two horizontal and one vertical) used in design, at the major plant equipment elevations and points of support.		
	C. For the multiple time history analysis option, procedures used to account for uncertainties (by variation of parameters) and to develop design responses, including justification for the statistical relationship between input design time histories and output responses. (For example, if the average response spectra		

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	generated from the multiple design time histories are used to envelop the design response spectra, then the average responses generated from the multiple analyses are used in design.)		
3.7.2-SAC-03	Procedures Used for Analytical Modeling. A nuclear power plant facility consists of very complex structural systems. To be acceptable, the stiffness, mass, and damping characteristics of the structural systems should be adequately incorporated into the analytical models. Specifically, the following items should be considered in analytical modeling:	Y	3.7.2.3
	A. Designation of Systems Versus Subsystems. Category I structures that are considered in conjunction with the foundation and its supporting media are defined as "seismic systems." Other Category I SSCs that are not designated as "seismic systems" should be considered as "seismic subsystems."		
	B. Decoupling Criteria for Subsystems. It can be shown, in general, that frequencies of systems and subsystems have a negligible effect on the error due to decoupling. It can be shown that the mass ratio, R_m, and the frequency ratio, R_f, govern the results where R_m and R_f are defined as:		
	$R_m = \frac{\text{Total mass of the supported subsystem}}{\text{Total mass of the supporting system}}$ $R_f = \frac{\text{Fundamental frequency of the supported subsystem}}{\text{Dominant frequency of the support motion}}$		

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	The following criteria are acceptable:		
	i If Rm < 0.01, decoupling can be done for any Rf.		
	ii If 0.01 < Rm < 0.1, decoupling can be done for any Rf.		
	iii If Rm > 0.1, a subsystem model should be included in the primary system model.		
	If the subsystem is rigid compared to the supporting system, and also is rigidly connected to the supporting system, it is sufficient to include only the mass of the subsystem at the support point in the primary system model. On the other hand, in case of a subsystem supported by very flexible connections, e.g., pipe supported by hangers, the subsystem need not be included in the primary model. In most cases, the equipment and components, which come under the definition of subsystems, are analyzed (or tested) as a decoupled system from the primary structure and the seismic input for the former is obtained by the analysis of the latter. One important exception to this procedure is the reactor coolant system, which is considered a subsystem but is usually analyzed using a coupled model of the reactor coolant system and primary structure.		
	C. Modeling of Structures. Two types of structural models are widely used by the nuclear industry: lumped-mass stick model and finite element model. Either of these two types of modeling techniques is acceptable if the following guidelines are met:		
	i Lumped-Mass Stick Model For a lumped-mass model, the eccentricities between the centroid (the		

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	neutral axis for axial and bending deformation), the center of rigidity (the neutral axis for shear and torsional deformation), and the center of mass of structures should be included in the seismic model. For selecting an adequate number of discrete mass degrees of freedom in the dynamic modeling to determine the response of all seismic Category I and applicable non-seismic I structures, the acceptance criteria given in Subsection II.1.a.iv of this SRP section are acceptable.		
	ii Finite Element Model The type of finite element used for modeling a structural system should depend on the structural details, the purpose of the analysis, and the theoretical formulation upon which the element is based. The mathematical discretization of the structure should consider the effect of element size, shape, and aspect ratio on solution accuracy. The element mesh size should be selected on the basis that further refinement has only a negligible effect on the solution results.		
	iii In developing either a lumped-mass stick model or a finite element model for dynamic response, it is necessary to consider that local regions of the structure, such as individual floor slabs or walls, may have fundamental vibration modes that can be excited by the dynamic seismic loading. These local vibration modes should be adequately represented in the dynamic response model, in order to ensure that the in-structure response spectra include the additional amplification. Also, the additional seismic loading on the overall structure and on the local region is needed for detailed structural design. In general, three-dimensional models should be used for seismic analyses. However, simpler models can be used if justification can		

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	be provided that the coupling effects of those degrees of freedom that are omitted from the three-dimensional models are not significant.		
	D. Representation of Floor Loads, Live Loads, and Major Equipment in Dynamic Model. In addition to the structural mass, mass equivalent to a floor load of 50 pounds per square foot should be included, to represent miscellaneous dead weights such as minor equipment, piping, and raceways. Also, mass equivalent to 25 percent of the floor design live load and 75 percent of the roof design snow load, as applicable, should be included. The mass of major equipment should be distributed over a representative floor area or included as concentrated lumped masses at the equipment locations.		
	E. Special Consideration for Dynamic Modeling of Structures. It has been common practice that the dynamic model used to predict the seismic response of a structure is not as detailed as the structural model used for the detailed design analysis of all applicable load combinations. Therefore, a methodology is needed to transfer the seismic response loads determined from the dynamic model to the structural model used for the detailed design analysis of all applicable load combinations. This is reviewed for technical adequacy on a case-by-case basis.		
3.7.2-SAC-04	Soil-Structure Interaction. A complete SSI analysis should properly account for all effects due to kinematic and inertial interaction for surface or embedded structures. Any analysis method based on either a direct approach or a substructure approach can be used provided the following conditions are met:	Y	3.7.2.4
	A. The structure, foundation, and soil are properly modeled to ensure that the		

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	results of analyses properly capture spatial variation of ground motion, three dimensional effects of radiation damping and soil layering, as well as nonlinear effects from site response analyses.		
	B. The design earthquake ground motions used as input to the SSI analyses should be consistent with the design response spectra as defined in **SRP Section 3.7.1**.		
	It is noted that there is enough confidence in the current methods used to perform the SSI analysis to capture the basic phenomenon and provide adequate design information; however, the confidence in the ability to implement these methodologies is uncertain. Therefore, in order to ensure proper implementation, the following considerations should be addressed in performing SSI analysis:		
	A. Perform sensitivity studies to identify important parameters (e.g., potential separation and sliding of soil from sidewalls, non-symmetry of embedment, location of boundaries) and to assist in judging the adequacy of the final results. These sensitivity studies can be performed by the use of well-founded and properly substantiated simple models to give better insight;		
	B. Through the use of some appropriate benchmark problems, the user should demonstrate its capability to properly implement any SSI methodologies; and		
	C. Perform enough parametric studies with the proper variation of parameters (e.g., soil properties) to address the uncertainties (as applicable to the given site) discussed in **subsection I.4 of this SRP section**.		
	For sites where SSI effects are considered insignificant and fixed base analyses of structures are performed, bases and justification for not performing SSI analyses are reviewed on a case-by-case basis. If the SSI analysis is not required, the input		

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	motion at the base of the structures will be the design motion reviewed in **SRP Section 3.7.1**. The acceptance criteria for the constituent parts of the entire SSI system are summarized as follows:		
	A. Modeling of Structure. The acceptance criteria given under **subsection II.3 of this SRP section** are applicable.		
	B. Modeling of Supporting Soil. The effect of embedment of structure, groundwater effects, and the layering effect of soil should be accounted for. For the half-space modeling of the soil media, the lumped parameter (soil spring) method and the compliance function methods are acceptable provided that frequency variations and layering effects are incorporated. For the method of modeling soil media with finite boundaries, all boundaries should be properly simulated and the use of types of boundaries should be justified and reviewed on a case-by-case basis. Finite element and finite difference methods are acceptable methods for discretization of a continuum. The properties used in the SSI analysis should be those that are consistent with soil strains developed in free-field site response analyses.		
	C. Input Ground Motion. The acceptance criteria for generating the input ground motion to be used in the SSI analysis are summarized in the following:		
	i If the design earthquake ground motion is defined from generic response spectral shapes (e.g., **Reg. Guide 1.60** or **NUREG-0098**), the location of the ground motion should be consistent with the properties of the soil		

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		profile. For profiles consisting of competent soil or rock, with relatively uniform variation of properties with depth, the ground motion should be located at the soil surface at the top of the finished grade. For profiles consisting of one or more soft and/or thin soil layers overlaying competent material, the ground motion should be located at an outcrop (real or hypothetical) at the top of the competent material in the vicinity of the site.		
	ii	If the design earthquake ground motion is defined from site-specific evaluations of uniform hazard spectra, the location of the ground motion should be at the ground surface in the free-field. In developing the ground motion at the surface, the potential effects of soft soil layers need to be considered. For sites with soil layers near the surface that will be completely excavated to expose competent material, the ground motion response spectra are specified on an outcrop or a hypothetical outcrop that will exist after excavation. Motions at this hypothetical outcrop should be developed as a free surface motion, not as an in-column motion. Competent material is defined as in-situ material having a minimum shear wave velocity of 1,000 feet/second (fps).		
	iii	When the guidance for SSI analysis presented above is not completely implemented, the spectral amplitude of the acceleration response spectra (horizontal component of motion) in the free field at the foundation depth shall be not less than 60 per cent of the corresponding design response spectra at the finished grade in the free field. When variation in soil properties are considered (as required by the “Specific Guidelines for SSI Analysis” below), the 60 percent limitation may be satisfied using an envelope of the three spectra corresponding to the three soil properties. If		

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	the accompanying rotational components of the input motion are ignored, no reduction is permitted in the horizontal component at the foundation level.		
	Specific Guidelines for SSI Analysis The following specific guidelines are provided here to facilitate the review and draw the attention of reviewers to some important aspects of the SSI analysis. These guidelines are not necessarily requirements for the acceptance of any methodologies or an SSI analysis. • The behavior of soil, though recognized to be nonlinear, can often be approximated by linear techniques. Truly nonlinear analysis is not required unless the comparison of results from large-scale tests or actual earthquakes and analytical results indicate deficiencies that cannot be accounted for in any other manner. The nonlinear soil behavior may be accounted for by the following: - Using equivalent linear soil material properties typically determined from an iterative linear analysis of the free-field soil deposit. This accounts for the primary nonlinearity, or - Performing an iterative linear analysis of the coupled soil-structure system. This accounts for the primary and secondary nonlinearities. In the event the nonlinear analysis is chosen, the results of the nonlinear analysis should be judged on the basis of the linear or equivalent linear analysis (**NUREG/CP-0054**). • Superposition of horizontal and vertical response as determined from separate analyses is acceptable (assuming nonlinear effects are not important)		

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	considering the simple material models now available. • The strain-dependent soil properties (e.g., shear modulus, damping) estimated from analysis of the seismic motion in the free field shall be consistent with the geotechnical information reviewed in **SRP Section 2.5.4**. • For cases using standard plant designs, where the site specific spectra fall below the standard plant design spectra, the SSI evaluations are addressed in the standard plant design. • Enough SSI analyses should be performed so as to account for the effects of the potential variability in the properties of the soils and rock at the site. At least three soil/rock profiles should be considered in these analyses, namely, a best estimate (BE) profile, a lower bound (LB) and an upper bound (UB) profile in the evaluation of SSI effects. The properties of each layer of the site profile are typically defined in terms of its low-strain shear modulus and strain-dependent modulus degradation and strain-dependent hysteretic damping properties. These may be determined from dynamic laboratory testing of the site materials, information obtained from the published literature, or both. The set of properties appropriate for a given soil is reviewed for its adequacy. For a particular site, the iterated shear modulus and damping values are typically determined from the results of a number of free-field site response analyses, which are intended to account for the effects of the site-specific design ground motions as well as the site nonlinear properties. If only a single site response calculation is performed, with the low strain property of each material layer selected at its BE value, the resulting iterated property is then determined. The upper and lower bound values of soil/rock shear modulus (G) can then be defined in terms of their best estimate values as:		

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	$G_{LB} = G_{BE} / (1+COV)$ $G_{UB} = G_{BE} x (1+COV)$ where COV is the coefficient of variation considered appropriate for the site materials. The corresponding damping properties should be defined at the compatible strains associated with the shear moduli. If many site response calculations are performed (30 to 60 site response calculations) using Monte Carlo techniques to develop site properties, these calculations are typically used to determine the BE, LB and UB iterated site properties. The BE properties are determined from the mean of the resulting properties and the UB and LB values selected from the +/- one sigma values. A sufficient number of site response calculations need to be performed, to ensure that a stable value of sigma for each material of the profile is obtained. For well-investigated sites (see **RGs 1.132 and 1.138)**, the COV should be no less than 0.5. For sites that are not well investigated, the COV for shear modulus shall be at least 1.0. These COV requirements apply to the "single site response calculation", as well as the "many site response calculations" described above. In no case should the lower bound shear modulus be less than that value consistent with standard foundation analysis that yields foundation settlement under static loads exceeding design allowables. The upper bound shear modulus should not be less than the best estimate shear modulus defined at low strain and as determined from the geophysical testing program. In no case should the material soil damping as expressed by the hysteretic damping ratio exceed 15 percent (**NUREG/CR-1161**). For the case of analyses using generic broad-banded ground motion spectra,		

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	the best estimate shear modulus and damping of each material of the site profile can be defined in terms of its low strain values. The upper and low bound shear moduli can then be defined at twice and one-half the best estimate values, with damping maintained at its low strain value. Alternate approaches can be reviewed on a case-by-case basis. • For dipping soil and rock strata, it is necessary to account for the coupling between the horizontal and vertical degrees of freedom in the stiffness and free-field seismic motion definitions. Also, there may be sites where the reactor building or a seismic Category I structure may have an embedded foundation close to an embankment or a natural slope that preclude the assumption of uniform foundation condition. For such sites, modeling and analysis techniques are reviewed on a case-by-case basis. • Finite Boundary Modeling or Direct Solution Technique The direct solution method is characterized as follows: - Each analysis of the soil and structures is performed in one step. - Finite element or finite difference discrete methods of analysis are used to spatially discretize the soil-structure system. - Definition of the motion along the boundaries of the model (bottom and sides) is either known, assumed, or computed as a precondition of the analysis. Dynamic analysis can be performed using either frequency-domain (limited to linear analysis) or time-integration methods. The mesh size should be adequate for representing the static stress distribution under the foundation and transmitting the frequency content of interest.		

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	The following limitations should be observed for deep soil sites: - The model depth, generally, should be at least twice the base dimension below the foundation level, which should be verified by parametric studies. - The fundamental frequency of the soil (or backfill) stratum should be well below the structural frequencies of interest. - All structural modes of significance should be included. • Half Space or Substructure Solution Technique The half space or substructure approach generally comprises the following steps: (1) Determine the motion of the massless foundation, including both translational and rotational components. (2) Determine the foundation stiffness in terms of frequency dependent impedance functions. (3) Perform SSI analysis. The procedures, modeling assumptions and analytical bases adopted for performing the half space or substructure analysis, including use of frequency independent soil spring parameters, and the spring and damping coefficients, will be reviewed on a case-by-case basis. • There are advanced analytical methods that are being considered by the nuclear industry (e.g., the effects of incoherent ground motion) to reduce the potential effects of high frequency ground motion input. These might be used when a site acceptability determination is performed as discussed in **subsection II.4 of SRP Section 3.7.1**. If incoherency is used to reduce the high frequency response, the potential effects of increasing other responses		

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	(e.g., overturning and torsional responses) shall be considered. When approved for use by the NRC, via issuance of interim staff guidance, it should be noted that the effects of incoherent ground motion may be considered either at the Design Certification stage, or at the site-specific application stage, but not both. If any advanced analytical methods are utilized, the technical basis and analysis results are subject to detailed review on a case-by-case basis.		
3.7.2-SAC-05	<u>Development of In-Structure Response Spectra</u>. **RG 1.122** describes methods generally acceptable to the staff for developing the two horizontal and the vertical in-structure response spectra (e.g., floor response spectra) from the time history motions resulting from the dynamic analysis of the supporting structure. The topics addressed are:	Y	3.7.2.5
	A. SRSS Combination of the three in-structure response spectra in a given direction(e.g., x direction), developed from the output time histories from separate analyses of the three directions (x, y, z) of input motion. SRSS combination is not applicable, if the three directions of the input motion are applied simultaneously in a single analysis.		
	B. Frequency increments for calculation of spectral accelerations.		
	C. Spectrum smoothing and broadening to account for uncertainty.		
	The guidance in **RG 1.122** is augmented as follows:		
	(1) SRSS combination applies to all cases where the three directions of input motion are analyzed separately. There is no longer a distinction made between symmetric and unsymmetric structures.		

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	(2) The 3 Hz frequency increment in the last row of **RG 1.122, Table 1**, applies up to the highest frequency of interest. This typically will be the PGA frequency of the design ground response spectrum, which in some cases may significantly exceed 33 Hz.		
	(3a) When a single set of three artificial time histories is used as the input motion to the supporting structure, the in-structure response spectra are smoothed and broadened in accordance with the provisions of **RG 1.122**, to account for uncertainty.		
	(3b) When multiple sets of three time histories, derived from actual earthquake records, are used as the input motion to the supporting structure, the multiple sets of in-structure response spectra already account for some of the uncertainty. Therefore, the provisions of **RG 1.122**, to account for uncertainty, do not strictly apply. The use of multiple sets of time histories to generate in-structure response spectra is reviewed and accepted on a case-by-case basis. Particularly, the basis for procedures used to account for uncertainties (by variation of parameters) are evaluated. The same acceptance criteria apply to the in-structure response spectra as apply to the design ground response spectrum, reviewed in **subsection II.I.B of SRP Section 3.7.1**. As an example, if the average of the multiple response spectra generated from the multiple design time histories is used to envelop the design ground response spectrum, then the average of the multiple in-structure response spectra generated from the multiple analyses (each of which used one of the multiple design time histories) are used in design.		

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	An evaluation of the statistical correlation between the input ground response spectrum and the output in-structure response spectra should also be provided.		
	The methods used for direct generation of in-structure response spectra are reviewed and accepted on a case-by-case basis.		
3.7.2-SAC-06	Three Components of Earthquake Motion. **RG 1.92,** describes acceptable methods for combining the responses due to three components of earthquake motion, for both the response spectrum method and the time history method. Use of alternate methods are evaluated on a case-by-case basis for acceptability. When the three components of earthquake motion are applied simultaneously, using a set of three artificial time histories, the statistical independence of the time histories should be demonstrated. See **subsection II.1.B of SRP 3.7.1** for the acceptance criteria to demonstrate statistical independence.	Y	3.7.2.6
3.7.2-SAC-07	Combination of Modal Responses. **RG 1.92,** describes acceptable methods for combination of modal responses, including consideration of closely-spaced modes and high-frequency modes, when the response spectrum method of analysis is used to determine the dynamic response of damped linear systems. Use of alternate methods are evaluated on a case-by-case basis for acceptability. When the modal superposition time history method of analysis is used, modal responses are combined algebraically, at each output time step. In accordance with **RG 1.92**, only modes with natural frequencies less than or equal to the ZPA frequency of the input spectrum are included in the modal superposition time	Y	3.7.2.7

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	history analysis. The contribution of the higher frequency modes to the total response is calculated by the missing mass approach. Since this contribution is in-phase with the input time history, it is treated as one additional modal response, that is scaled by the input time history normalized to the ZPA, and combined algebraically with the modal superposition time history solution at each output time step.		
3.7.2-SAC-08	Interaction of Non-Category I Structures with Category I SSCs. All non-Category I structures should be assessed to determine whether their failure under SSE conditions could impair the integrity of seismic Category I SSCs, or result in incapacitating injury to control room occupants. Each non-Category I structure should meet at least one of the following criteria:	Y	3.7.2.8
	A. The collapse of the non-Category I structure will not cause the non-Category I structure to strike a Category I SSC.		
	B. The collapse of the non-Category I structure will not impair the integrity of seismic Category I SSCs, nor result in incapacitating injury to control room occupants.		
	C. The non-Category I structure will be analyzed and designed to prevent its failure under SSE conditions, such that the margin of safety is equivalent to that of Category I structures.		
	The disposition of each non-Category I structure should be formally documented. For criterion (b), it is necessary to provide the technical basis for the determination that collapse of the non-Category I structure is acceptable. This should include a description of any additional loads imposed on the Category I SSCs and the method used to conclude that these loads are not damaging. Also, any protective		

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	shields installed to prevent direct impact on Category I SSCs should be described.		
3.7.2-SAC-09	Effects of Parameter Variations on Floor Response Spectra. Consideration should be given in the analysis to the effects on floor response spectra (e.g., peak width) of expected variations of structural properties, damping values, soil properties, and SSI. The acceptance criteria for the consideration of the effects of parameter variations are provided in **subsection II.5 of this SRP section**. In addition, for concrete structures, the effect of potential concrete cracking on the structural stiffness should be specifically addressed.	Y	3.7.2.9
3.7.2-SAC-10	Use of Equivalent Vertical Static Factors. The use of equivalent static load factors to calculate vertical response loads for the seismic design of Category I SSCs, in lieu of the use of a vertical seismic system dynamic analysis, is acceptable only if it can be demonstrated that the SSC is rigid in the vertical direction, or the acceptance criteria in **subsection 3.7.2.II.1.b of this SRP section** are satisfied. The criterion for rigidity is that the lowest frequency in the vertical direction is higher than the ZPA frequency of the input ground or in-structure spectrum.	Y	3.7.2.10
3.7.2-SAC-11	Methods Used to Account for Torsional Effects. An acceptable method to account for torsional effects in the seismic analysis of Category I structures is to perform a dynamic analysis that incorporates the torsional degrees of freedom. An acceptable alternative, if properly justified, is the use of static factors to account for torsional accelerations in the seismic design of Category I structures.	Y	3.7.2.11

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	To account for accidental torsion, an additional eccentricity of ± 5 percent of the maximum building dimension shall be assumed for both horizontal directions. The magnitude and location of the two eccentricities is determined separately for each floor elevation.		
3.7.2-SAC-12	<u>Comparison of Responses</u>. If both the time history analysis method and the response spectrum analysis method are used to analyze an SSC, the peak responses obtained from these two methods should be compared, to demonstrate approximate equivalency between the two methods.	Y	3.7.2.12
3.7.2-SAC-13	<u>Analysis Procedure for Damping</u>. Either the composite modal damping approach or the modal synthesis technique can be used to account for element-associated damping. Use of composite modal damping for computing the response of systems with nonclassical modes may lead to unconservative results (Miller, et al., 1985). Therefore, the composite modal damping approach is acceptable provided the composite modal damping is limited to 20 percent. One of the other methods mentioned below is generally applicable if the composite modal damping exceeds 20 percent. A. Time domain analysis using complex modes/frequencies, B. Frequency domain analysis, or C. Direct integration of uncoupled equation of motion. For the composite modal damping approach, two techniques of determining an equivalent modal damping matrix or composite damping matrix are commonly used. They are based on the use of the mass or stiffness as a weighting function in generating the composite modal damping. The formulations lead to:	Y	3.7.2.15

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	$\overline{\beta_j} = \{\phi^T\}[\overline{M}]\{\phi\}$ (1) $\beta_j = \frac{\{\phi\}^T[\overline{K}]\{\phi\}}{K^*}$ (2) where K^* = $\{\varphi\}^T$ [K] $\{\varphi\}$, [K] = assembled stiffness matrix, $\overline{\beta_j}$ = equivalent modal damping ratio of the jth mode, $[\overline{K}], [\overline{M}]$ = the modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix , and $\{\varphi\}$ = jth normalized modal vector. For models that take SSI into account by the lumped soil spring approach, the method defined by equation (2) is acceptable. For fixed base models, either equation (1) or (2) may be used. Other techniques based on modal synthesis have been developed and are particularly useful when more detailed data on the damping characteristics of structural subsystems are available. The modal synthesis analysis procedure consists of (1) extraction of sufficient modes from the structure model, (2) extraction of sufficient modes from the finite element soil model, and (3) performance of a coupled analysis using the modal synthesis		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	technique, which uses the data obtained in steps (1) and (2) with appropriate damping ratios for structure and soil subsystems. This method is based upon satisfaction of displacement compatibility and force equilibrium at the system interfaces and uses subsystem eigenvectors as internal generalized coordinates. This method results in a nonproportional damping matrix for the composite structure, and equations of motion have to be solved by direct integration or by uncoupling them by use of complex eigenvectors. Other techniques for estimating the equivalent modal damping of a SSI model are reviewed on a case-by-case basis.		
3.7.2-SAC-14	<u>Determination of Seismic Overturning Moments and Sliding Forces for Seismic Category I Structures</u>. To be acceptable, the determination of the design overturning moment and sliding force should incorporate the following items: A. Three components of input motion. B. Conservative consideration of the simultaneous action of vertical and horizontal seismic forces. Additional information on load combinations is provided in **SRP Section 3.8.5**.	Y	3.7.2.14
SRP 3.7.3	**Seismic Subsystem Analysis (R3, 03/2007)**		
3.7.3-AC-01	**10 CFR Part 50, General Design Criterion (GDC) 2 -** The design basis shall reflect appropriate consideration of the most severe earthquakes reported to have affected the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data have been	N/A-COL	3.7.3

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	accumulated		
3.7.3-AC-02	**10 CFR Part 100, Subpart A**, which is applicable to power reactor site applications before January 10, 1997, refers to Appendix A of this part for seismic criteria.**10 CFR Part 100, Appendix A** indicates that the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) shall be considered in the design of safety-related structures, systems, and components (SSCs). 10 CFR Part 100, Appendix A, further states that the design used to ensure that the required safety functions are maintained during and after the vibratory ground motion associated with the SSE shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that SSCs can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism. **10 CFR Part 100, Subpart B**, which is applicable to power reactor site applications on or after January 10, 1997, refers to **10 CFR 100.23** of this part for seismic criteria.10 CFR 100.23 describes the criteria and nature of investigations required to obtain the geologic and seismic data necessary to determine the suitability of the proposed site and the plant design bases. 10 CFR 100.23 also indicates that applications to engineering design are contained in **10 CFR Part 50, Appendix S**.	N/A-COL	3.7.3
3.7.3-AC-03	**10 CFR Part 50, Appendix S**, is applicable to applications for a design certification or combined license to **10 CFR Part 52** or a construction permit or operating license pursuant to 10 CFR Part 50 on or after January 10, 1997. For SSE ground motions, SSCs will remain functional and within applicable stress, strain, and deformation limits. The required safety functions of SSCs must be assured during and after the vibratory ground motion through design, testing, or	Y	3.7.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	qualification methods. The evaluation must take into account soil-structure interaction effects and the expected duration of the vibratory motion. If the OBE is set at one-third or less of the SSE, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the applicable stress, strain, and deformation limits are satisfied.		
3.7.3-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.7.3-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.7.3-SAC-01	Seismic Analysis Methods. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.1**, are applicable	Y	3.7.3.1
3.7.3-SAC-02	Determination of Number of Earthquake Cycles.	Y	3.7.3.2

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	During the plant life at least one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBEs), if applicable, should be assumed. The number of cycles per earthquake should be obtained from the time history used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed. When the OBE is defined as less than one-third the SSE (and therefore the OBE does not need to be considered in design), there may be certain structural elements which still need to be evaluated for fatigue due to the OBE induced stress cycles. In these instances, the guidance for determining the number of earthquake cycles for use in fatigue calculations should be the same as the guidance provided in **SRM for SECY-93-087** dated July 21, 1993 for piping systems. The number of earthquake cycles to consider are two SSE events with 10 maximum stress cycles per event. This is considered to be equivalent to the cyclic load basis of one SSE and five OBEs. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with **Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1987, Appendix D**.		
3.7.3-SAC-03	Procedures Used for Analytical Modeling. The acceptance criteria provided in SRP **Section 3.7.2**, subsection II.3, are applicable.	Y	3.7.3.3
3.7.3-SAC-04	Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and equipment	Y	3.7.3.4

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	should preferably be selected to be less than ½ or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.		
3.7.3-SAC-05	<u>Analysis Procedure for Damping</u>. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.13**, are applicable.	Y	3.7.3.5
3.7.3-SAC-06	<u>Three Components of Earthquake Motion</u>. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.6**, are applicable.	Y	3.7.3.6
3.7.3-SAC-07	<u>Combination of Modal Responses</u>. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.7**, are applicable.	Y	3.7.3.7
3.7.3-SAC-08	<u>Interaction of Other Systems With Seismic Category I Systems</u>. To be acceptable, each non-seismic Category I system should be designed to be isolated from any seismic Category I system by either a constraint or barrier, or should be remotely located with regard to the seismic Category I system. If it is not feasible or practical to isolate the seismic Category I system, adjacent non-seismic Category I systems should be analyzed according to the same seismic criteria as applicable to the seismic Category I system. For non-seismic Category I systems attached to seismic Category I systems, the dynamic effects of the non-seismic Category I systems should be simulated in the modeling of the seismic Category I	Y	3.7.3.8

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	system. The attached non-seismic Category I systems, up to the first anchor beyond the interface, should also be designed in such a manner that during an earthquake of SSE intensity it will not cause a failure of the seismic Category I system. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.8**, are applicable to all seismic Category I SSCs at the system and subsystem level.		
3.7.3-SAC-09	Multiply-Supported Equipment and Components With Distinct Inputs. Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different. A conservative and acceptable approach for analyzing equipment items supported at two or more locations is to define a uniform response spectrum (URS) that envelopes all of the individual response spectra at the various support locations. The URS is applied at all locations to calculate the maximum inertial responses of the equipment. This is referred to as the uniform support motion (USM) method. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures are acceptable for this purpose. The maximum relative support displacements can be obtained from the building structural response calculations. The support displacements can then be imposed on the supported equipment in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method. The URS method described above can result in considerable overestimation of seismic responses. In the case of multiply- supported equipment in a single	Y	3.7.3.9

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	structure and/or spanning between structures, an alternate method that can be used is the independent support motion (ISM) approach. Guidance and criteria for the use of the ISM method is given in **NUREG-1061, Section 2, Volume 4**.. If the ISM method is utilized, all of the criteria presented in NUREG-1061 related to the ISM method must be followed. In lieu of the response spectrum approach, time histories of support motions may be used as input excitations to the subsystems. The time history approach is considered to provide more realistic results as compared to the USM or ISM methods		
3.7.3-SAC-10	Use of Equivalent Vertical Static Factors. The acceptance criteria provided in **SRP Section 3.7.2, subsection II.10**, are applicable.	Y	3.7.3.10
3.7.3-SAC-11	Torsional Effects of Eccentric Masses. For seismic Category I subsystems, when the torsional effect of an eccentric mass is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for judging the significance will be reviewed on a case-by-case basis.	Y	3.7.3.11
3.7.3-SAC-12	Seismic Category I Buried Piping, Conduits, and Tunnels. For seismic Category I buried piping, conduits, tunnels, and any other subsystems, the following items should be considered in the analysis: A. Two types of groundshaking-induced loadings must be considered for design. i. Relative deformations imposed by seismic waves traveling through the	Y	3.7.3.12

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	surrounding soil or by differential deformations between the soil and anchor points. ii. Lateral earth pressures and ground-water effects acting on structures. B. The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Procedures using the principles of the theory of structures on elastic foundations are acceptable. C. When applicable, the effects due to local soil settlements, soil arching, etc., should also be considered in the analysis. D. Actual methods used for determining the design parameters associated with seismically induced transient relative deformations are reviewed and accepted on a case-by-case basis. Additional information, for guidance purposes only, can be found in **NUREG/CR-1161**, page 26, in American Society of Civil Engineers **(ASCE) Standard 4-98, Section 3.5.2** and in **ASCE Report - Seismic Response of Buried Pipes and Structural Components**.		
3.7.3-SAC-13	Methods for Seismic Analysis of Seismic Category I Concrete Dams. For the analysis of all seismic Category I concrete dams, an appropriate approach that takes into consideration the dynamic nature of forces (due to both horizontal and vertical earthquake loadings), the behavior of the dam material under earthquake loadings, soil-structure interaction (SSI) effects, and nonlinear stress-strain relations for the soil, should be used. Analysis of earthen dams is reviewed under **SRP Section 2.5.5**, “Stability of Slopes.”	N/A-COL	3.7.3.15
3.7.3-SAC-14	Methods for Seismic Analysis of Above-Ground Tanks.	Y	3.7.3.14

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	Most above-ground fluid-containing vertical tanks do not warrant sophisticated, finite element, fluid-structure interaction analyses for seismic loading. However, the commonly used alternative of analyzing such tanks by the "Housner-method" described in TID-7024 may be inadequate in some cases. The major problem is that direct application of this method is consistent with the assumption that the combined fluid-tank system in the horizontal impulsive mode is sufficiently rigid to justify the assumption of a rigid tank. For flat-bottomed tanks mounted directly on their bases, or tanks with very stiff skirt supports, the assumption leads to the usage of a spectral acceleration equal to the zero-period base acceleration. Recent studies (Veletsos (1974 and 1984), Veletsos and Yang (1977), Veletsos and Tang (1989), Haroun and Housner (1981), have shown that for typical tank designs, the frequency for this fundamental horizontal impulsive mode of the tank shell and contained fluid is such that the spectral acceleration may be significantly greater than the zero-period acceleration. Thus, the assumption of a rigid tank could lead to inadequate design loadings. The SSI effects may also be very important for tank responses, and they may need to be considered for both horizontal and vertical motions. The acceptance criteria below are based upon the information contained in **TID-7024 and NUREG/CR-1161**. Additional guidance is provided in **ASCE Standard 4-98, Section 3.5.4**. These references also contain acceptable calculational techniques for the implementation of these criteria. The use of other approaches meeting the intent of these criteria can also be considered if adequate justification is provided. A. A minimum acceptable analysis must incorporate at least two horizontal modes of combined fluid-tank vibration and at least one vertical mode of fluid		

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	vibration. The horizontal response analysis must include at least one impulsive mode in which the response of the tank shell and roof are coupled together with the portion of the fluid contents that moves in unison with the shell. In addition, the fundamental sloshing (convective) mode of the fluid must be included in the horizontal analysis. B. The fundamental natural horizontal impulsive mode of vibration of the fluid-tank system must be estimated giving due consideration to the flexibility of the supporting medium and to any uplifting tendencies for the tank. It is unacceptable to assume a rigid tank unless the assumption can be justified. The horizontal impulsive-mode spectral acceleration, Sa1, is then determined using this frequency and the appropriate damping for the fluid-tank system. Alternatively, the maximum spectral acceleration corresponding to the relevant damping may be used. C. Damping values used to determine the spectral acceleration in the impulsive mode shall be based upon the system damping associated with the tank shell material as well as with the SSI, as specified in **NUREG/CR-1161 and Veletsos and Tang (1989)**. D. In determining the spectral acceleration in the horizontal convective mode, Sa2, the fluid damping ratio shall be 0.5 percent of critical damping unless a higher value can be substantiated by experimental results. E. The maximum overturning moment, M_o, at the base of the tank should be obtained by the modal and spatial combination methods discussed in **subsection II of SRP Section 3.7.2**. The uplift tension resulting from M_o must be resisted either by tying the tank to the foundation with anchor bolts, etc., or by mobilizing enough fluid weight on a thickened base skirt plate. The latter		

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	method of resisting M_o must be shown to be conservative. F. The seismically induced hydrodynamic pressures on the tank shell at any level can be determined by the modal and spatial combination methods in **SRP Section 3.7.2**. The maximum hoop forces in the tank wall must be evaluated with due regard for the contribution of the vertical component of ground shaking. The effects of soil-structure interaction should be considered in this evaluation unless justified otherwise. The hydrodynamic pressure at any level must be added to the hydrostatic pressure at that level to determine the hoop tension in the tank shell. G. Either the tank top head must be located at elevation higher than the slosh height above the top of the fluid or else must be designed for pressures resulting from fluid sloshing against this head. H. At the point of attachment, the tank shell must be designed to withstand the seismic forces imposed by the attached piping. An appropriate analysis must be performed to verify this design. I. The tank foundation (see also **SRP Section 3.8.5**) must be designed to accommodate the seismic forces imposed on it. These forces include the hydrodynamic fluid pressures imposed on the base of the tank as well as the tank shell longitudinal compressive and tensile forces resulting from M_o. J. In addition to the above, a consideration must be given to prevent buckling of tank walls and roof, failure of connecting piping, and sliding of the tank.		
SRP 3.7.4	**Seismic Instrumentation**		
3.7.4-AC-01	**10 CFR Part 20** and **10 CFR Part 50, Appendix S** as they relate to meeting the capabilities and performance of the instrumentation system to adequately measure	Y	3.7.4.1

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	the effects of earthquakes.		
3.7.4-AC-02	**10 CFR Part 20** requires licensees to make every reasonable effort to maintain radiation exposure as low as is reasonably achievable (ALARA).	Y	3.7.4.1
3.7.4-AC-03	**10 CFR Part 50, Appendix S**, requires that suitable instrumentation be provided to promptly evaluate the seismic response of nuclear power plant features important to safety after an earthquake. Appendix S also requires shutdown of the nuclear power plant if vibratory ground motion exceeding that of the OBE occurs.	Y	3.7.4.3
3.7.4-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
3.7.4-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.7.4-SAC-01	Comparison with **RG 1.12**. The seismic instrumentation program is considered to be acceptable if it is in accordance with guidance provided in **RG 1.12**. The bases for elements of the proposed seismic instrumentation program that differ from **RG 1.12** must be provided. This guide recommends installation of solid-state digital	N/A-COL	3.7.4.1

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	time-history accelerographs at appropriate locations in order to provide time history data on the seismic response of the free-field, containment structure, and other Seismic Category I structures. The COL, DC, and construction permit (CP) applicants should provide solid-state digital instrumentation that will enable the processing of data at the plant site within 4 hours of the seismic event. A triaxial time-history accelerograph should be provided at each of the locations specified in **RG 1.12**. Triggering of the free-field or any foundation-level accelerograph should be annunciated in the control room. In addition, applicants should provide a rationale for the placement of instrumentation which is consistent with maintaining occupational radiation exposures ALARA for the location.		
3.7.4-SAC-02	Comparison with **RG 1.166**. The seismic instrumentation program is considered to be acceptable if it contains pre-earthquake planning and post-earthquake actions in accordance with **RG 1.166**. The bases for elements of the proposed seismic instrumentation program that differ from **RG 1.166** must be provided. This guide provides guidance for a timely evaluation after an earthquake of the recorded seismic instrumentation data and for determining whether plant shutdown is required. The COL, DC, and CP applicants should provide a description of both pre-earthquake planning and post-earthquake actions in order to make a rapid determination of the degree of severity of the seismic event. The data from the seismic instrumentation, coupled with information obtained from a plant walkdown, should be used to make the initial determination of whether the plant must be shut down. With regard to the necessary baseline data, information related to seismic	N/A-COL	3.7.4.4

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	instrumentation, including instrument calibration, should be kept at the plant. The applicant's program should also describe the necessary actions, such as selecting equipment and structures for inspections and the content of the baseline inspections, that are to be taken immediately after an earthquake, as described in **RG 1.166**. With regard to the evaluation of ground motion records, the applicant's program should describe data identification (i.e., record collection log), data collection, and record evaluation procedures. Shutdown of the nuclear power plant is required if the vibratory ground motion experienced exceeds that of the OBE. A criterion for determining exceedance of the OBE is provided in the Electric Power Research Institute (EPRI) document **EPRI NP-5930**, "A Criterion for Determining Exceedance of the Operating Basis Earthquake." This criterion is based on a threshold response spectrum ordinate check and a cumulative absolute velocity (CAV) check. The ground motion evaluation should consist of a check on the response spectrum and CAV and a check on the operability of the instrumentation as described in **RG 1.166**. This evaluation should take place within 4 hours of the earthquake.		
SRP 3.8.1	**Concrete Containment (R2, 03/2007)**		
3.8.1-AC-01	**10 CFR 50.55a and GDC 1,** as they relate to concrete containment being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.	Y	3.8.1.2 3.8.1.3 3.8.1.4 3.8.1.5 3.8.1.6
3.8.1-AC-02	**GDC 2**, as it relates to the design of the concrete containment being able to	Y	3.8.1.2

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	withstand the most severe natural phenomena such as winds, tornadoes, floods, and earthquakes and the appropriate combination of all loads.		3.8.1.3 3.8.1.5
3.8.1-AC-03	**GDC 4**, as it relates to the concrete containment being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.	Y	3.8.1.2 3.8.1.3 3.8.1.5
3.8.1-AC-04	**GDC 16**, as it relates to the capability of the concrete containment to act as a leak-tight membrane to prevent the uncontrolled release of radioactive effluents to the environment.	Y	3.8.1.1 3.8.1.2 3.8.1.3 3.8.1.4 3.8.1.5 3.8.1.6
3.8.1-AC-05	**GDC 50**, as it relates to the concrete containment being designed with sufficient margin of safety to accommodate appropriate design loads.	Y	3.8.1.1 3.8.1.3 3.8.1.4 3.8.1.5
3.8.1-AC-06	**10 CFR Part 50, Appendix B** as it relates to the quality assurance criteria for nuclear power plants.	Y	3.8.1.2 3.8.1.6
3.8.1-AC-07	**10 CFR 50.34(f)**, as it relates to demonstrating containment integrity of applicable plants for loads associated with an accidental release of hydrogen generated from metal-water reaction of the fuel cladding, accompanied by hydrogen burning or added pressure from postaccident inerting	Y	3.8.1.3.1 3.8.1.3.2

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3.8.1-AC-08	**10 CFR 50.44**, as it relates to demonstrating the structural integrity of BWRs with Mark III type containments, all PWRs with ice condenser containments, and all containments used in future water-cooled reactors for loads associated with combustible gas generation.	Y	3.8.1.3.1 3.8.1.3.2
3.8.1-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.8.1-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.8.1-SAC-01	Description of the Containment. The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in **Section 3.8.1.1 of RG 1.206**. If the concrete containment has new or unique features that are not specifically covered in **RG 1.206**, the reviewer determines whether the information necessary to accomplish a meaningful review of the structural aspects of these new or unique features is presented.	Y	3.8.1.1

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	RG 1.206 provides the basis for evaluating the description of Seismic Category I structures to be included in a DC or a COL application. **RG 1.70** provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).		
3.8.1-SAC-02	Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of concrete containments are covered by codes, standards, specifications, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable: Codes Title **ASME Code** **Section III, Division 2**, Subsection CC, “Code for Concrete Reactor Vessels and Containments” **ASME Code** **Section XI, Subsection IWL**, “Requirements for Class CC Concrete Components of Light-Water Cooled Plants” **ASME Code** **Section XI, Subsection IWE**, “Requirements for Class MC and Metallic Liners of Class CC Concrete Components of Light-Water Cooled Power Plants” **RG** Title **1.7** Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident **1.35** Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments **1.35.1** Determining Prestressing Forces for Inspection of Prestressed Concrete	Y	3.8.1.2.3 3.8.1.2.5

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	Containments **1.90** Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons **1.91** Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants **1.107** Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures **1.115** Protection Against Low Trajectory Turbine Missiles **1.136** Materials, Construction, and Testing of Concrete Containments		
3.8.1-SAC-03	Loads and Loading Combinations. The specified loads and load combinations are acceptable if found to be in accordance with **Article CC-3000 of the ASME Code** with the exceptions listed below applied to the requirements specified in Table CC-3230-1. **RG 1.136**, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," provides additional guidance for design requirements, including load and load combinations, which should be considered in the design of concrete containments.	Y	3.8.1.2.3 3.8.1.2.5 3.8.1.3
	A. The maximum values of P_a, T_a, R_a, R_{rr}, R_{rj}, and R_{rm} should be applied simultaneously, where appropriate, unless a time-history analysis is performed to justify doing otherwise.	Y	3.8.1.3.1 3.8.1.3.2
	B. Hydrodynamic loads resulting from LOCA and/or SRV actuation should be combined as indicated in the appendix to this SRP section. Fluid structure interaction associated with these hydrodynamic loads and those from	Y	3.8.1.3.1

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	earthquakes should be considered.		
	C. As noted in **Appendix S to 10 CFR Part 50**, the OBE is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the containment remains functional and is within applicable stress, strain, and deformation limits. **SRP Section 3.7** provides further guidance on the use of OBE. When the OBE is defined as less than one-third of the SSE (and therefore the OBE does not need to be considered in design), certain structural elements of the containment (e.g., penetrations or bellows) still need to be evaluated for fatigue resulting from the OBE-induced stress cycles. In these instances, the guidance for determining the number of earthquake cycles for use in fatigue calculations should be the same as the guidance provided in the **staff requirements memorandum (SRM) for SECY-93-087** for piping systems. The number of earthquake cycles to consider is two SSE events with 10 maximum stress cycles per event. Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with **Appendix D of IEEE Standard 344-1987**.	Y	3.8.1.3.1
	D. Where post-LOCA flooding is a design consideration for the plant, the load combination in the ASME Code containing LOCA flooding along with OBE should be considered. Where post-LOCA flooding is combined with the OBE set at one-third or less of the SSE for the plant, this load combination may be	Y	3.8.1.3.1

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	eliminated provided the load combination is shown to be less severe than one of the other load combinations.		
	E. For those plants to which **10 CFR 50.34(f)(3)(v)** applies, the requirements regarding loads and loading combinations include the following: Containment integrity should be maintained by meeting the requirements of **Subarticle CC-3720 of the ASME Code** (considering pressure and dead load alone) during an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding and accompanied by either hydrogen burning or added pressure from postaccident inerting (assuming carbon dioxide is the inerting agent). At a minimum, the ASME Code requirements will be met for a combination of dead load and an internal pressure of 310 Kilo Pascals (KPa) or 45 pounds per square in gauge (psig). The containment structure should be designed against the loadings produced by the inadvertent full actuation of a postaccident inerting hydrogen control system (assuming carbon dioxide), excluding seismic or design-basis accident loadings. Under these conditions, the loadings should not produce strains in the containment liner in excess of the limits established in Subarticle CC-3720 of the ASME Code. The requirements of **Subarticle CC-3720 of the ASME Code** should be met when the containment structure is exposed to the following loading conditions: i. For the factored load category: $D + P_{g1} + [P_{g2} \text{ or } P_{g3}]$ ii. For the service load category, the strains in the containment liner should not exceed the limits set forth in Subarticle CC-3720 when exposed to	Y	3.8.1.3.1 3.8.1.3.2

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	pressure P_{g3}. iii. As a minimum design condition for either condition i or ii above, the following load combination must be satisfied: D + 310 kPa (45 psig) where D = Dead load P_{g1} = Pressure resulting from an accident that releases hydrogen generated from 100-percent metal-water reaction of the fuel cladding P_{g2} = Pressure resulting from uncontrolled hydrogen burning P_{g3} = Pressure resulting from postaccident inerting, assuming carbon dioxide is the inerting agent		
	F. **10 CFR 50.44** requires that an analysis be performed that demonstrates that the containment structural integrity is maintained under loads resulting from combustible gases generated from metal-water reaction of the fuel cladding. An analytical technique accepted by the NRC staff should demonstrate the containment structural integrity. This analysis should include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. **RG 1.7** presents further guidance on the analytical technique, loads, loading combination, and acceptance criteria.	Y	3.8.1.3.1 3.8.1.3.2
	G. Other site-related or plant-related loads applicable to containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures need to be considered. The staff reviews the inclusion of these loads in the factored load combinations on a	N/A-COL	3.8.1.3

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	case-by-case basis.		
	H. The review considers those loads encountered during construction of the containment, which include dead loads, live loads, prestress loads, temperature, wind, earth pressure, snow, rain, and ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) Standard 37 gives additional guidance on construction loads for use in the load combination for construction given in Table CC-3230-1 of the ASME Code. When SEI/ASCE Standard 37 and the ASME Code/SRP provide conflicting criteria, then the ASME Code/SRP should govern.	Y	3.8.1.3.1 3.8.1.3.2
3.8.1-SAC-04	Design and Analysis Procedures. The procedures for design and analysis used for the concrete containment, including the steel liner, are acceptable if found in accordance with those stipulated in **Article CC-3300 of the ASME Code** and **RG 1.136** (see Subsection II.3 of this SRP section). In particular, for the areas of review outlined in Subsection I.4 above, the following procedures are, in general, acceptable:	Y	3.8.1.4
	A. Assumptions on Boundary Conditions. The boundary conditions depend on the methods of analysis to be used and the portions of the containment shell to be separately analyzed. If the analysis is to involve the use of the finite element technique and is to include the foundation media, the boundary would be the demarcation lines separating the foundation mass taken into consideration in the analysis from the surrounding	Y	3.8.1.4.2

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	media. The boundaries of the foundation mass considered should be selected to provide comparable or conservative results to those corresponding to a further extension of the boundaries. This is reviewed on a case-by-case basis. If the analysis considers only the containment shell and its foundation mat, then the bottom of the foundation slab is the boundary of the analytical model. The foundation media should be represented by appropriate soil springs. If separate analyses of the containment shell and the base mat are to be used, it is considered acceptable if strain compatibility of the bottom portion of the shell with the base mat is maintained.		
	B. Axisymmetric and Nonaxisymmetric Loads. Even with the large penetrations and buttresses that may be used in the shell, the overall behavior of the shell has been shown to be axisymmetric under pressure. Therefore, it is acceptable to make such an assumption with respect to the containment geometry. However, for loads such as those induced by wind, tornadoes, earthquakes, and pipe rupture, the analysis should consider the nonaxisymmetric effect of these loads.	Y	3.8.1.4.3
	C. Transient and Localized Loads. During normal operation, a linear temperature gradient across the containment wall thickness may develop. After a LOCA, however, the sudden increase in temperature in the steel liner and the adjacent concrete may produce a nonlinear transient temperature gradient across the containment wall thickness. The analysis should consider the effects of such transient loads. In a PWR ice condenser containment, nonaxisymmetric and transient pressure loads resulting from compartmentalization inside the containment will develop	Y	3.8.1.4.4

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	after a LOCA. For a BWR pressure-suppression containment, the analysis should consider nonaxisymmetric and transient pressure loads resulting from earthquakes, LOCA, and/or SRV actuation (including fluid-structure interaction). For the effects of such localized and transient loads, the overall behavior of the containment structure should first be determined. A portion of the containment shell, within which the localized or transient load is located, should then be analyzed, using the results obtained from the analysis of the overall vessel behavior as boundary conditions.		
	D. Creep, Shrinkage, and Cracking of Concrete. Creep and shrinkage values for concrete should be established by tests performed on the concrete to be used in the containment structure or from data obtained on completed containments constructed of the same kind of concrete. In establishing these values, the analysis should consider the differences in the environment between the test samples and the actual concrete in the structure. For some containments, cracking of concrete is expected to occur based on the structural integrity test performed in accordance with **Article CC-6000 of the ASME Code**. Also, based on load combinations that include the design pressure load with earthquake loads, additional concrete cracking would be expected to occur. Concrete cracking can cause redistribution of member forces because of the various loadings applied to the structure. Concrete cracking can also affect the stiffness of the containment and cause shifting of the natural frequency, thereby affecting the response/loads used to design the containment. Accordingly, the analysis used to calculate the dynamic response of the containment resulting from dynamic loads such as earthquake and	Y	3.8.1.4.5

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	hydrodynamic loads (if applicable) needs to consider the potential effects of concrete cracking, if significant. The approach used should include the effect of redistribution of the various loads caused by concrete cracking. With improvements in the development of computer programs for analysis of concrete structures, the evaluation of concrete cracking can be analyzed directly within the finite element model. Alternatively, additional analyses can treat the effect of concrete cracking by determining the response of the containment to variation in the stiffness characteristics of the containment shell (e.g., shear stiffness and tensile membrane stiffness reduction). As stated in **CC-3320 of the ASME Code**, the effects of reduction in shear stiffness and tensile membrane stiffness resulting from cracking of the concrete should be considered in methods for predicting the maximum strains and deformations of the containment. Thus, concrete cracking needs to be considered depending on the stress levels caused by the most severe seismic load combination. Provide technical justification, if cracking is not considered or is determined to be insignificant. **Sections 3.1.3 and C 3.1.3 of ASCE 4-98** provide additional guidance for modeling the stiffness of concrete elements. The staff reviews the methods used for considering creep, shrinkage, and concrete cracking, or the justification for not considering these effects, on a case-by-case basis.		
	E. Dynamic Soil Pressure. Consideration of dynamic lateral soil pressures on embedded walls of a concrete containment (if applicable) is acceptable if the lateral earth pressure loads are evaluated for two cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure	N/A-OTHER	3.8.1.4.6

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	calculated in accordance with **ASCE 4-98 Section 3.5.3.2** and (2) lateral earth pressure equal to the passive earth pressure. If the above methods are shown to be overly conservative for the cases considered, then any alternative methods proposed will be reviewed on a case-by-case basis.		
	F. Computer Programs. The computer programs used in the design and analysis should be described and validated by any of the following procedures or criteria: i. The computer program is recognized in the public domain and has had sufficient history of use to justify its applicability and validity without further demonstration. ii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained by a similar and independently written and recognized program in the public domain. The test problems should be demonstrated to be similar to or within the range of applicability of the problems analyzed by the public domain computer program. iii. The computer program's solutions to a series of test problems have been demonstrated to be substantially identical to those obtained from classical solutions or from accepted experimental tests or to analytical results published in technical literature. The test problems should be demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program. A summary comparison should be provided for the results obtained in the validation of each computer program.	Y	3.8.1.4.1

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	G. Tangential Shear. Design and analysis procedures for tangential shear are acceptable if in accordance with those contained in **Article CC-3000 of the ASME Code**. The regulatory staff should note the exceptions taken to the provisions of this article, as contained in Subsection II.5 of this SRP section.	Y	3.8.1.4.7
	H. Variation in Physical Material Properties. For the analysis of the effects of possible variations in the physical properties of materials on the analytical results, the upper and lower bounds of these properties should be used, wherever critical. The physical properties that may be critical include the soil modulus, modulus of elasticity, and Poisson's ratio of concrete.	Y	3.8.1.4.8
	I. Thickened Penetrations. The effect of the large, thickened penetration regions on the overall behavior of the containment may be treated by the same method used for localized loads as discussed in Subsection II.4.C.	Y	3.8.1.4.9
	J. Steel Liner Plate and Anchors. For the design and analysis of the liner plate and its anchorage system, the procedures furnished are found adequate and acceptable if in accordance with the provisions of **Subarticle CC-3600 of the ASME Code**. In general, the liner plate analysis should consider deviations in geometry resulting from fabrication and erection tolerances and variations of the assumed physical properties of the liner and anchor material. Since the liner plate is usually anchored at relatively closely spaced intervals, the analysis procedures are acceptable if based on either the classical plate or beam theory. Since the concrete shell is	Y	3.8.1.4.10

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	much stiffer than the liner plate, the strains in the liner will essentially follow those in the concrete. The strains in the concrete under the various load combinations as obtainable from the analysis of the shell are thus imposed on the liner plate, and the resulting strains and stresses in the liner and its anchors should be lower than the allowable limits defined in **Tables CC-3720-1 and CC-3730-1 of the ASME Code**.		
	K. Ultimate Capacity of Concrete Containment. Regulatory criteria require a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure. i. Reinforced Concrete Containments One acceptable methodology for cylindrical reinforced concrete containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1 percent. The specific location of interest is the steel reinforcement in the hoop direction, closest to the inside surface of the concrete. The inside radius of the concrete wall should be used in calculating the strain in the hoop reinforcing steel. To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered for the reinforcing steel. The stress-strain curve for the reinforcing steel should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of the specific grade of reinforcing steel. The stress-strain curve must be developed for the	Y	3.8.1.4.11

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<tr><td></td><td>design-basis accident temperature.
The use of an alternate failure criteria for the analyses of noncylindrical containments and cylindrical containments are reviewed on a case by case basis.
Guidance on computer modeling of reinforced concrete containments for internal pressure capacity calculations appears in NUREG/CR-6906.
NOTE: In applying the analysis methodology to existing containment structures, it is permissible to use as-built material properties for the reinforcing steel and concrete. Sufficient data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for the design-basis accident temperature. For deterministic assessments, the lower-bound values should be used. For probabilistic risk assessment, calculations of failure probability versus pressure should consider the statistical distribution of the material properties.
ii. <u>Prestressed Concrete Containments</u>
One acceptable methodology for cylindrical prestressed concrete containments is to estimate the capacity based on attaining a maximum global membrane strain away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 0.8 percent. This strain limit is applicable to all materials which contribute to resisting the internal pressure (i.e., tendons, rebars, and liner (if considered)). When calculating the pressure capacity contribution from the tendons, the</td><td></td><td></td></tr>
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	above-specified strain limit is applicable to the full range of strain (from 0.0 psi at 0.0-percent strain up to the tendon contribution to pressure capacity at 0.8-percent strain). The other items described previously for reinforced concrete containment, after the first paragraph identifying global strain limits, are also applicable to the approach used for prestressed concrete containments. The criteria presented for consideration of nonlinear material behavior of the reinforcing steel also apply to the tendons. iii. <u>Containment Penetrations</u> The methodologies described above apply to the containment structure. A complete evaluation of the internal pressure capacity must also address major containment penetrations, such as the removable drywell head and ventlines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. The analysis should also address other potential containment leak paths through mechanical and electrical penetrations. iv. <u>Special Considerations for Steel Elliptical and Torispherical Heads:</u> Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. The analysis needs to evaluate this potential mode of failure to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If appropriately		

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	demonstrated, residual postbuckling strength can be considered in determining the pressure capacity. The details of the analysis and the results should be submitted in report form with the following identifiable information: (1) The original design pressure, P_a, as defined in the ASME Code (2) Calculated static pressure capacity (3) Equivalent static pressure response calculated from dynamic pressure (4) The associated failure mode (5) The stress-strain relation of the liner steel and reinforcing and/or prestressing steel and the behavior of the liner under the postulated loading conditions in relation to that of the reinforcing and/or prestressing steel (6) The criteria governing the original design and the criteria used to establish failure (7) Analysis details and general results (8) Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure		
	L. <u>Structural Audit</u>. **Appendix B to SRP Section 3.8.4** describes the conduct of a structural audit.	N/A-INFO	N/A

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	M. Design Report. The design report is considered acceptable when it satisfies the guidelines of **Appendix C to SRP Section 3.8.4**.	Y (for Seismic Category I Structures in Design Report)	3.8.1.4.11
		N/A-COL (for All Seismic Category I Structures)	3.8.1.4.11
3.8.1-SAC-05	Structural Acceptance Criteria		
	A. For the structural portions of the containment, the specified allowable limits for stresses and strains are acceptable if they are in accordance with **Subsection CC-3400 of the ASME Code** and **RG 1.136** (see Subsection II.3 of this SRP section), with the following exceptions: CC-3421.5 For existing (older vintage) plants where a portion of the tangential shear stress, v_c, was permitted to be carried by the concrete, v_c is limited to 276 kPa or 40 pounds per square inch (psi) and 414 kPa (60 psi) for the load combinations of **Table CC-3230-1**, representing abnormal/severe environmental and abnormal/extreme environmental conditions, respectively. The criteria for design of steel reinforcement to resist the excess shear load above v_c should meet the provisions of the code of record for the containment design. For other plants, the concrete should carry no tangential shear stress as indicated in **Subsection CC-3421.5 of the ASME Code**. The tangential shear	Y	3.8.1.5

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	strength provided by orthogonal reinforcement should be limited to the following: $0.833\sqrt{f'_c}\ (\mathrm{MPa}); [10\sqrt{f'_c}\ (\mathrm{psi})]$ where the value of *f ' c* is in units of MPa and psi in the first and second expression, respectively, in accordance with the ASME Code. For prestressed concrete containments, the principal tensile stress should not exceed the following: $\frac{1}{3}\sqrt{f'_c}\ (\mathrm{MPa}); [4\sqrt{f'_c}\ (\mathrm{psi})]$ where the value of *f* '*c* is in units of MPa and psi in the first and second expression, respectively, in accordance with the **ASME Code**.		
	B. For the liner plate and its anchorage system, the specified limits for stresses and strains are acceptable if in accordance with **Tables CC-3720-1 and CC-3730-1 of the ASME Code**, respectively.	Y	3.8.1.4.10
3.8.1-SAC-06	<u>Materials, Quality Control, and Special Construction Techniques</u>		
	A, The specified materials of construction are acceptable if found to be in accordance with Article CC-2000 of the ASME Code with additional guidance	Y	3.8.1.6

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	provided by **RG 1.107 and 1.136**.		
	B. Quality control programs are acceptable if found to be in accordance with applicable portions of **Articles CC-4000 and CC-5000 of the ASME Code** with additional guidance provided by **RG 1.136** for quality assurance requirements.	Y	3.8.1.6
	C. Special construction techniques, if any, are reviewed on a case-by-case basis.	N/A-COL	3.8.1.6.8
3.8.1-SAC-07	Testing and Inservice Surveillance Requirements	ITAAC	Tier 1
	A. Procedures for the postconstruction, preoperational structural proof test proposed for the containment are acceptable if found in accordance with those delineated in **Article CC-6000 of the ASME Code**.		
	B. For reinforced and prestressed concrete containments, **10 CFR 50.55a** imposes the examination requirements of **Section XI, Subsections IWL and IWE, of the ASME Code**. These subsections provide preservice examination, inservice inspection, and repair/replacement requirements, and acceptance criteria. The scope of Subsection IWL includes the concrete and unbonded posttensioning systems. Subsection IWE covers examination requirements for steel liners of concrete containments and their integral attachments; metallic shell portions of containment (e.g., steel head); containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The regulations in **10 CFR 50.55a(b)(2)** specify the acceptable edition of the ASME Code and additional requirements beyond those contained in these subsections of the ASME Code. **10 CFR 55a (b)(2)(viii)(E)** requires that licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.		

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	C. For concrete containments, it is important to accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of containments is essential for plant safety. The staff reviews on a case-by-case basis any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containments. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 ppm, sulfates <1500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.		
	D. For prestressed concrete containments, inservice surveillance requirements for the tendons, as presented in the technical specifications of the operating license, are acceptable if in accordance with **Section XI, Subsection IWL of the ASME Code**; **10 CFR 50.55a**; and **RG 1.35 and 1.35.1** for ungrouted tendons and **1.90** for grouted tendons, respectively.		
	E. **SRP Section 6.2.6** presents the preoperational and inservice integrated leak-rate testing criteria.		

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	F. For new and unique containment designs (e.g., incorporating integrally connected passive systems with pools), the preoperational tests and inspections of containment discussed above need to consider items included in these unique features.		
SRP 3.8.2	**Steel Containment (R2, 03/2007)**		
3.8.2-AC-01	**10 CFR 50.55a and 10 CFR Part 50**, **Appendix A, General Design Criterion (GDC) 1** as they relate to designing, fabricating, erecting, testing, and inspecting steel containments to quality standards commensurate with the importance of the safety function to be performed.	Y	3.8.2.2
3.8.2-AC-02	**GDC 2**, as it relates to designing steel containments to be capable of withstanding the most severe natural phenomena such as winds, tornados, floods, and earthquakes and the appropriate combination of all loads.	Y	3.8.2.3
3.8.2-AC-03	**GDC 4**, as it relates to the capability of steel containments to withstand the dynamic effects of equipment failures, including missiles, pipe whipping, and blowdown loads associated with LOCAs.	Y	3.8.2.3
3.8.2-AC-04	**GDC 16**, as it relates to the capability of the steel containment to act as a leaktight membrane to prevent the uncontrolled release of radioactive effluents to the environment.	Y	3.8.2.1 3.8.2.2 3.8.2.3
3.8.2-AC-05	**GDC 50**, as it relates to designing steel containments with sufficient margin of safety to accommodate appropriate design loads.	Y	3.8.2.2 3.8.2.3 3.8.2.5
3.8.2-AC-06	**10 CFR 50.34(f)**, as it relates to the capability of the steel containment of specific	Y	3.8.2.3.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	identified plants to resist (1) those loads that are generated by pressure and dead loads during an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction and accompanied by either hydrogen burning or added pressure from postaccident inerting, and (2) those loads that are generated as a result of an inadvertent full actuation of a postaccident inerting hydrogen control system, excluding seismic or design-basis accident loadings		3.8.2.3.2
3.8.2-AC-07	**10 CFR 50.44**, as it relates to the capability of the steel containment of existing plants and new plants to resist those loads associated with combustible gas generation from a metal-water reaction of the fuel cladding.	Y	3.8.2.3.1 3.8.2.3.2
3.8.2-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.8.2-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.8.2-SAC-01	Description of the Containment. The descriptive information in the safety analysis report (SAR) is acceptable if it	Y	3.8.2.6

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	meets the criteria set forth in **Section 3.8.2.1 of RG 1.206**. If the steel containment has new or unique features that **RG 1.206** does not specifically cover, adequate information necessary to accomplish a meaningful review of the structural aspects of these new or unique features need to be presented such that an evaluation can be made that it is equivalent in function and complies with the applicable requirements. **RG 1.206** provides the basis for evaluating the description of structures to be included in a DC or a COL application. **RG 1.70** provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).		
3.8.2-SAC-02	Applicable Codes, Standards, and Specifications. Codes, standards, and specifications, acceptable either in their entirety or in part, cover the design, materials, fabrication, erection, inspection, testing, and inservice surveillance of steel containments. The following codes and guides are acceptable: Code/Guide — Title **ASME Code** — **Section III, Division 1, Subsection NE**, “Class MC Components” **ASME Code** — **Section XI, Subsection IWE**, “Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants” **RG 1.7** — Control of Combustible Gas Concentrations in Containment **RG 1.57** — Design Limits and Loading Combinations for Metal Primary Reactor	Y	3.8.2.2.3 3.8.2.2.5

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	Containment System Components		
3.8.2-SAC-03	Loads and Loading Combinations.		
	A. Currently, **ASME Code, Section III, Division 1, Subsection NE**, and **RG 1.57** do not explicitly state the loads and load combinations that should be considered in the design of steel containments. The staff has issued as a proposed revision to **RG 1.57**, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.” This draft guide or subsequent revision to **RG 1.57** provides additional guidance for design requirements, including load and load combinations, which should be considered in the design of steel containments. The specified loads and load combinations are acceptable if found to be in accordance with the following: A. Loads D — Dead loads L — Live loads, including all loads resulting from platform flexibility and deformation and from crane loading, if applicable P_t — Test pressure T_t — Test temperature T_o — Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition R_o — Pipe reactions during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state	Y	3.8.2.2 3.8.2.3

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	condition P_o —External pressure loads resulting from pressure variation either inside or outside containment E — Loads generated by the OBE, including sloshing effects, if applicable E' — Loads generated by the SSE, including sloshing effects, if applicable P_a —Pressure load generated by the postulated pipe break accident (including pressure generated by postulated small-break or intermediate-break pipe ruptures), pool swell, and subsequent hydrodynamic loads Note: For loading combinations B, Service Conditions (iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. T_a —Thermal loads under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic reaction loads Note: For loading combinations B, Service Conditions (iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. R_a —Pipe reactions under thermal conditions generated by the postulated pipe break accident, pool swell, and subsequent		

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	hydrodynamic reaction loads Note: For loading combinations B, Service Conditions (iii), for (1)(d), (3)(c), and (4)(b), a small or intermediate pipe break accident is postulated; for all other load combinations, the design-basis LOCA is postulated. P_s —All pressure loads that are caused by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic loads, if applicable T_s —All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads, if applicable R_s —All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads, if applicable Y_r — Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident Y_j — Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident Y_m — Missile impact equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping F_L — Load generated by the post-LOCA flooding of the containment, if applicable P_{g1} — Pressure load generated from 100-percent fuel clad metal-water reaction		

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	P_{g2} — Pressure loads generated by hydrogen burning, if applicable P_{g3} — Pressure load from postaccident inerting, assuming carbon dioxide is the inerting agent, if applicable		
	B. Loading Combinations The loading combinations for which the containment might be designed or subjected to during the expected life of the plant include the following: i. Testing Condition This includes the testing condition of the containment to verify its leak integrity. The loading combination in this case includes— $D + L + T_t + P_t$ ii. Design Conditions These include all design loadings for which the containment vessel or portions thereof might be designed during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design-basis LOCA. The loading combination in this case includes— $D + L + P_a + T_a + R_a$ iii. Service Conditions The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits, and the postflooding condition. The loads may be combined by their actual time history of occurrence taking into consideration their dynamic effect upon the structure. (1) Level A Service Limits	Y	3.8.2.3

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	These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, except only those categorized as Level B, Level C, Level D, or testing loadings. The loading combinations corresponding to these limits include the following: (a) Normal operating plant condition $D + L + T_o + R_o + P_o$ (b) Operating plant condition in conjunction with the actuation of multiple SRVs $D + L + T_s + R_s + P_s$ (c) Design-basis LOCA $D + L + T_a + R_a + P_a$ (d) Multiple SRV actuations in combination with small- or intermediate-break accident $D + L + T_a + R_a + P_a + T_s + R_s + P_s$ (e) Normal operating plant conditions in combination with inadvertent full actuation of a postaccident inerting hydrogen control system (**10 CFR 50.34(f)(3)(v)(B)(1)**) $D + L + T_o + R_o + P_o + P_{g3}$ (f) Pressure test load to ensure that the containment will safely withstand the pressure calculated to result from carbon-dioxide inerting (**10 CFR 50.34(f)(3)(v)(B)(2)**)		

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	$D + 1.10 \times P_{g3}$ (2) Level B Service Limits These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena during which the plant must remain operational. The loading combinations corresponding to these limits include the following: (a) Design-basis LOCA in combination with OBE (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered) D + L + Ta + Ra + Pa + E (b) Operating plant condition in combination with OBE (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered) D + L + To + Ro + Po + E (c) Operating plant condition in combination with OBE and multiple SRV actuations (if E ≤ one-third E', only its contribution to cyclic loading needs to be considered) D + L + Ts + Rs + Ps + E (d) Design-basis LOCA in combination with a single active component failure causing one SRV discharge $D + L + T_a + P_a + R_a + T_s + R_s + P_s$ (3) Level C Service Limits These service limits include the loads subject to Level A service limits plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations		

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	corresponding to these limits include the following: (a) Design-basis LOCA in combination with SSE D + L + Ta + Ra + Pa + E' (b) Operating plant condition in combination with SSE D + L + To + Ro + Po + E' (c) Multiple SRV actuations in combination with small- or intermediate-break accident and SSE $D + L + T_a + R_a + P_a + T_s + R_s + P_s + E'$ (d) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by hydrogen burning (**10 CFR 50.34(f)(3)(v)(A)(1)**, **10 CFR 50.44**) $D + Pg1 + P_{g2}$ Note: In this load combination, $P_{g1} + P_{g2}$ should not be less than 310 kilo Pascals (kPa) or 45 pounds per square in gauge (psig). (e) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100-percent fuel clad metal-water reaction accompanied by the added pressure from postaccident inerting, assuming carbon dioxide as the inerting agent (**10 CFR 50.34(f)(3)(v)(A)(1)**) $D + P_{g1} + P_{g3}$ Note: In this load combination, $P_{g1} + P_{g3}$ should not be less than 310 kPa or (45 psig).		

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	(4) Level D Service Limits These service limits include other applicable service limits and loadings of a local dynamic nature for which the containment function is required. The load combinations corresponding to these limits include the following: (a) Design-basis LOCA in combination with SSE and local dynamic loadings $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$ (b) Multiple SRV actuations in combination with small- or intermediate-break accident, SSE, and local dynamic loadings $D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + P_s + T_s + R_s + E'$ (5) Postflooding Condition This includes the post-LOCA flooding of the containment in combination with OBE-basis earthquake $D + L + F_L + E$		
	C. Construction Loads Temporary construction loads and the effects of environmental loads during the construction stage need to be considered. **ASME Code, Section III, Subsection NE**, does not address this. The sections of Structural Engineering **Institute/American Society of Civil Engineers (SEI/ASCE) Standard 37-02** pertaining to steel structures may be used for guidance.	Y	3.8.2.3.1
	D. External Environmental Loads A concrete shield building typically protects steel containments from the	Y	3.8.2.3.1

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	environment. If environmental loads external to the steel containment (e.g., wind, tornado, external flooding) either directly or indirectly impose loads on the steel containment, the design of the steel containment also needs to consider these loads. Load combinations and acceptance criteria that are consistent with those specified in **SRP Section 3.8.1** for concrete containments should be used.		
	As noted in **10 CFR 50, Appendix S**, the OBE is only associated with plant shutdown and inspection, unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, explicit analysis is not required. The only exceptions are the postflooding condition and cyclic loading considerations. The staff requirements memorandum for **SECY-93-087** provides guidance on the treatment of cyclic loading for the OBE. If the OBE is set at a value greater than one-third of the SSE, explicit analysis must be performed to demonstrate that the applicable load combinations meet the Service Level B stress, strain, deformation, and fatigue limits.		
3.8.2-SAC-04	Design and Analysis Procedures. **Article NE-3000 of ASME Code, Section III, Division 1, Subsection NE**, covers design and analysis procedures for steel containments. The procedures given in the ASME Code, with additional guidance provided in the applicable provisions of **RG 1.57**, constitute an acceptable basis for design and analysis. Moreover, for the specific areas of review described in Subsection I.4 of this SRP section, the following criteria are acceptable:	Y	3.8.2.3 3.8.2.4
	A. Treatment of Nonaxisymmetric and Localized Loads For most containments, the nonaxisymmetric loads that apply are the	Y	3.8.2

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	horizontal seismic and associated sloshing loads, pool swell, and its related hydrodynamic loads caused either by LOCA or by SRV actuation. Other possible nonaxisymmetric and localized loads are those induced by pipe rupture, such as reactions, jet impingement forces, and missiles. For the PWR ice-condenser containment, the design-basis accident may result in a nonaxisymmetric pressure load caused by compartmentation of the containment interior. For such localized loads, the analyses should include a determination of the local effects of the loads. These effects should then be superimposed on the overall effects. For the overall effects of nonaxisymmetric loads on shells of revolution, an acceptable general procedure is to expand the load by a Fourier series. Any other applicable methods proposed for a large thin shell, will be reviewed on a case-by-case basis.		
	B. Treatment of Buckling Effects Earthquake loads and localized pressure loads (such as those encountered in PWR ice-condenser containments) require consideration of shell buckling. An acceptable approach to the problem is to perform a nonlinear dynamic analysis. If a static analysis is performed, an appropriate dynamic load factor should be used to obtain the effective static load. **Subarticle NE-3133 of ASME Code, Section III, Division 1, Subsection NE**, is acceptable to address buckling of shell geometries and loadings covered therein. Buckling of shells with more complex geometries or loading conditions than those covered by Subarticle NE-3133 may be considered in accordance with the criteria described in **ASME Code Case N-284, Revision 1**, with additional guidance provided in **RG 1.193**. Each application of ASME Code	Y	3.8.2

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	Case N-284, Revision 1, is subject to review on a case by case basis. Buckling of shells under internal pressure (e.g., torispherical heads) may also be considered in accordance with the criteria described in **ASME Code Case N-284, Revision 1**, with guidance provided in **RG 1.193**. Each application of ASME Code Case N-284, Revision 1, is subject to review on a case by case basis. The staff will review the use of alternate methodologies to address the buckling of steel containments on a case-by-case basis. **RG 1.84**, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" and **RG 1.193**, " Code cases not approved for Use," provide additional guidance for code case acceptability which should be considered in the design of steel containments. Any Code cases not currently approved by NRC requires review on a case by case basis.		
	C. Computer Programs The computer programs used in the design and analysis should be described and validated by procedures or criteria described in Subsection II.4.e of **SRP Section 3.8.1**.	Y	3.8.2
	D. Ultimate Capacity of Steel Containment For new reactors, regulatory criteria require a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure. One methodology acceptable to the staff for cylindrical steel containments is to estimate the capacity based on attaining a maximum global membrane strain	Y	3.8.2

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<tr><th>SRP Criterion</th><th>Description
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<tr><td></td><td>away from discontinuities (i.e., the hoop membrane strain in a cylinder) of 1.5 percent.
To conduct the necessary analysis, both nonlinear material behavior and nonlinear geometric behavior must be considered. The stress-strain curve for the steel containment material should be based on the code-specified minimum yield strength and a stress-strain relationship above yield that is representative of that specific grade of steel. The stress-strain curve must be developed for the design-basis accident temperature.
Analyses of noncylindrical containments and analyses of cylindrical containments that use alternate failure criteria will be subject to detailed staff review, on a case-by-case basis.
The NRC has published guidance on computer modeling of steel containments for internal pressure capacity calculations in **NUREG/CR-6906**.
Note: In applying the analysis methodology to existing containment structures, it is permissible to use as-built material properties for the steel containment material. Sufficient material certification data must be available to establish with reasonable confidence a lower bound, a median, and an upper bound value for the important material parameters. These values must be adjusted for the design-basis accident temperature. For deterministic assessments, the lower bound values should be used. For probabilistic risk assessment, calculations of failure probability vs. pressure should consider the statistical distribution of the material properties.
Containment Penetrations:</td><td></td><td></td></tr>
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	The methodology described above applies to the containment structure. A complete evaluation of the internal pressure capacity must also address major containment penetrations, such as the removable drywell head and ventlines for BWR designs, equipment hatches, personnel airlocks, and major piping penetrations. Other potential containment leak paths through mechanical and electrical penetrations should also be addressed. Special Considerations for Steel Ellipsoidal and Torispherical Heads: Under internal pressure, a potential failure mode of steel ellipsoidal and torispherical heads is buckling, resulting from a hoop compression zone in the knuckle region. This potential mode of failure needs to be evaluated, to determine if it is the limiting condition for the pressure capacity of the containment. The analysis should consider nonlinear material and geometric behavior and address the effect of initial geometric imperfections either explicitly (direct modeling) or implicitly (through the use of appropriate imperfection sensitivity knockdown factors). If appropriately demonstrated, residual postbuckling strength can be considered in determining the pressure capacity. The details of the analysis and the results should be submitted in a report form with the following identifiable information: i. Original design pressure, P, as defined in **ASME Code, Section III, Division 1, Subsection NE, Subarticle NE-3112.1** ii. Calculated static pressure capacity iii. Equivalent static pressure response calculated from dynamic pressure iv. Associated failure mode		

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	v. Criteria governing the original design and the criteria used to establish failure vi. Analysis details and general results vii. Appropriate engineering drawings adequate to allow verification of modeling and evaluation of analyses employed for the containment structure		
	E. Structural Audit Structural audits are conducted as described in **SRP Section 3.8.4, Appendix B**.	N/A-INFO	N/A
	F. Design Report The design report is considered acceptable when it satisfies the guidelines provided in **SRP Section 3.8.4, Appendix C**.	Y (for Seismic Category I Structures in Design Report)	3.8.2.4.3
		N/A-COL (for All Seismic Category I Structures)	3.8.2.4.3
3.8.2-SAC-05	<u>Structural Acceptance Criteria</u>. Stresses at various locations of the shell of the containment for various design loads are determined by analysis. Total stresses for the combination of loads delineated in Subsection II.3 of this SRP section are acceptable if found to be within the limits defined by **ASME Code, Section III, Division 1, Subsection NE, Subarticles NE-3221.1, NE-3221.2, NE-3221.3, and NE-3221.4** for Service Levels A, B, C, and D, respectively.	Y	3.8.2.5

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	For the postflooding load combination (Subsection II.3.b(iii)(5)), Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress. Evaluation of primary plus secondary plus peak stress is not required. If external environmental loads need to be considered in the steel containment design, the staff will review the adequacy of the approach and acceptance criteria on a case-by-case basis.		
3.8.2-SAC-06	Materials, Quality Control, and Special Construction Techniques.		
	A. The materials of construction are acceptable if in accordance with **Article NE-2000 of ASME Code, Section III, Division 1, Subsection NE**. The organization responsible to review material properties will review corrosion protection.	Y	3.8.2.6
	B. Quality control programs are acceptable if in accordance with **Articles NE-2000, NE-4000, and NE-5000 of ASME Code, Section III, Division 1, Subsection NE**.	Y	3.8.2.6
	C. The acceptability of special construction techniques, if any, are evaluated on a case-by-case basis.	N/A-OTHER	N/A
	D. The staff will review the consideration of temporary construction loads and the effects of environmental loads during the construction stage on a case-by-case basis.	Y	3.8.2.3.1
3.8.2-SAC-07	Testing and Inservice Surveillance Requirements.	ITAAC	Tier 1
	A. Procedures for the preoperational structural proof test are acceptable if the procedures are in accordance with **Article NE-6000 of ASME Code, Section**		

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	III, Division 1, Subsection NE.		
	B. For steel containments, **10 CFR 50.55a** requires examination be conducted as outlined in **ASME Code Section XI, Subsection IWE**. Subsection IWE provides preservice examination, inservice inspection, and repair/replacement requirements and corresponding acceptance criteria. The scope of Subsection IWE includes the steel containment shell; integral attachments; containment hatches and airlocks; seals, gaskets, and moisture barriers; and pressure-retaining bolting. **10 CFR 50.55a(b)(2)** specifies the acceptable edition of the ASME Code and additional requirements beyond those contained in Subsection IWE. **10 CFR 55a (b)(2)(viii)(E)** requires that licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.		
	C. The staff will review any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of the steel containment on a case-by-case basis.		
SRP 3.8.3	**Concrete and Steel Internal Structures of Steel or Concrete Containments (R2, 03/2007)**		
3.8.3-AC-01	**10 CFR 50.55a and 10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, as they relate to the design, fabrication, erection, and testing of containment internal structures in accordance with quality standards commensurate with the importance of the safety function to be performed.	Y	3.8.3.2

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3.8.3-AC-02	**GDC 2**, as it relates to the ability of the containment internal structures without loss of capability to perform their safety function, to withstand the effects of natural phenomena, such as earthquakes, tornadoes, floods, and the appropriate combination of all loads.	Y	3.8.3.2 3.8.3.3
3.8.3-AC-03	**GDC 4**, as it relates to the protection of containment internal structures against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.	Y	3.8.3.3
3.8.3-AC-04	**GDC 5**, as it relates to safety-related structures not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.	Y	3.8.3.1
3.8.3-AC-05	**GDC 50**, as it relates to the design of containment internal structures with sufficient margin of safety to accommodate appropriate design loads.	Y	3.8.3.3
3.8.3-AC-06	**10 CFR 50**, Appendix B, as it relates to the quality assurance criteria for nuclear power plants.	Y	3.8.3.2.4
3.8.3-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.8.3-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency	N/A-COL	N/A

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	planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.8.3-SAC-01	Description of the Internal Structures. The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section **3.8.3.1 of RG 1.70 or 1.206**. During the application acceptance review, the reviewer identifies deficient areas of descriptive information and initiates a request for additional information. New or unique design features that are not specifically covered in **RG 1.70 or RG 1.206** may require a more detailed review. The reviewer determines whether additional information is required to accomplish a meaningful review of the structural aspects of such new or unique features. **RG 1.206** provides the basis for evaluating the description of structures to be included in a DC or a COL application. **RG 1.70** provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).	Y	3.8.3.1
3.8.3-SAC-02	Applicable Codes, Standards, and Specifications. The design, materials, fabrication, erection, inspection, testing, and inservice surveillance, if any, of containment internal structures are covered by codes, standards, and guides that are applicable either in their entirety or in part. The following codes and guides are acceptable:	Y	3.8.3.2

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	Code, Standard, or Specification Title **ACI 349** Code Requirements for Nuclear Safety-Related Concrete Structures (supplemented with additional guidance by **RG 1.142 and 1.199**) **ASME Code** **Section III, Division 2, Subsection CC**, “Code for Concrete Reactor Vessels and Containments” **ASME Code** **Section III, Division 1, Subsection NE**, “Class MC Components” **ANSI/AISC N690-1994 including Supplement 2 (2004)** Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities **Regulatory Guides** **1.57** Design Limits and Loading Combinations for Metal Primary Reactor Containment **1.69** Concrete Radiation Shields for Nuclear Power Plants **1.136** Materials, Construction, and Testing of Concrete Containments **1.142** Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) **1.143** Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants **1.160** Monitoring the Effectiveness of Maintenance at Nuclear Power Plants		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	1.199 Anchoring Components and Structural Supports in Concrete		
3.8.3-SAC-03	Loads and Load Combinations. The loads and load combinations for containment internal structures described in Subsection I.1 of this SRP are acceptable if they are consistent with the guidance given below. The loads and load combinations for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and the drywell of the BWR containment are presented following the general criteria given for concrete and steel structures.		
	A. Concrete Structures All loads and load combinations are to be in accordance with ACI 349 and **RG 1.142**. Supplemental criteria on the use of loads and load combinations are presented below. Dead loads include hydrostatic loads, and, for equipment supports, they include static and dynamic head and fluid flow effects. Live loads include any movable equipment loads and other loads that vary with intensity and occurrence. For equipment supports, they also include loads caused by vibration and any support movement effects. Alternate load cases in which the magnitudes and locations of the live loads are arranged so that worst-case conditions are included in the design should be investigated, as appropriate. As per **10 CFR 50, Appendix S**, the OBE is only associated with plant shutdown and inspection unless the applicant specifically selects it as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value	Y	3.8.3.2 3.8.3.3

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	greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the containment internal structures remain functional and are within applicable stress, strain, and deformation limits. **SRP Section 3.7.1 and 3.7.2** provides further guidance on the use of OBE. For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or SRV actuation, such loads should be considered as indicated in the **appendix to SRP Section 3.8.1**. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account. The design of concrete structures must consider the loads and load combinations that may occur during their construction. These loads consist of dead loads, live loads, temperature, wind, snow, rain, and ice. Applicable construction loads include material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. **Structural Engineering Institute (SEI)/ASCE Standard 37** provides additional guidance on construction loads. This standard may be used for supplemental guidance. When the standard and the Code/SRP provide conflicting criteria, the criteria provided in Code/SRP governs.		
	B. Steel Structures All loads and load combinations are to be in accordance with **ANSI/AISC N690-1994 including Supplement 2 (2004)**. This specification uses the allowable stress design (ASD) method. Use of the load and resistance factor design (LRFD) version of the specification (N690L) is reviewed on a case-by-case basis. The supplemental criteria on the use of loads and load	Y	3.8.3.2 3.8.3.3

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	combinations presented above for concrete structures also apply to steel structures.		
	C. Divider Barrier and Ice-Condenser of the PWR Ice-Condenser Containment Specific load and load combination criteria applicable to the divider barrier and ice-condenser elements are given below. Supplemental criteria presented in Subsection II.3.A of this SRP section are also applicable. i. Divider Barrier Because the structural integrity of the divider barrier and, to a certain extent, its leaktight integrity are important to the proper functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the loads and load combinations of **Article CC-3000 of the ASME Code, Section III, Division 2**, with additional guidance provided by applicable portions of **SRP Section 3.8.1** and **RG 1.136**. For other concrete portions of the divider barrier, the loads and load combinations as defined in Subsection II.3.A apply. Steel portions of the divider barrier that resist the design differential pressure and are not backed by concrete, such as penetrations, hatches, locks, and guard pipes, should be designed in accordance with the appropriate sections of **Subsection NE of the ASME Code, Section III, Division 1**, with additional criteria provided by applicable portions of **SRP Section 3.8.2** and **RG 1.57** apply. For other steel portions of the divider barrier, the loads and load	N/A-ICE	N/A

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	combinations as defined in Subsection II.3.B apply. ii. Ice-Condenser Elements The structural integrity of the ice baskets, ice-bed framing, and their supports is important to the functional integrity of the ice-condenser containment system. Loads and load combinations for the ice-condenser elements are acceptable if found to be in accordance with **ANSI/AISC N690-1994 including Supplement 2 (2004)**. For the ice-condenser, the load P_a is the LOCA pressure load induced by drag and change in the momentum of flowing air and steam		
	D. BWR Containment Drywell This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner. Because the structural integrity of the drywell and, to a certain extent, its leaktight integrity are critically important to the proper functioning of the pressure-suppression system, the drywell is treated, for design and testing purposes only, in a manner similar to the containment itself. Accordingly, for the concrete pressure-resisting portions of the drywell, the loads and loading combinations of **Article CC-3000 of ASME Code, Section III, Division 2**, will apply, with additional criteria provided by applicable portions of **SRP Section 3.8.1** and **RG 1.136**. For steel components of the drywell that resist pressure and are not backed by concrete, the appropriate sections of **Subsection NE of ASME Code, Section III, Division 1**, should be used with additional guidance provided by applicable portions of **SRP Section 3.8.2** and **RG 1.57**. Specifically, the loads and load	N/A-BWR	N/A

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	combinations of Subsection II.3 of SRP Section 3.8.2 apply. Additional criteria presented in Subsection II.3.A of this SRP section are also applicable to the BWR containment drywell. For the lower vent portion of the drywell, the following conditions apply: i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the criteria that apply to concrete portions of the drywell as described above will apply. ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the drywell are used for structural purposes, the criteria that apply to steel portions of the drywell as described above will apply. iii. If other structural systems are used in the vent region, the loads and load combinations are reviewed and judged on a case-by-case basis.		
3.8.3-SAC-04	Design and Analysis Procedures. The design and analysis procedures used for the containment internal structures are acceptable if found to be in accordance with the following:		
	A. PWR Dry Containment Internal Structures i. Primary Shield Wall and Reactor Cavity The design and analysis procedures used for the shield wall are acceptable if found to be in accordance with ACI 349 with additional guidance provided by **RG 1.142**. This code is based on the strength design method. The design and analysis of anchors (steel embedments) used for component	Y	3.8.3.2 3.8.3.3 3.8.3.4

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	and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by **RG 1.199**. Analyses for LOCA loads applicable to the primary shield wall, such as the cavity differential pressure combined with pipe rupture reaction forces, are acceptable if these loads are treated as dynamic time-dependent loads. This requires that either a detailed time-history analysis be performed or a static analysis using the peak of the forcing function amplified by an appropriate chosen dynamic factor be employed. Elastic behavior of the wall should be maintained under the differential pressure. However, for the concentrated accident loads, such as Yr, Yj, or Ym, elasto-plastic behavior may be assumed if the deflections are limited to maintain functional requirements. Simplified methods for determining effective dynamic load factors for elastic behavior are acceptable if found to be in accordance with recognized dynamic analysis methods. ii. Secondary Shield Walls Design and analysis procedures used for the secondary shield walls are acceptable if found to be in accordance with conventional beam/slab design and analysis procedures described in ACI 349, with additional guidance provided **RG 1.142**. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found to be in accordance with ACI 349, Appendix B, with additional guidance provided by **RG 1.199**. Similar to the primary shield wall, the secondary shield walls are also subject to dynamic LOCA loads and the methods described in Subsection		

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	II.4.A.i are, therefore, applicable and acceptable. iii. Other Interior Structures Most of the other interior structures that are reviewed are combinations of reinforced concrete slabs, walls, beams, and columns, and steel beams and columns, which are classified as Category I structures subject to the loads and load combinations described in Subsection II.3 of this SRP section. Analytical techniques for these structures are acceptable if found to be in accordance with those described in ACI 349, and with additional guidance provided by **RG 1.142 and 1.199** for concrete and anchors (steel embedments), respectively, and with **ANSI/AISC N690-1994 including Supplement 2 (2004)** for steel.		
	B. PWR Ice-Condenser Containment Internal Structures i. Divider Barrier The most important loads that usually govern the design of the divider barrier are those induced by a LOCA, including the differential pressure across the barrier and any concentrated jet impingement loads. Because the structural integrity of the divider barrier and, to a certain extent, its leaktight integrity are important to the proper functioning of the ice-condenser containment system, it is treated, for design purposes, in a manner similar to the containment itself. Accordingly, for concrete pressure-resisting portions of the divider barrier, the design and analysis procedures of Article CC-3000 of the ASME Code, Section III, Division 2, apply with additional guidance provided by applicable portions of SRP Section 3.8.1 and **RG 1.136**. For the other concrete portions of the divider	N/A-ICE	N/A

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	barrier, the design and analysis procedures are acceptable if found to be in accordance with ACI 349, with additional guidance provided by **RG 1.142 and 1.199**. These methods are based on linear elastic design methods unless the structure is subjected to concentrated accident loads, as discussed in Subsection II.4.A.i, in which elasto-plastic behavior may be assumed. For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of Subsection NE of the ASME Code, Section III, Division 1 apply, with additional guidance provided by applicable portions of SRP Section 3.8.2 and **RG 1.57**. ii. Ice-Condenser Elements The design and analysis procedures for the ice-condenser and its various components are acceptable if found to be in accordance with either the elastic/linear design method of Part 1 of **ANSI/AISC N690-1994 including Supplement 2 (2004)**, or the plastic design method of Part 2 of the same specifications. For components using experimental testing to verify the design, the testing procedures are acceptable if found to be in accordance with recognized prototype or model testing procedures that consider the effect of scaling and similitude.		
	C. BWR Containment Internal Structures This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner.	N/A-BWR	N/A

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	i. Drywell The design and analysis procedures used for concrete portions of the drywell are acceptable if found to be in accordance with Subsection II.4 of **SRP Section 3.8.1**. For steel portions of the drywell that resist pressure but are not backed by structural concrete, the design and analysis procedures are acceptable if found to be in accordance with the applicable provisions of **SRP Section 3.8.2**, Subsection II.4. ii. Weir Wall One of the major loads to which the weir wall may be subjected is a jet impingement load induced by a pipe rupture in a nearby recirculation loop. The deflection of the wall under such a load must be limited so as not to impair the pressure-suppression performance. The procedures used to analyze the wall for such a dynamic time-dependent load are acceptable if a detailed time-history dynamic analysis is performed or if an equivalent static analysis is performed using the peak of the jet load amplified by an appropriately chosen dynamic load factor. The design and analysis procedures for concrete weir walls are acceptable if found to be in accordance with conventional methods described in **ACI 349**, with additional guidance provided by **RG 1.142 and 1.199**, for concrete and anchors (steel embedments), respectively. iii. Refueling Pool and Operating Floor The refueling pool and the operating floor, which may be supported on the walls of the refueling pool on one side and on the containment shell on the other side, are constructed of a combination of reinforced concrete and		

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	structural steel. The design and analysis procedures are acceptable if found to be in accordance with conventional methods described in **ACI 349**, with additional guidance provided by **RG 1.142 and 1.199**, for concrete and anchors (steel embedments), respectively, and in **ANSI/AISC N690-1994 including Supplement 2 (2004)** for structural steel. iv. Supports for Reactor The support system for the reactor vessel, described in Subsection I of this SRP section, should be designed to resist various combinations of loadings as indicated in Subsection II.3 of this SRP section. Among the major loads that should be considered are normal operating loads, seismic loads, and LOCA loads. The design and analysis procedures used for the reactor supports (beyond the jurisdictional boundary of the ASME-designed supports) are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor. v. Reactor Pedestal The reactor pedestal, which supports the reactor and must withstand the loads transmitted through the reactor supports, should be subjected to most of the loads described in Subsection II.3 of this SRP section and should be designed for all applicable load combinations. The design and analysis procedures used for the reactor pedestal are acceptable if found to be in accordance with the same criteria for concrete applicable to the refueling pool and operating floor.		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	vi. Reactor Shield Wall This cylindrical wall, which surrounds the reactor and provides biological shielding, should be subjected to most of the loads described in Subsection II.3 of this SRP section. In many cases, the wall is used to anchor most of the pipe restraints placed around the reactor coolant system piping. A pipe rupture in the vicinity of the reactor nozzles may pressurize the space within the wall. The wall may be lined on both faces with steel plates which may constitute the major structural elements relied upon to resist the design loads. Like the reactor pedestal, the biological shield wall is also subjected to dynamic LOCA loads and the same methods are, therefore, applicable and acceptable. The design and analysis procedures used for the reactor shield wall are acceptable if found to be in accordance with the same criteria for concrete that apply to the refueling pool and operating floor. If the shield wall is constructed from steel plates filled with unreinforced concrete, then the design and analysis procedures are reviewed on a case-by-case basis. vii. Miscellaneous Platforms Platforms inside the drywell are usually constructed of structural steel and their main structural function is to provide foundations for the pipe restraints inside the drywell. Platforms outside the drywell are usually combinations of steel and concrete. The design and analysis procedures used for miscellaneous platforms are acceptable if found to be in accordance with the same criteria for concrete and steel that apply to the refueling pool and operating floor. Of particular		

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	interest are the dynamic loads induced on these floors by pool swell during a LOCA.		
	D. For all containment internal structures, the design and analysis methods described in Subsections II.4 of SRP Sections 3.8.1 and 3.8.2, which are applicable to the containment internal concrete and steel structures, respectively, also need to be considered. These items include assumptions on boundary conditions, axisymmetric and nonaxisymmetric loads, transient and localized loads, shrinkage and cracking of concrete, computer programs, and evaluation of liner plates and anchors.	Y	3.8.3.2 3.8.3.3 3.8.3.4
	E. Design of structures that use modular construction methods are reviewed on a case-by-case basis. **NUREG/CR-6486** provides guidance related to the use of modular construction methods. **Appendix B to NUREG/CR-6486** includes proposed modular construction review criteria.	N/A-OTHER (Only for Modular Construction)	3.8.3.6.5
	F. A structural design audit is conducted as described in Appendix B to SRP Section 3.8.4.	N/A-INFO	N/A
	G. The applicant's design report is considered acceptable if it satisfies the guidelines of Appendix C to SRP Section 3.8.4.	Y (for Seismic Category I Structures in Design Report)	3.8.3.4.5
		N/A-COL (for All Seismic Category I Structures)	3.8.3.4.5
3.8.3-SAC-05	Structural Acceptance Criteria.		

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	The structural acceptance criteria for containment internal structures described in Subsection I.1 of this SRP section are acceptable if found to be in accordance with the guidance given below. The acceptance criteria for the divider-barrier and ice-condenser elements of the ice-condenser PWR containment and the drywell of the BWR containment are presented following the criteria given for concrete and steel structures. The structural acceptance criteria for structures that use modular construction methods are reviewed on a case-by-case basis. See Section II.4.E of this SRP section for criteria relating to modular construction.		
	A. Concrete Structures **ACI 349** and **RG 1.142** define the structural acceptance criteria for concrete structures. The structural acceptance criteria for anchors (steel embedments) used for support of systems and components to concrete structures are acceptable if found to be in accordance with **Appendix B to ACI 349**, with additional guidance provided by **RG 1.199**.	Y	3.8.3.5
	B. Steel Structures **ANSI/AISC N690-1994 including Supplement 2 (2004)** defines the structural acceptance criteria for steel structures. This specification uses the ASD method. Use of the LRFD version of the specification (N690L) is reviewed on a case-by-case basis.	Y	3.8.3.5
	C. Divider Barrier and Ice-Condenser of PWR Ice-Condenser Containment i. Divider Barrier For concrete pressure-resisting portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with **Subsection CC-3400 of ASME Code Section III, Division 2**, with	N/A-ICE	N/A

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	additional guidance provided by applicable portions of **SRP Section 3.8.1** and **RG 1.136**. For steel portions of the divider barrier that resist pressure but are not backed by structural concrete, the design should be similar to that of steel containments. Accordingly the stress limits are acceptable if found to be in accordance with Subsection NE of the ASME Code, Section III, Division 1, with additional guidance provided by applicable portions of **SRP Section 3.8.2** and **RG 1.57**. For the other concrete and steel portions of the divider barrier, the specified limits for stresses and strains are acceptable if found to be in accordance with those provided in Subsections II.5.A and B for concrete and steel, respectively. ii. Ice-Condenser Elements For load combination delineated in Subsection II.3 of this SRP section, the specified limits for stresses and strain are acceptable if found to be in accordance with those given in **ANSI/AISC N690-1994 including Supplement 2 (2004)**.		
	D. BWR Containment Drywell This SRP section is oriented toward the BWR Mark III containment concept. Other BWR containment types are reviewed in a similar manner. For concrete and steel portions of the drywell, the specified limits for stresses and strain are acceptable if found to be in accordance with the acceptance criteria of item II.5.C.i as described for the divider barrier.	N/A-BWR	N/A

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	For the lower vent portion of the drywell, the following conditions apply: i. If the main reinforcement of the drywell is carried down between the vent holes, and the reinforced concrete section is relied upon for structural purposes, the structural acceptance criteria are the same as for item II.5.C.i above for concrete. ii. If the main reinforcement of the drywell is terminated above the vent holes, and two steel plates lining both faces of the wall are used for structural purposes, the acceptance criteria are reviewed on a case-by-case basis. iii. If other structural systems are used in the vent region, the acceptance criteria are also reviewed on a case-by-case basis		
3.8.3-SAC-06	Materials, Quality Control, and Special Construction Techniques. The specified materials of construction and quality control programs are acceptable if found to be in accordance with the public code or standard as indicated in Subsection I.6 of this SRP section. Special construction techniques, if any, are treated on a case-by-case basis. For modular construction, the materials, quality control, and special construction techniques are also reviewed on a case-by-case basis. See Section II.4.E of this SRP section for further information.	Y	3.8.3.6
3.8.3-SAC-07	Testing and Inservice Surveillance Requirements.		
	BWR containment drywells, such as those used for the Mark III containment, should be subjected to a structural proof test. Such a test is acceptable if found to be in accordance with the following:	N/A-BWR	N/A
	A. The drywell should be subjected to an acceptance test that increases the		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	drywell internal pressure in three or more approximately equal pressure increments ranging from atmospheric pressure to at least the design pressure. The drywell should be depressurized in the same number of increments. Measurements should be recorded at atmospheric pressure and at each pressure level of the pressurization and depressurization cycles. At each level, the pressure should be held constant for at least 1 hour before the deflections and strains are recorded.		
	B. So that the overall deflection pattern can be determined in prototype drywells, radial deflections should be measured at a minimum of three points along each of at least three meridians equally spaced around the drywell, including locations with varying stiffness characteristics. Radial deflections should be measured at the lower vent region, about mid-height, and near the top of the cylindrical design. Measurement points may be relocated, depending on the distribution of stresses and deformations anticipated in each particular design.		
	C. In prototype drywells only, strain measurements sufficient to permit an evaluation of strain distribution should be recorded for at least two opposing meridians at the following locations on the wall: i. At the bottom of the wall ii. At mid-height of the wall These strain measurements should be made at a minimum of three positions within the wall section - one at the center and one each near the inner and outer surfaces.		
	D. In nonprototype drywells, deflection and strain measurements need not be made if strain levels have been correlated with deflection measurements		

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	during the acceptance test of a prototype drywell when measured strains and deflections are within the predefined tolerance of their predicted responses.		
	E. Any reliable system of displacement meters, optical devices, strain gauges, or other suitable apparatus may be used for the measurements.		
	F. If the test pressure drops as a result of unexpected conditions to or below the next lower pressure level, the entire test sequence should be repeated. Significant deviations from the previous test should be recorded and evaluated.		
	G. If any significant modifications or repairs are made to the drywell following, and because of, the initial test, the test should be repeated.		
	H. A description of the proposed acceptance test and instrumentation requirements should be included in the preliminary SAR.		
	I. The following information should be submitted before the performance of the test: i. The numerical values of the predicted responses of the structure which will be measured ii. The tolerances to be permitted on the predicted responses iii. The bases on which the predicted responses and the tolerances were established		
	J. The following information should be included in the final test report: i. A description of the actual test and instrumentation ii. A comparison of the test measurements with the allowable limits (predicted response plus tolerance) for deflections and strains		

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	iii. An evaluation of the accuracy of the measurements iv. An evaluation of any deviations (i.e., test results that exceed the allowable limits), the disposition of the deviations, and the need for corrective measures v. A discussion of the calculated safety margin provided by the structure as deduced from the test results		
	For Category I structures inside containment, structures monitoring and maintenance requirements are acceptable if found to be in accordance with **10 CFR 50.65** and **RG 1.160**. It is important that Category I structures inside containment accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of the containment internal structures is essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, providing alternative means for identifying conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high radiation areas) to accommodate inservice inspection of containment internal structures are reviewed on a case-by-case basis.	ITAAC	Tier 1
SRP 3.8.4	**Other Seismic Category I Structures (R2, 03/2007)**		
3.8.4-AC-01	**10 CFR 50.55a and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1** as they relate to structures, systems, and components being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.	Y	3.8.4.2
3.8.4-AC-02	**GDC 2**, as it relates to the design of the safety-related structures being able to withstand the most severe natural phenomena such as wind, tornadoes, floods,	Y	3.8.4.2

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	and earthquakes and the appropriate combination of all loads.		3.8.4.3
3.8.4-AC-03	**GDC 4**, as it relates to safety-related structures being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.	Y	3.8.4.2 3.8.4.3
3.8.4-AC-04	**GDC 5**, as it relates to safety-related structures not being shared among nuclear power units, unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.	Y	3.8.4.1
3.8.4-AC-05	**10 CFR Part 50, Appendix B**, as it relates to the quality assurance criteria for nuclear power plants.	Y	3.8.4.6
3.8.4-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
3.8.4-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

<table>
<tr><th colspan="4">CHAPTER 3
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<tr><th>SRP Criterion</th><th>Description
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<tr><td>3.8.4-SAC-01</td><td><u>Description of the Structures</u>.
The descriptive information in the safety analysis report (SAR) is considered acceptable if it meets the criteria set forth in Section 3.8.4. and RG 1.206. New or unique design features that are not specifically covered in RG 1.70 or RG 1.206 may require a more detailed review. The reviewer determines the additional information that may be needed to accomplish a meaningful review of the structural aspects of such new or unique features.
RG 1.206 provides the basis for evaluating the description of structures to be included in a DC or a COL application.
RG 1.70 provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).</td><td>Y</td><td>3.8.4.1</td></tr>
<tr><td>3.8.4-SAC-02</td><td><u>Applicable Codes, Standards, and Specifications</u>.
The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Seismic Category I structures are covered by codes, standards, and guides that are either applicable in their entirety or in portions thereof. A list of such documents follows:
<table><tr><td><u>Codes/Specifications</u></td><td><u>Title</u></td></tr><tr><td>ACI 349</td><td>“Code Requirements for Nuclear Safety-Related Concrete Structures” (with additional criteria provided in RG 1.142)</td></tr><tr><td>ANSI/AISC N690-1994 including Supplement 2 (2004)</td><td>“Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear</td></tr></table></td><td>Y</td><td>3.8.4.2</td></tr>
</table>

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Facilities **RG** **1.69** “Concrete Radiation Shields for Nuclear Power Plants” **1.91** “Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants” **1.115** “Protection Against Low-Trajectory Turbine Missiles” **1.127** “Inspection of Water-Control Structures Associated with Nuclear Power Plants” **1.142** “Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)” **1.143** “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants” **1.160** “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” **1.199** “Anchoring Components and Structural Supports in Concrete”		
3.8.4-SAC-03	Loads and Load Combinations. The specified loads and load combinations are acceptable if found to be in accordance with the guidance given below:		
	A. Concrete Structures All loads and load combinations are to be in accordance with **ACI 349** and **RG 1.142**. Supplemental criteria on the use of loads and load combinations are presented below. Dead loads include hydrostatic loads and, for equipment supports, include	Y	3.8.4.3

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	static and dynamic head and fluid flow effects. Live loads include any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure. The dynamic effects of lateral soil pressure should be accounted for in accordance with the provisions of Subsection II.4(H) of this SRP section. For equipment supports, live loads also include loads resulting from vibration and any support movement effects. Alternate load cases, in which the magnitudes and locations of the live loads are arranged so that the design includes worst-case conditions, should be investigated, as appropriate. As noted in **Appendix S to 10 CFR Part 50**, the OBE is associated only with plant shutdown and inspection unless specifically selected by the applicant as a design input. If the OBE is set at one-third or less of the SSE ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the Seismic Category I structures remain functional and are within applicable stress, strain, and deformation limits. **SRP Section 3.7** provides further guidance on the use of OBE. For structures or structural components subjected to hydrodynamic loads resulting from LOCA and/or SRV actuation, the consideration of such loads should be as indicated in the appendix to **SRP Section 3.8.1**. Fluid structure interaction associated with these hydrodynamic loads and those from earthquakes should be taken into account. The design of concrete structures needs to consider the loads and load combinations that may occur during their construction. These loads consist of		

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	dead loads, live loads, temperature, wind, snow, rain, ice, and construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. **Structural Engineering Institute (SEI)/American Society of Civil Engineers (ASCE) Standard 37** gives additional guidance on construction loads. This standard provides supplemental guidance, and in cases where the criteria in the standard and in the Code/SRP conflict, then the Code/SRP shall govern. The analysis should consider other site-related or plant-related loads applicable to Seismic Category I structures outside the containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures. The inclusion of these loads and the related load combinations are reviewed on a case-by-case basis.		
	B. Steel Structures All loads and load combinations are to be in accordance with **AISC N690-1994 including Supplement 2 (2004)**. This specification uses the allowable stress design (ASD) method. The supplemental criteria on the use of loads and load combinations presented above for concrete structures also apply to steel structures.	Y	3.8.4.3
3.8.4-SAC-04	Design and Analysis Procedures. The design and analysis procedures used for Seismic Category I structures, including assumptions about boundary conditions and expected behavior under loads, are acceptable if found to be in accordance with the following:		

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	A. For concrete structures, the procedures are in accordance with **ACI 349**, as supplemented by **RG 1.142**. The design and analysis of anchors (steel embedments) used for component and structural supports on concrete structures are acceptable if found in accordance with **Appendix B to ACI 349**, as supplemented by **RG 1.199**.	Y	3.8.4.4
	B. The design and analysis methods described in Subsections II.4 of **SRP 3.8.1 and 3.8.2**, which apply to the other Category I concrete and steel structures, respectively, also need to be considered. Items to be considered include assumptions on boundary conditions, transient and localized loads, and shrinkage and cracking of concrete.	Y	3.8.4.2 3.8.4.3 3.8.4.4
	C. For steel structures, the procedures are in accordance with **ANSI/AISC N690-1994, including Supplement 2 (2004)**.	Y	3.8.4.4
	D. Computer programs are acceptable if the validation provided follows the procedures delineated in Subsection II.4.E of **SRP Section 3.8.1**.	Y	3.8.4.4
	E. The design report is considered acceptable if it contains the information specified in Appendix C to this SRP section.	Y (for Seismic Category I Structures in Design Report)	3.8.4.4.6
		N/A-COL (for All Seismic Category I Structures)	3.8.4.4.6
	F The structural audit is conducted in accordance with the provisions of Appendix B to this SRP section.	N/A-INFO	N/A

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	G. The design of the spent fuel pool and racks is considered acceptable when it meets the criteria of Appendix D to this SRP section.	Y	3.8.4.4.2
	H. Consideration of dynamic lateral soil pressures on embedded walls is acceptable if the lateral earth pressure loads are evaluated for two cases. These are (1) lateral earth pressure equal to the sum of the static earth pressure plus the dynamic earth pressure calculated in accordance with **ASCE 4-98, Section 3.5.3.2**, and (2) lateral earth pressure equal to the passive earth pressure. If these methods are shown to be overly conservative for the cases considered, then the staff reviews alternative methods on a case-by-case basis. For earth retaining walls, the guidance in **ASCE 4-98 Sections 3.5.3.1 through 3.5.3.3** is acceptable.	Y	3.8.4.4.2
	I. The design of masonry walls is considered acceptable when it meets the requirements of Appendix A of this SRP.	N/A	3.8.4.1.10
	J. The design of structures that use modular construction methods are reviewed and evaluated on a case-by-case basis. NUREG/CR-6486 provides guidance related to the use of modular construction methods. **Appendix B to NUREG/CR-6486** includes proposed modular construction review criteria.	N/A-OTHER (Only for Modular Construction)	3.8.4.6.3
3.8.4-SAC-05	<u>Structural Acceptance Criteria</u>. For each of the loading combinations delineated in Subsection II.3 of this SRP section, the structural acceptance criteria appear in **ACI 349** and **RG 1.142** for concrete structures, and **AISC N690-1994, including Supplement 2 (2004)**, for steel structures. The structural acceptance criteria for structures that use modular construction methods are evaluated on a case-by-case basis. See Subsection II.4.J of this	Y	3.8.4.5

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	SRP for information.		
3.8.4-SAC-06	Materials, Quality Control, and Special Construction Techniques. For Seismic Category I structures outside the containment, the materials and quality control programs are acceptable if found in accordance with the codes and standards indicated in Subsection I.6 of this SRP section. Special construction techniques, if any, are evaluated on a case-by-case basis. For modular construction, reviewers evaluate the materials, quality control, and special construction techniques on a case-by-case basis. See Subsection II.4.J of this SRP section for more information.	Y	3.8.4.6
3.8.4-SAC-07	Testing and Inservice Surveillance Requirements. For Seismic Category I structures outside containment, structures monitoring and maintenance requirements are acceptable if program is in accordance with **10 CFR 50.65** and **RG 1.160**. For water control structures, inservice inspection programs are acceptable if in accordance with **RG 1.127**. Water control structures covered by this program include concrete structures, embankment structures, spillway structures and outlet works, reservoirs, cooling water channels and canals and intake and discharge structures, and safety and performance instrumentation. For Seismic Category I structures, it is important to accommodate inservice inspection of critical areas. The staff considers that monitoring and maintaining the condition of other Category I structures is essential for plant safety. The staff reviews any special design provisions (e.g., providing sufficient physical access, providing alternative means for identification of conditions in inaccessible areas that can lead to degradation, remote visual monitoring of high-radiation areas) to	ITAAC	Tier 1

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	accommodate inservice inspection of other Category I structures on a case-by-case basis. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 ppm, sulfates <1500 ppm), an acceptable program for normally inaccessible, below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete, when excavated for any reason, for signs of degradation; and (2) conduct periodic site monitoring of ground water chemistry, to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., it exceeds any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible, below-grade concrete for signs of degradation.		
3.8.4-SAC-08	Masonry Walls. Appendix A to this SRP section contains the acceptance criteria for masonry walls.	N/A-OTHER	3.8.4.1.10
SRP 3.8.5	**Foundations (R2, 03/2007)**		
3.8.5-AC-01	**10 CFR 50.55a** and **10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1**, as they relate to safety-related structures being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.	Y	3.8.5.2
3.8.5-AC-02	**GDC 2**, as it relates to the design of the safety-related structures that are capable of withstanding the most severe natural phenomena such as wind, tornadoes, floods, and earthquakes and the appropriate combination of all loads.	Y	3.8.5.3
3.8.5-AC-03	**GDC 4**, as it relates to appropriately protecting safety-related structures against	Y	3.8.5.2

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	dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.		3.8.5.3 3.8.5.4
3.8.5-AC-04	**GDC 5**, as it relates to not sharing safety-related structures among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions.	Y	3.8.5.1
3.8.5-AC-05	**Appendix B to 10 CFR Part 50**, as it relates to the quality assurance criteria for nuclear power plants.	Y	3.8.5.6
3.8.5-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
3.8.5-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.8.5-SAC-01	Description of the Foundation. The descriptive information in the safety analysis report (SAR) is acceptable if it meets the criteria in **Section 3.8.5.1 of RG 1.206**. New or unique design features that are not specifically covered in **RG 1.206** or **RG**	Y	3.8.5.1

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	1.70 may require a more detailed review. The reviewer determines the additional information that may be needed to accomplish a meaningful review of the structural aspects of such new or unique features. **RG 1.206** provides the basis for evaluating the description of structures to be included in a DC or a COL application. **RG 1.70** provides guidance for information to be submitted with an application for construction permit (CP) or operating license (OL).		
3.8.5-SAC-02	<u>Applicable Codes, Standards, and Specifications</u>. The design, materials, fabrication, erection, inspection, testing, and surveillance, if any, of Seismic Category I foundations are covered by codes, standards, and guides that apply either in their entirety or in part. Subsection II.2 of **SRP Section 3.8.4** includes a list of such documents. In addition, the documents listed in Subsection II.2 of **SRP Section 3.8.1** are acceptable for the containment foundation.	Y	3.8.5.2
3.8.5-SAC-03	<u>Loads and Load Combinations</u>. The specified loads and load combinations used in the design of Seismic Category I foundations are acceptable if found to be in accordance with those combinations referenced in Subsection II.3 of **SRP Section 3.8.1** for the containment foundation and with those combinations listed in Subsection II.3 of **SRP Section 3.8.4** for all other Seismic Category I foundations. In addition to the load combinations referenced above, the combinations used to check against sliding and overturning attributable to earthquakes, winds, tornadoes and against flotation because of floods are acceptable if found to be in	Y	3.8.5.3

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	accordance with the following: A. D + H + E B. D + H + W C. D + H + E' D. D + H + W_t E. D + F' Where D, E, W, E', and W_t are as referenced in Subsection II.3 of **SRP Section 3.8.4**, where H is the lateral earth pressure, and F' is the buoyant force of the design-basis flood. Justification should be provided for including live loads or portions thereof in these combinations. As noted in **Appendix S to 10 CFR Part 50**, the operating-basis earthquake (OBE), designated as E above, is only associated with plant shutdown and inspection unless the applicant specifically selects it as a design input. If the OBE is set at one-third or less of the safe-shutdown earthquake (SSE) ground motion, an explicit response or design analysis is not required. If the OBE is set at a value greater than one-third of the SSE, an analysis and design must be performed to demonstrate that the Seismic Category I foundations remain functional and fall within applicable stress, strain, and deformation limits. **SRP Section 3.7** provides additional guidance on OBE use.		
3.8.5-SAC-04	Design and Analysis Procedures.		

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	The design and analysis procedures used for Seismic Category I foundations are acceptable if found to be in accordance with the following:		
	A. The design should consider the soil-structure interaction, hydrodynamic effect, and dynamic soil pressure.	Y	3.8.5.4
	B. For Seismic Category I concrete foundations other than the containment foundations, the procedures are in accordance with the **ACI 349**, with additional guidance provided by **RG 1.142**.	Y	3.8.5.4
	C. For Category I steel foundations, the procedures are in accordance with **ANSI/AISC N690-1994 including Supplement 2 (2004)**.	Y	3.8.5.4
	D. For the containment foundation, if in accordance with the design and analysis procedures referenced in **SRP Section 3.8.1**, Subsection II.4.	Y	3.8.5.4
	E. The design report is acceptable if it satisfies the guidelines provided in **SRP Section 3.8.4, Appendix C**.	Y (for Seismic Category I Structures in Design Report)	3.8.5.4.5
		N/A-COL (for All Seismic Category I Structures)	3.8.5.4.5
	F. The structural audit is conducted in accordance with **SRP Section 3.8.4, Appendix B**.	N/A-INFO	N/A
	Methods for determining the overturning moment attributable to an earthquake should be in accordance with the methods described in **SRP Section 3.7.2**.	Y	3.8.5.4

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	Computer programs are acceptable if the validation provided is found to be in accordance with the procedures delineated in Subsection II.4.E of **SRP Section 3.8.1**.		
	In addition to the above, the design and analysis procedures for the following details are reviewed on a case-by-case basis: A. Method for determination of the bending moments and shear forces in the foundation mat for seismic loads? B. Performance of the sliding analysis method and how the analysis adequately accounts for potential foundation mat liftoff effects, if appropriate? The method to calculate the factor of safety against sliding. If sliding resistance is the sum of shear friction along the base mat and passive pressures induced by embedment effects, how these effects are considered in an analysis based on a consistent lateral displacement criterion? C. Evaluation of the capability of a foundation to transfer shear when waterproofing is used for a range of site conditions (soil sites with shear wave velocity of 1000 feet per second to hard rock)? D. The definition of dead load for uplift evaluations (floatation and seismic overturning), including the treatment of the stored volume of water in any pools? E. Detail explanation of how settlement (including potential effects of static or dynamic differential settlement) was considered. Evaluation and consideration of the effects of settlement on construction procedures. Evaluation of the allowable settlement (total and differential) that can be accommodated in the foundation/structures?	N/A-INFO	N/A

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	F. The maximum toe pressure for base mat design under worst-case static and dynamic loads and its justification. G. The stiff and soft spots evaluation in the foundation soil to maximize the bending moments used in the design of the foundation mat. H. Description of the design details of critical locations, such as the junction of sidewall and base mat and the junctions of base mat to sumps. I. Detail explanation of the load path from all superstructures to the foundation mat to the subgrade. Discussion of any unique design features that occur in the load path (e.g., any safety-related function that the tendon gallery may have as part of the foundation in a prestressed containment or the connection of any internal structures to a steel containment and its supporting foundation). J. Explanation of how loads attributable to construction are considered in the design. Some examples of items to be discussed include the excavation sequence and loads from the construction sequence of the foundation mat and walls, as well as the potential for loss of subgrade contact (e.g., because of loss of cement from a mud mat) that may lead to a differential pressure distribution on the mat.		
3.8.5-SAC-05	<u>Structural Acceptance Criteria</u>. For the loading combinations referenced in the first paragraph of Subsection II.3 of this SRP section, the allowable limits that constitute the acceptance criteria are referenced in Subsection II.5 of **SRP Section 3.8.1** for the containment foundation and in Subsection II.5 of **SRP Section 3.8.4** for all other foundations. In addition, for the five other load combinations in Subsection II.3 of this SRP section, the factors of safety against overturning, sliding, and flotation are	Y	3.8.5.5

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	acceptable if found to be in accordance with the following: Minimum Factors of Safety For Combination Overturning Sliding Flotation a.----------------------1.5 1.5 --- b.----------------------1.5 1.5 --- c.----------------------1.1 1.1 --- d.----------------------1.1 1.1 --- e.---------------------- --- --- 1.1		
3.8.5-SAC-06	Materials, Quality Control, and Special Construction Techniques. For the containment foundation, Subsection II.6 of **SRP Section 3.8.1** references the acceptance criteria for materials, quality control, and any special construction techniques. For all other Seismic Category I foundations, the materials and quality control programs are acceptable if found to be in accordance with the codes and standards indicated in Subsection I.6 of this SRP section. Special construction techniques, if any, are treated on a case-by-case basis.	Y	3.8.5.6
3.8.5-SAC-07	Testing and Inservice Surveillance Requirements. For Category I foundations, structure monitoring and maintenance requirements are acceptable if found to be in accordance with **10 CFR 50.65** and **RG 1.160**.	ITAAC	Tier 2

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	For water control structures, inservice inspection programs are acceptable if found to be in accordance with **RG 1.127**. Water control structures covered by this program include concrete structures, embankment structures, spillway structures and outlet works, reservoirs, cooling water channels and canals, as well as intake and discharge structures, and safety and performance instrumentation. For Category I foundations, it is important to accommodate inservice inspection of critical areas. The staff considers monitoring and maintaining the condition of Category I foundations as essential for plant safety. Any special design provisions (e.g., providing sufficient physical access, supplying a means for identification of conditions in inaccessible areas that can lead to degradation, performing remote visual monitoring of high-radiation areas) to accommodate inservice inspection of Category I foundations are reviewed on a case-by-case basis. For plants with nonaggressive ground water/soil (i.e., pH > 5.5, chlorides < 500 parts per million (ppm), sulfates < 1500 ppm), an acceptable program for normally inaccessible below-grade concrete walls and foundations is to (1) examine the exposed portions of below-grade concrete for signs of degradation, when excavated for any reason, and (2) conduct periodic site monitoring of ground-water chemistry to confirm that the ground water remains nonaggressive. For plants with aggressive ground water/soil (i.e., exceeding any of the limits noted above), an acceptable approach is to implement a periodic surveillance program to monitor the condition of normally inaccessible below-grade concrete for signs of degradation. Subsection II.7 of **SRP Section 3.8.1** covers additional testing and surveillance requirements for the containment foundation. Design of any special foundations		

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	will be reviewed on a case-by-case basis.		
SRP 3.9.1	**Special Topics for Mechanical Components (R3, 03/2007)**		
3.9.1-AC-01	**10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, which requires, in part, that components important to safety be designed, fabricated, erected, and, tested to quality standards commensurate with the importance of the safety functions to be performed.	Y	3.9.1
3.9.1-AC-02	**GDC 2**, which requires, in part, that components important to safety be designed to withstand seismic events without loss of capability to perform their safety functions.	Y	3.9.2
3.9.1-AC-03	**GDC 14**, which requires that the reactor coolant pressure boundary be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y	3.9.1
		N/A-COL (For fracture analysis of reactor coolant pressure boundary)	3.9.2
3.9.1-AC-04	**GDC 15**, which requires that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Y	3.9.2
3.9.1-AC-05	**10 CFR Part 50, Appendix B, Section III**, as it relates to quality of design control.	Y	3.9.1
3.9.1-AC-06	**10 CFR Part 50, Appendix S**, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic	N/A-COL (Confirm site specific	3.9.1

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	characteristics.	seismic is bounded by seismic design basis)	
3.9.1-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.1-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.9.1-SAC-01	To meet the requirements of **GDCs 1, 2, 14, 15**, and **10 CFR Part 50, Appendix S**, the applicant should provide a complete list of transients to be used in the design and fatigue analysis of all Code Class 1 and core support components, supports, and reactor internals within the reactor coolant pressure boundary. The number of events for each transient and the number of load and stress cycles per event and for events in combination should be included. All transients, such as startup and shutdown operations, power level changes, emergency and recovery conditions (including, for new applications, natural convection cooldown), switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients from single	Y	3.9.1

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	operator errors, inservice hydrostatic tests, seismic events as determined from the criteria specified in Appendix S to 10 CFR Part 50, and design-basis events contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary, should be specified. The section of the applicant's SAR on transients will be acceptable if the transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure conditions caused by those transients. To a large extent the selection of these specific transient conditions is based upon engineering judgment and experience. Some guidance on the selection of these transients and combinations can be found in SRP Section 3.9.3. Transients and consequent loads and load combinations with appropriate specified design and service limits should provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant. The staff should consider the number of transients appropriate for the design life of the plant. Also, environmental conditions to which equipment important to safety will be exposed (e.g., chemistry of the coolant water) should be considered to minimize the degradation of materials due to corrosion.		
3.9.1-SAC-02	To meet the requirements of **10 CFR Part 50, Appendix B**, and **GDC 1**, a list of computer programs to be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses should be provided. For each program the following information should be provided to demonstrate applicability and validity:	Y (Per AREVA Topical Report ANP-10264)	3.9.1
	A. The author, source, dated version, and facility.		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	B. A description and the extent and limitation of its application.		
	C. The computer program solutions to a series of test problems demonstrated to be substantially similar to solutions obtained from any one of sources (i) through (iv) and source (v): (i) Hand calculations (ii) Analytical results published in relevant engineering literature (iii) Acceptable experimental tests (iv) Results from a similar program within the acceptable margins. (v) The benchmark problems prescribed in **NUREG/CR-1677**, “Piping Benchmark Problems.” Vols. I and II.		
	A summary comparison of the solution obtained from sources (i) through (iv) should be provided in either graphical or numerical form. For source (v), the complete computer printout of the input and the solution should be submitted for every benchmark problem. These solutions may be referenced, and need not be resubmitted, in subsequent license applications, provided the information submitted under Items A and B remains unchanged.		
3.9.1-SAC-03	To meet the requirements of **GDCs 1, 14, and 15**, if experimental stress analysis methods are used in lieu of analytical methods for any seismic Category I Code or non-Code items, the section of the SAR addressing the experimental stress analysis methods is acceptable if the information meets the provisions of **Appendix II to ASME Code, Section III, Division 1** and, as in the case of analytical methods, if the information is sufficiently detailed to show the design meeting the provisions of the Code-required "Design Specifications."	NA-OTHER (No experimental methods used)	3.9.1

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3.9.1-SAC-04	To meet the requirements of **GDCs 1, 14, and 15** when **Service Level D limits** are specified by the applicant **for Code Class 1** and core support components and for supports, reactor internals, and other non-Code items, the methods of analysis to calculate the stresses and deformations should conform to the methods outlined in **Appendix F to ASME Code, Section III, Division 1**, subject to the conditions addressed in **subsection III.4 of this SRP section**.	Y	3.9.1
		N/A-COL (To complete site-specific faulted condition stress analyses)	3.9.2
SRP 3.9.2	**Dynamic Testing and Analysis of Systems, Structures, and Components (R3, 03/2007)**		
3.9.2-AC-01	**10 CFR 50.55a** and **General Design Criterion (GDC) 1 to 10 CFR Part 50, Appendix A**, as they relate to the testing of systems and components to quality standards commensurate with the importance of the safety function to be performed.	Y	3.9.2
		N/A-COL (For reactor pressure vessel internals testing)	3.9.1
		N/A-COL (For start-up testing)	14.2.2
3.9.2-AC-02	**GDC 2** and **10 CFR Part 50, Appendix S**, as they relate to systems, structures, and components important to safety designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena.	Y	3.9.2
		N/A-COL (For secondary stress and fatigue analysis)	3.9.2
		N/A-COL (For reporting test results)	3.9.12
3.9.2-AC-03	**GDC 4**, as it relates to systems, structures, and components important to safety	N/A-COL	3.6.1

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	appropriately protected against the dynamic effects of discharging fluids.	(For pipe break hazards analysis)	3.6.2
3.9.2-AC-04	**GDC 14**, as it relates to designing systems, structures, and components of the reactor coolant pressure boundary to have an extremely low probability of rapidly propagating failure and of gross rupture.	Y	3.9.2
		N/A-COL (For fracture analysis of reactor coolant pressure boundary)	3.9.2
3.9.2-AC-05	**GDC 15**, as it relates to designing the reactor coolant system with sufficient margin to assure that the reactor coolant pressure boundary is not exceeded during normal operating conditions, including anticipated operational occurrences.	Y	3.9.2
		N/A-COL (For secondary stress and fatigue analysis)	3.9.2
3.9.2-AC-06	**Appendix B to 10 CFR Part 50**, as it relates to quality assurance in the dynamic testing and analysis of systems, structures, and components.	Y	3.9.2
3.9.2-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.2-AC-08	10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections,	N/A-COL	N/A

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	tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.9.2-SAC-01	Relevant requirements of **GDCs 1, 2, 4, 14, and 15** are met if vibration, thermal expansion, and dynamic effects testing are conducted during startup functional testing for specified high- and moderate-energy piping and their supports and restraints. The purposes of these tests are to confirm that the piping, components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions encountered during service as required by the code and to confirm that no unacceptable restraint of normal thermal motion occurs. An acceptable test program to confirm the adequacy of the designs should include the following:	Y N/A-COL (For start-up testing)	3.9.2 14.2.2
	A. A list of systems to be monitored.	Y	3.9.2
	B. A list of the flow modes of operation and transients like pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see **RG 1.68**). For example, the transients of the reactor coolant system heatup tests should include but not necessarily be limited to: (i) Reactor coolant pump start. (ii) Reactor coolant pump trip. (iii) Operation of pressure-relieving valves. (iv) Closure of a turbine stop valve.	Y	3.9.2
	C. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak), pressure, or other	Y	3.9.2

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	appropriate criteria to show that the stress and fatigue limits are within the design levels should be provided.		
	D. A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position.	N/A-COL	3.9.13
	E. A description of the thermal motion monitoring program (i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and how motion will be measured).	Y	3.9.2
	F. If vibration is noted beyond the acceptance levels set by the criteria of Item II.1.C above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should determine whether the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in Item II.1.D of the acceptance criteria, the corrective action to be taken to ensure that the snubber is operable should be described.	Y	3.9.2
3.9.2-SAC-02	To meet the requirements of **GDC 2**, acceptance criteria for the areas of review described in subsection I.2 of this SRP section are given below. Other approaches which can be justified as equivalent to or more conservative than the stated acceptance criteria may be used to confirm the ability of all Seismic Category I systems and components and their supports to function as needed during and after an earthquake.	Y (Per AREVA Topical Report ANP-10264)	3.9.2
	A. <u>Seismic Analysis Methods</u>. The seismic analysis of all Category I systems, components, equipment, and	Y (Per AREVA Topical	3.7.3 3.9.2

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	their supports (including supports for conduit and cable trays and ventilation ducts) should utilize either a suitable dynamic analysis method or an equivalent static load method, if justified.	Report ANP 10264)	
	i. <u>Dynamic Analysis Method</u>. A dynamic analysis (e.g., response spectrum method, time history method, etc.) should be used when the use of the equivalent static load method cannot be justified. To be acceptable such analyses should consider the following items: (1) Use of either the time history or the response spectrum method. (2) Use of an adequate number of masses or degrees of freedom in dynamic modeling to determine the response of all Category I and applicable non-Category I systems and plant equipment. The number is adequate when additional degrees of freedom do not result in more than a 10-percent increase in responses. Alternately, the number of degrees of freedom may be taken as equal to twice the number of modes with frequencies less than 33 Hz. (3) Investigation of a sufficient number of modes to ensure participation of all significant modes. The criterion for sufficiency is that the inclusion of additional modes does not result in more than a 10-percent increase in responses. (4) Consideration of maximum relative displacements among supports of Category I systems and components. (5) Inclusion of such significant effects as piping interactions, externally-applied structural restraints, hydrodynamic (both mass and stiffness	Y (Per AREVA Topical Report ANP 10264)	3.9.2

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	effects) loads, and nonlinear responses.		
	ii. Equivalent Static Load Method. An equivalent static load method is acceptable if: (1) There is justification that the system can be realistically represented by a simple model and the method produces conservative results in responses. Typical examples or published results for similar systems may be submitted in support of the use of the simplified method. (2) The design and simplified analysis account for the relative motion between all points of support. (3) To obtain an equivalent static load of equipment or components which can be represented by a simple model, a factor of 1.5 is applied to the peak acceleration of the applicable floor response spectrum. A factor of less than 1.5 may be used with adequate justification. In addition, for equipment which can be modeled adequately as a one-degree-of-freedom system, the use of a static load equivalent to the peak of the floor response spectra is acceptable. For piping supported at only two points, the use of a static load equivalent to the peak of the floor response spectra is also acceptable.	Y (Per AREVA Topical Report ANP 10264)	3.9.2
	B. Determination of Number of Earthquake Cycles. The number of earthquake cycles during one seismic event, the maximum number of cycles for which applicable systems and components are designed, and the criteria and the applicant's procedures to establish these parameters are reviewed by the staff in accordance with the guidance of **SRP Section 3.7.3**.	Y (Per AREVA Topical Report ANP 10264)	3.7.3

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	C. Basis for Selection of Frequencies. To avoid resonance, the fundamental frequencies of components and equipment selected preferably should be less than ½ or more than twice the dominant frequencies of the support structure. Use of equipment frequencies within this range is acceptable if the equipment is adequately designed for the applicable loads.	Y	3.7.3
	D. Three Components of Earthquake Motion. Depending upon what basic methods are used in the seismic analysis (i.e., response spectra or time history method) the following two approaches are acceptable for the combination of three-dimensional earthquake effects. (i) Response Spectra Method. When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum codirectional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model. (ii) Time History Analysis Method. When the time history analysis method is employed for seismic analysis, two types of analysis are generally performed depending on the complexity of the problem. (1) to obtain maximum responses to each of the three components of the earthquake motion: in this case the method for combining the three-dimensional effects is identical to that described in Item (i) except that the maximum responses are calculated by the time	Y (Per AREVA Topical Report ANP 10264)	3.9.2

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	history method instead of the spectrum method. (2) To obtain time history responses from each of the three components of the earthquake motion and combine them at each time step algebraically: the maximum response in this case can be obtained from the combined time solution. When this method is used, to be acceptable the earthquake motions specified in the three different directions should be statistically independent.		
	E. Combination of Modal Responses. **SRP Section 3.7.2** and **RG 1.92**,"Combining Modal Responses and Spatial Components in Seismic Response Analysis," present criteria and guidance for modal response combination methods acceptable to the staff.	Y (Per AREVA Topical Report ANP 10264)	3.9.2
	F. Analytical Procedures for Piping Systems. The seismic analysis of Category I piping may use either a dynamic analysis or an equivalent static load method. The acceptance criteria for the dynamic analysis or equivalent static load methods are described in subsection II.2.A of this SRP section.	Y (Per AREVA Topical Report ANP 10264)	3.9.2
	G. Multiply-Supported Equipment and Components With Distinct Inputs. Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the primary structure or structures at each of the support points may be quite different. A conservative and acceptable approach for equipment items supported at two or more locations is to use an upper-bound envelope of all the individual response spectra for these locations to calculate maximum inertial responses of multiply-supported items. In addition, the relative displacements at the support points should be considered. Conventional static analysis procedures	Y	3.7.3

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	are acceptable for this purpose. The maximum relative support displacements can be obtained from the structural response calculations or, as a conservative approximation, from the floor response spectra. For the latter option, the maximum displacement of each support (Sd) is predicted by: $S_d = S_a g / \omega^2$ where Sa is the spectral acceleration in "g's" at the high frequency end of the spectrum curve (which, in turn, is equal to the maximum floor acceleration), g is the gravity constant, and ω is the fundamental frequency of the primary support structure in radians per second. The support displacements can then be imposed on the supported item in the most unfavorable combination. The responses due to the inertia effect and relative displacements should be combined by the absolute sum method. In the case of multiple supports located in a single structure, an alternate acceptable method using the floor response spectra determines dynamic responses due to the worst single floor response spectrum selected from a set of floor response spectra at various floors and applied identically to all the floors provided there is no significant shift in frequencies of the spectra peaks. In addition, the support displacements should be imposed on the supported item in the most unfavorable combination by static analysis procedures. Further criteria and methods for the evaluation of multiple support arrangement analysis issues are described in **SRP Sections 3.7.2 and 3.7.3**. These methods can result in overestimation of seismic responses. Acceptable alternate response spectrum analysis methods that provide more realistic estimation of seismic responses are discussed in subsection II.9 of **SRP**		

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	Section 3.7.3. In lieu of the response spectrum approach, time histories of support motions may be used as excitations to the systems. Because of the increased analytical effort compared to the response spectrum techniques, usually only a major equipment system would warrant a time history approach. The time history approach does, however, provide more realistic results in some cases as compared to the response spectrum envelope method for multiply-supported systems.		
	H. Use of Constant Vertical Static Factors. The use of constant vertical load factors as vertical response loads for the seismic design of all Category I systems, components, equipment, and their supports in lieu of a vertical seismic system dynamic analysis is acceptable only if the structure is demonstrably rigid in the vertical direction. The criterion for rigidity is that the lowest frequency in the vertical direction be more than 33 Hz.	Y	3.7.3
	I. Torsional Effects of Eccentric Masses. For Seismic Category I systems, if the torsional effect of an eccentric mass like a valve operator in a piping system is judged to be significant, the eccentric mass and its eccentricity should be included in the mathematical model. The criteria for significance will have to be determined case by case.	Y (Per AREVA Topical Report ANP 10264)	3.9.2
	J. Category I Buried Piping Systems. For Category I buried piping systems, the following items should be considered in the analysis: (i) The inertial effects due to an earthquake upon buried piping systems	Y (Per AREVA Topical Report ANP 10264)	3.9.2

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	should be adequately considered in the analysis. Use of the procedures described in the references is acceptable. (ii) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., should be adequately considered. Use of the procedures described in the references is acceptable. (iii) When applicable, the effects of local soil settlements, soil arching, etc., also should be considered in the analysis.		
	K. Interaction of Other Piping with Category I Piping. To be acceptable, each non-Category I piping system should be designed to be isolated from any Category I piping system by either a constraint or barrier or should be located remotely from the seismic Category I piping system. If isolation of the Category I piping system is not feasible or practical, adjacent non-Category I piping should be analyzed according to the same seismic criteria applicable to the Category I piping system. For non-Category I piping systems attached to Category I piping systems, the dynamic effects of the non-Category I piping should be simulated in the modeling of the Category I piping. The attached non-Category I piping, up to the first anchor beyond the interface, also should be designed not to cause a failure of the Category I piping during an earthquake of SSE intensity.	Y (Per AREVA Topical Report ANP 10264)	3.9.2
	L. Criteria Used for Damping. **RG 1.61**,"Damping Values for Seismic Design of Nuclear Power Plants," provides acceptable values which may be used. The methods for analysis of	Y (Per AREVA Topical Report ANP 10264)	3.9.2

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	damping should be consistent with those described in **SRP Section 3.7.2**.		
3.9.2-SAC-03	To meet the requirements of **GDCs 1 and 4**, the following guidelines, in addition to **RG 1.20** "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing", apply to the analytical solutions to predict vibrations of reactor internals for prototype plants. Generally, this analysis is required only for prototype designs and power uprate of existing plants; However, it is not required for non-prototypes except that segments of an analysis (in particular, assessments of any potential adverse flow effects) may be necessary if there are deviations from the prototype internals design or operating conditions or if the non-prototype is based on a conditional prototype which has experienced problems from adverse flow effects. If the reactor internal structures are a non-prototype design, the applicant should refer to the results of tests and analyses for the prototype reactor and give a brief summary of the results. A more detailed summary of results of assessment of the potential of any adverse flow effects also should be given.	Y	3.9.2
		N/A-COL (For comprehensive vibration assessment program)	3.9.1
	A. The results of vibration and stress calculations should consist of the following: (i) Dynamic responses to operating transients at critical locations of the internal structures should be determined and, in particular, at the locations where vibration sensors will be mounted on the reactor internals. For each location, the maximum response, the modal contribution to the total response, (in case of cyclic or resonant behavior), and the response causing the maximum stress amplitude should be calculated. (ii) The damping factors for different modes should be properly selected and substantiated. In prior submissions, utilities have cited NRC damping guidance for very low frequency seismic analyses as justification for high	Y	3.9.2
		N/A-COL (For comprehensive vibration assessment program)	3.9.1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	damping factors for mid-to-high frequency analyses. **RG 1.20** corrects this guidance and requires that damping factors used in structural dynamic modeling be based on mid- to high-frequency measurements or rigorous analyses conducted on structures typical of the reactor internal structure modeled. (iii) The dynamic properties of internal structures, including the natural frequencies and shapes of the dominant modes, should be characterized. In analyses of a component structural element basis, the presence of dynamic coupling among component structure elements should be investigated. Upper bounds on the uncertainties of all natural frequencies of the relevant resonance modes should be provided. The uncertainties and bias errors of the amplitudes of the frequency response functions (FRFs) also should be provided. The uncertainties and bias errors may be estimated from comparisons of simulations to measurements made on structures similar in construction to the reactor internal being modeled. The performance of hammer tests would be expected for replacement steam dryers. (iv) Dynamic responses of reactor internals to self-excited flow oscillations should be estimated. The applicants/licensees should analyze in detail adverse flow effects generated by various excitation mechanisms like vortex-induced vibration flow-excited acoustic resonance, fluid-elastic instability, and other flow instabilities (e.g., separated and impinging flow instabilities). These mechanisms may be assessed by theoretical, numerical, or experimental techniques, including scale model testing. The analysis should clearly identify whether each mechanism will be excited		

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	during the planned operating range of the power plant. Full dynamic analysis is requested for mechanisms expected to generate adverse flow effects, including estimation of vibration and stress amplitudes at the critical locations and, in particular, where vibration sensors will be mounted on the reactor internals. **RG 1.20, Section C.2.1.3** provides more guidance on self-excited flow instabilities. (v) The dependance of the dynamic response on hydrodynamic excitation forces like coolant recirculation pump frequencies and the flow path configuration should be evaluated. Any frequency coincidence between the pump blade passing frequency and the natural frequencies of the internal structures should be identified and supplemented with error and uncertainty analysis. (vi) Acceptance criteria should be established for allowable responses and for the location of vibration sensors. Such criteria relate to the code-allowable stresses, strains, and limits of deflection established to preclude loss of function of the reactor core structures and fuel assemblies.		
	B. The forcing functions should account for the effects of transient flow conditions and the frequency content. Any potential amplification of a forcing function caused by self-excitation or "lock-in" of a flow instability with a structural or acoustic resonance should be clearly quantified (See **RG 1.20, Section C.2.1.3** for more guidance on self-excited flow instabilities). Acceptable methods for formulating forcing functions for vibration prediction include the following:	Y	3.9.2
		N/A-COL (For comprehensive vibration assessment program)	3.9.1
	i. Analytical method: based on standard hydrodynamic theory, the governing differential equations for vibratory motions should be developed and	N/A-OPT	N/A

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	solutions obtained with appropriate boundary conditions and parameters. This method is acceptable where the geometry along the fluid flow paths is mathematically tractable.	(See SRP 3.9.2, 3.9.2-SAC-03.B.ii)	
	ii. Test-analysis combination method: based on data obtained from plant or scale model tests (e.g., velocity or pressure distribution data), forcing functions should be formulated to include the effects of complex flow path configurations and wide variations of pressure distributions. The suitability of any approach used to define forcing functions should be assessed with expected bias errors and uncertainties of the selected approach. In addition to direct measurements in nuclear power plants, the following approaches may be used to formulate the forcing functions. (1) Scale Model Tests (SMTs): If SMTs are used to develop forcing functions, the following areas should be considered. (a) The scale model should be dynamically similar to the prototype. The dynamic similarity should cover all fluid, structural (such as piping dimensions and elbow locations), and acoustic parameters relevant to the phenomenon considered. If some distortions in the dimension-less parameters of the scale model should be made, the applicants/licensees should show that these distortions are conservative. As an example, sound attenuation in scale models is normally substantially higher than that of the prototype due to viscous heat conduction and other losses higher in small-size models tested at low pressures, leading to the requirement that the scale model size and its test pressure be sufficiently large to ensure the re-production of such specific flow phenomena as flow-induced	Y	3.9.2
		N/A-COL (For comprehensive vibration assessment program)	3.9.1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	vibration and acoustic resonance present in the prototype. (b) The effects of structural damping and sound attenuation (in the test medium) on the loading function measured in the scale model should be considered carefully. Any non-conservative deviations in these parameters from those of the prototype reactor should be corrected when the loading function is scaled to that of a full-size reactor pressure vessel (RPV). (c) The conservative simulation of boundary conditions in the scale model. (d) Whether the size of the scale model is sufficiently large to allow investigation of small relevant details in geometry (e.g., branch line openings). (e) Validation of the SMT results by measurements in nuclear power plants. (2) CFD: If CFD simulations are used to develop unsteady forcing functions, the following areas should be considered. (a) Include acoustic/vibration coupling to simulate enhancement of flow instabilities (if any). (b) Grid size sensitivity tests. (c) The Courant number requirement should be met. (d) There should be unsteady simulations using Large Eddy Simulation (LES) or Direct Numerical Simulation (DNS) at high Reynolds number flow and including compressibility effects to model any coupling of the flow with the acoustic waves in the fluid (self-		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	excitation or lock-in effects). (e) Real gas simulation should be used (i.e., use state equation of steam as real gas). (f) The simulation procedures should be validated on similar (i.e., complex and high Reynolds number) flow situations. (3) Acoustic Modeling of Steam System: If an acoustic model of the steam system (the steam within the MSLs and the RPV) computes fluctuating pressures within the RPV and on BWR steam dryers inferred from measurements of fluctuating pressures within the MSLs connected to the RPV, the following areas should be considered. (a) There should be at least two measurement locations on each MSL in a BWR; however, three measurement locations on the MSLs improve input data to an acoustic model, particularly if the locations are spaced logarithmically, reducing uncertainty in describing the waves coming from and going into the RPV. With two or three measurement locations, there should be no acoustic sources between the measurement locations, unless justified. (b) Strain gages (at least four gages circumferentially oriented and placed at equal distance along the circumference) may be used to relate the hoop strain in the MSL to the internal pressure. Strain gages should be calibrated according to the MSL dimensions (diameter, thickness, and static pressure). Alternatively, pressure measurements made with transducers flush-mounted against the MSL internal surface may be used. The effects of flow turbulence on any direct pressure measurements should be considered,		

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	however. (c) The speed of sound in any acoustic models should not be changed from plant to plant but rather be a function of temperature and steam quality. (d) Reflection coefficients at any boundary between steam and water should be based on rigorous modeling or on direct measurement. The uncertainty of the reflection coefficients should be clearly defined. (e) Any sound attenuation coefficients should be a function of steam quality (variable between the chimney and reactor dome) rather than constant throughout a steam volume (like the volume within the RPV). (f) Once validated, the same speed of sound, attenuation coefficient, and reflection coefficient should be used in other plants; however, different flow conditions (temperature, pressure, quality factor) may require adjustments of these parameters. (4) Response-deduction method: based on a derivation of response characteristics from plant or SMT data, forcing functions should be formulated; however, as such functions may not be unique and are also expected to depend on material properties and loss factors, the computational procedures and the basis for selection of the representative forcing functions should be described together with all bias errors and uncertainties (see subsection II.3.B.(ii)(1) of this SRP section, “Scale Model Tests,” for guidelines on inferring forcing functions from plant or scale model testing data).		

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	Alternately, the applicant/licensee may use other approaches to formulate the forcing function. However, sufficient supporting justification should be provided to demonstrate that the selected approach is technically sound and realistically predicts the forcing function. In addition, an assessment of bias errors and uncertainties should be provided.		
	C. Acceptable methods of obtaining dynamic responses for vibration and stress predictions are as follows: (i) If a numerical model is used to compute mode shapes and FRFs, the modeling approach should be documented along with the model itself. Uncertainties and bias errors for both the approach and the specific model should be provided along with their bases. Additional guidance on numerical uncertainties and bias errors can be found in **RG 1.20**. (ii) Force-response computations are acceptable if the characteristics of the forcing functions are predetermined conservatively and the mathematical model of the reactor internals is appropriately typical of the design. (iii) If the forcing functions are not predetermined, either a special analysis of response signals measured from reactor internals of similar design may predict amplitude and modal contributions or parameter studies useful for extrapolating the results from tests of internals or components of similar designs based on composite statistics may be used. The latter approach should be used only when the expectation that flow-induced vibration or acoustic resonance will not occur for the operation conditions covering the extrapolated range of the forcing functions is shown beyond doubt.	Y	3.9.2
		N/A-COL (For comprehensive vibration assessment program)	3.9.1

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	D. Vibration predictions should be verified by RPV, steam, feed water and condensate piping, and safety relief valve test results. This procedure should consider all sources of bias errors and uncertainties. If the test results differ substantially from the predicted response behavior, the vibration analysis should be modified appropriately for more agreement with test results and validation of the analytical method and input forcing functions as appropriate for predicting responses of the prototype unit as well as of other units where confirmatory tests are conducted.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
3.9.2-SAC-04	For requirements of **GDCs 1 and 4**, the preoperational vibration and stress test program for the internals of a prototype reactor, for existing reactors under consideration for power uprate, and for non-prototype reactors whose valid or conditional prototypes have experienced structural failures due to adverse flow effects in any plant (e.g., steam dryer cracking and valve failures) should conform to the requirements for a prototype test as specified in **RG 1.20**, including vibration prediction, vibration monitoring, adverse flow effects (flow-induced acoustic and structural resonances, data reduction, bias errors and uncertainty analysis, and walkdown and surface inspections. The test program to demonstrate design adequacy of the reactor internals should include, but not necessarily be limited to, the following:	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	A. The vibration testing should be conducted with the fuel elements in the core or with dummy elements with equivalent dynamic effects and flow characteristics. Testing without fuel elements in the core may be acceptable if testing in this mode is demonstrably conservative.	Y N/A-COL (For comprehensive vibration assessment program)	3.9.2 3.9.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	B. The vibration monitoring instrumentation should be described briefly, including instrument types and specifications (including useful frequency and amplitude ranges) and diagrams of locations, including those with the most severe vibratory motions or the most effect on safety functions.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	C. Testing to evaluate potential adverse flow effects on reactor internal components should include the steam dryer and MSL valves. The instrumentation directly mounted on the steam dryer should include pressure sensors, strain gages, and accelerometers. The MSLs also should be instrumented to collect data to determine steam pressure fluctuations to identify the presence of flow-excited acoustic resonances and to allow the analysis of those pressure fluctuations to calculate MSL valve loading and vibration and steam dryer loading and stress. Accelerometers should be mounted on the main steam valves to record the presence and the level of any flow-excited acoustic resonance or vibration.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	D. The planned duration of the test for the normal operation modes to ensure that all critical components are subjected to at least 106 cycles of vibration should be provided. For instance, if the lowest response frequency of the core internal structures is 10 Hz, a total test duration of 1.2 days or more is acceptable.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	E. Testing should include all of the flow modes of normal operation and upset transients. The proposed set of flow modes is acceptable if it provides a conservative basis for determining the dynamic response of the tested components and is reviewed on request. The power ascension program for startup testing should include specific hold points with sufficiently long duration to allow data recording and reduction, comparisons with predetermined limit	N/A-COL (For comprehensive vibration assessment program)	3.9.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	loading, and inspections and walkdowns for steam, feedwater, and condensate systems. The test program also should include details of actions to be taken if acceptance criteria are not satisfied. Further information on test procedure is addressed in **RG 1.20**.		
	F. The methods and procedures to process the test data for meaningful interpretation of the vibration behavior of various components should be provided. Vibration interpretation should include the amplitude, frequency content, stress state, and possible effects on safety functions. There should be detailed analysis of bias errors and uncertainties of instrumentation, data acquisition systems, and models to estimate loading functions from the measured data.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	G. Vibration predictions, test acceptance criteria and bases, and permissible deviations from the criteria should be provided before the test.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	H. The applicant/licensee is expected to provide a summary evaluation of plant startup and power ascension to the staff within 90 days of plant startup. If full licensed power is not achieved in that time period, the applicant/licensee is expected to provide a supplemental report within 30 days after achieving full licensed power.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	I. There should be walkdown inspections during and visual and nondestructive surface inspections after completion of the vibration tests. The inspection program description should include the areas subject to inspection, the methods of inspection, the design access provisions to the reactor internals,	N/A-COL (For comprehensive vibration assessment program)	3.9.1

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	and the equipment to be used for such inspections, which preferably should follow the removal of the internals from the reactor vessel. Where removal is not feasible, the inspections should be by means of equipment appropriate for in-situ inspection. The areas inspected should include all load-bearing interfaces, core restraint devices, high-stress locations, and locations critical to safety functions. MSL valves also should be inspected if adverse flow effects (flow-induced acoustic and structural resonances) are observed during the startup test. For later reactor internals with the same design, size, configuration, and operating conditions as the prototype, the vibration test program should comply with the requirements of the appropriate non-prototype program as specified in **RG 1.20**.		
3.9.2-SAC-05	For requirements of **GDCs 2, 4, 14, and 15** dynamic system analyses should confirm the structural design adequacy of the reactor internals and the reactor coolant piping (unbroken loops) to withstand the dynamic loadings of the most severe LOCA in combination with the SSE. Where a substantial separation between the forcing frequencies of the LOCA (or SSE) loading and the natural frequencies of the internal structures can be demonstrated, the analysis may treat the loadings statically. Evaluations performed under **SRP Section 3.6.3**, address review of applications that propose to eliminate consideration of design loads of the dynamic effects of pipe rupture. Evaluation in this Section should interface with the evaluation in Section 3.6.3. The most severe dynamic effects from LOCA loadings generally result from a	Y	3.9.2

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	postulated double-ended rupture of a primary coolant loop near a reactor vessel inlet or outlet nozzle with the reactor in the most critical normal operating mode. However, all other postulated break locations should be evaluated and the location producing the controlling effects should be identified. Mathematical models used for dynamic system analysis for LOCAs in combination with SSE effects should include the following:		
	A. Modeling should include reactor internals and dynamically-related piping, pipe supports, components, and fluid-structure interaction effects when applicable. Typical diagrams and the modeling basis should be developed and described.	Y	3.9.2
	B. Mathematical models should typify system such structural characteristics as flexibility, mass inertia effect, geometric configuration, and damping (including possible coexistence of viscous and Coulomb damping).	Y	3.9.2
	C. Any system structural partitioning and directional decoupling in the dynamic system modeling should be justified.	N/A-OTHER	N/A
	D. The effects of flow upon the mass and flexibility properties of the system should be addressed.	Y	3.9.2
	Typical diagrams and the basis for postulating the LOCA-induced forcing function should be provided, including a description of the governing hydrodynamic equations and the assumptions for mathematically tractable flow path geometries, tests for determining flow coefficients, and any semi-empirical formulations and scaled model flow testing for determining pressure differentials or velocity distributions. The acceptability of the hydraulic analysis, as reviewed on request, is based on established engineering practice and generic topical reviews by the staff.	Y	3.9.2

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	The methods and procedures for dynamic system analyses should be described, including the governing equations of motion and the computational scheme for deriving results. Time domain forced-response computation is acceptable for both LOCA and SSE analyses. The response spectrum modal method may be used for SSE analysis. The stability of such elements in compression as the core barrel and the control rod guide tubes under outlet pipe rupture loadings should be investigated. Either response spectra or time histories may be used for specifying seismic input motions of the SSE at the reactor core supports. The criteria for acceptance of the analytical results are described in **SRP Sections 3.9.3 and 3.9.5**. For PWRs, the criteria and review methods for verifying whether the applicant has appropriately addressed asymmetric blowdown loadings on reactor internals are described in **SRP Section 3.9.5**.		
3.9.2-SAC-06	For requirements of **GDC 1**, as to the correlation of tests and analyses of reactor internals, the applicant should address the following items to ensure the adequacy and sufficiency of the test and analysis results.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	A. Comparison of the measured response frequencies with the analytically obtained natural frequencies of the reactor internals for validation of the mathematical models used in the analysis. Comparison of the measured and predicted damping factors as a function of natural frequencies for validation of the damping assumed in the analysis.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	B. Comparison of the analytically obtained mode shapes with the shape of	N/A-COL	3.9.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	measured motion for identification of the modal combination or verification of a specific mode.	(For comprehensive vibration assessment program)	
	C. Comparison of the response amplitude time variation and the frequency content from test and analysis for verification of the postulated forcing function.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	D. Comparison of the measured amplitudes, frequencies, and time variations of loads with those predicted by test-analysis combination method for validation of the predicted forcing function.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	E. Comparison of the maximum responses from test and analysis for verification of stress levels.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	F. Comparison of the mathematical model for dynamic system analysis under operational flow transients and under combined LOCA and SSE loadings for similarities.	N/A-COL (For comprehensive vibration assessment program)	3.9.1
	G. Comparison of measurements and predictions of any adverse flow phenomena (e.g., flow-excited acoustic and/or structural resonances) for validation of the model(s) predicting the loading induced by the phenomena.	N/A-COL (For comprehensive vibration assessment program)	3.9.1

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3.9.2-SAC-07	For new applications, test specifications should be in accordance with **ASME OM-S/G-1990**, "Standards and Guides For Operation of Nuclear Power Plants," **Part 3**, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," **and Part 7**, "Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems."	N/A-COL (For comprehensive vibration assessment program)	3.9.1
SRP 3.9.3	**ASME Code Class 1, 2, and 3 Components and Component Supports, and Core Support Structures (R2, 03/2007)**		
3.9.3-AC-01	**10 CFR 50.55a** and **10 CFR 50, Appendix A, General Design Criterion (GDC) 1** as they relate to structures and components being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.	Y (Per AREVA Topical Report ANP-10264)	3.9.3
3.9.3-AC-02	**GDC 2** and **10 CFR Part 50, Appendix S**, as they relate to structures and components important to safety being designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.	Y (Per AREVA Topical Report ANP-10264)	3.9.3
3.9.3-AC-03	**GDC 4** as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents.	Y (Per AREVA Topical Report ANP-10264)	3.9.3
3.9.3-AC-04	**GDC 14** as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y (Per AREVA Topical Report ANP-10264)	3.9.3
3.9.3-AC-05	**GDC 15** as it relates to the reactor coolant system and associated auxiliary, control and protection systems being designed with sufficient margin to assure that the	Y (Per AREVA Topical	3.9.3

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	design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Report ANP-10264)	
3.9.3-AC-06	**10 CFR Part 52** requirements that the components and component supports, and core support structures will be designed and built in accordance with the certified design.	Y (Per AREVA Topical Report ANP-10264)	3.9.3
3.9.3-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.3-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.9.3-SAC-01	<u>Loading Combinations, System Operating Transients, and Stress Limits</u>. The design and service loading combinations, including system operating transients, and the associated design and service stress limits considered for each component and its supports should be sufficiently defined to provide the basis for design of Code Class 1, 2, and 3 components and component supports, and core	Y (Per AREVA Topical Report ANP-10264)	3.9.3

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	support structures for all conditions. The acceptability of the combination of design and service loadings (including system operating transients), applicable to the design of Class 1, 2, and 3 components and component supports, and core support structures, and of the designation of the appropriate design or service stress limit for each loading combination, is judged by comparison with positions stated in Appendix A, and with appropriate standards acceptable to the staff, developed by professional societies and standards organizations. The design criteria for internal parts of components such as valve discs, seats, and pump shafting should comply with applicable Code or Code Case criteria. In those instances where no Code criteria exist, the design criteria are acceptable if they ensure the structural integrity of the part such that no safety-related functions are impaired.		
3.9.3-SAC-02	<u>Design and Installation of Pressure Relief Devices</u>. The applicant should use design criteria for pressure relief installations specified in **Appendix O, ASME Code, Section III, Division 1**, "Rules for the Design of Safety Valve Installations." In addition, the following criteria are applicable:	Y (Per AREVA Topical Report ANP-10264)	3.9.3
	A. Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analyzing for the stress at any piping location should be that sequence which is estimated to induce the maximum instantaneous value of stress at that location.		
	B. Stresses should be evaluated, and applicable stress limits should be satisfied for all components of the pipe run and connecting systems and the pressure relief valve station, including supports and all connecting welds between these		

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	components.		
	C. In meeting the stress limit requirements, the contribution from the reaction force and the moments resulting from that force should include the effects of a Dynamic Load Factor (DLF) or should use the maximum instantaneous values of forces and moments for that location as determined by the dynamic hydraulic/structural system analysis. This requirement should be satisfied in demonstrating satisfaction of all design limits at all locations of the pipe run and the pressure relief valve for Class 1, 2, and 3 piping. A DLF of 2.0 may be used in lieu of a dynamic analysis to determine the DLF		
	The SAR should also include a description of the calculational procedures, computer programs, and other methods to be used in the analysis. The analysis should include the time history or equivalent effects of changes of momentum due to fluid flow changes of direction. The fluid states considered should include postulated water slugs where water seals are used and subcooled or saturated liquid if such fluid can be discharged under postulated transient or accident conditions. Applicants for plants utilizing suppression pools should also consider the applicable pool dynamic loads on the safety relief valve system. Stress computations and stress limits must be in accord with applicable rules of the Code.		
3.9.3-SAC-03	Component Supports. The component support designs should provide adequate margins of safety under all combinations of loadings. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination should meet the criteria in Appendix A, **Regulatory Guides (RG)**	Y (Per AREVA Topical Report ANP-10264)	3.9.3

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	1.124 and **RG 1.130** and **Subsection NF of the Code**		
	A. Component supports of active pumps and valves should be considered in context with the other features of the functionality assurance and seismic qualification program as presented in **SRP Section 3.10**. If the component support deformation can be expected to affect the operability requirements of the supported component, then deformation limits should also be specified. Such deformation limits should be compatible with the operability requirements of the supported components. These deformation limits should be incorporated into the functionality assurance and seismic qualification program. In establishing allowable equipment deformations, the possible movements of the support base structures must be taken into account.	Y (Per AREVA Topical Report ANP-10264)	3.9.3 3.10
	B. Criteria for snubber functionality assurance should contain the following elements:	Y (Per AREVA Topical Report ANP-10264)	3.9.3
	i. <u>Structural Analysis and Systems Evaluation</u> Systems and components which utilize snubbers as shock and vibration arresters should be analyzed to ascertain the interaction of such devices with the systems and components to which they are attached. Snubbers may be used as shock and vibration arresters and in some instances as dual purpose snubbers, and when so used fatigue strength should be considered. Important factors in the fatigue evaluation include: (1) unsupported system component movement or amplitude, (2) force imparted to snubber and corresponding reaction on system or component due to restricting motion (damped amplitude), (3) vibration frequency or number of load cycles, and		

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	(4_ verification of system or component and snubber fatigue strength. Snubbers used as shock arresters need not undergo a fatigue evaluation if it can be demonstrated that: (a) the number of load cycles which the snubber will experience during normal plant operating conditions is small (<2500) or (b) motion during normal plant operating conditions does not exceed snubber dead band. Snubbers utilized in systems or components which may experience high thermal growth rates, either during normal operating conditions or as a result of anticipated transients, should be checked to ensure that such thermal growth rates do not exceed the snubber lock-up velocity.		
	ii. <u>Characterization of Mechanical Properties</u> An important aspect of the structural analysis is realistic characterization of snubber mechanical properties (i.e., spring rates) in the analytical model. Since the "effective" stiffness of a snubber is generally greater than that for the snubber support assembly (i.e., the snubber plus clamp, transition tube extension, back-up support structure, etc.) the snubber response characteristics may be "washed out" by the added flexibility in the support structure. The combined effective stiffness of the snubber and support assembly should therefore be considered in evaluating the structural response of the system or component. Snubber spring rate should be determined independent of clearance/lost motion, activation level, or release rate. The stiffness should be based on structural and hydraulic compliance, and the effects of temperature should		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	be considered. The snubber end fitting clearance, mismatch of end fitting clearances, mismatch of activation and release rates, and lost motion should be minimized and should be considered when calculating snubber reaction loads and stress which are based on a linear analysis of the system or component. This is especially important in multiple snubber applications where mismatch of end fitting clearance has a greater effect on the load sharing of these snubbers than does the mismatch of activation level or release rate. Equal load sharing of multiple snubber supports should not be assumed if mismatch in end fitting clearance exists.		
	iii. Design Specifications. The required structural and mechanical performance of snubbers is determined from the applicant's structural analysis described in Subsections II.3.B(i) and (ii). The snubber Design Specification is the instrument provided by the purchaser to the supplier to ensure that the requirements are met. The Design Specification should contain: (1) the general functional requirements, (2) operating environment, (3) applicable codes and standards, (4) materials of construction and standards for hydraulic fluids and lubricants, (5) environmental, structural, and performance design verification tests, including the required dynamic qualification, testing and extrapolation methods supporting qualification of large bore hydraulic snubbers with		

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	rated load capacities of 50 Kips or more as recommended in **NUREG/CR-5416**. (6) production unit functional verification tests and certification, (7) packaging, shipping, handling, and storage requirements, and (8) description of provisions for attachments and installation. In addition, the procurement program should include provisions for the snubber manufacturer to submit quality assurance and assembly quality control procedures for review and acceptance by the purchaser.		
	iv. Use of Additional Snubbers Snubbers could in some instances be installed during or after plant construction. These snubbers may not have been included in the design analysis. This could occur as a result of unanticipated piping vibration, as discussed in **SRP Section 3.9.2** , or interference problems during construction. The effects of such snubbers should be fully evaluated and documented to demonstrate that normal plant operations and safety are not diminished.		
SRP 3.9.4	**Control Rod Drive Systems (R3, 03/2007)**		
3.9.4-AC-01	**GDC 1** and **10 CFR 50.55a**, as they relate to the CRDS, require that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.	Y	3.9.4
3.9.4-AC-02	**GDC 2**, as it relates to CRDS, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.	Y	3.9.4

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
3.9.4-AC-03	**GDC 14**, as it relates to CRDS, requires that the RCPB portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.	Y	3.9.4
3.9.4-AC-04	**GDC 26**, as it relates to CRDS, requires that the CRDS be one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including anticipated operational occurrences.	Y	3.9.4
3.9.4-AC-05	**GDC 27**, as it relates to CRDS, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.	Y	3.9.4
3.9.4-AC-06	**GDC 29**, as its relates to CRDS, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.	Y	3.9.4
3.9.4-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.4-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are	N/A-COL	N/A

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	necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations		
3.9.4-SAC-01	The descriptive information is determined to be sufficient provided the minimum requirements for such information meet **Section 3.9.4 of Regulatory Guide (RG) 1.29**.	Y	3.9.4
3.9.4-SAC-02	Construction (as defined in **NCA-1110 of Section III of the ASME Code**) should meet the following codes and standards utilized by the nuclear industry which have been reviewed and found acceptable:		
	A. For pressurized portions of equipment classified as Quality Group A, B, C (RG 1.26): **Section III of the ASME Code, Class 1, 2, or 3 as appropriate**.	Y	3.9.4
	B. For pressurized portions of equipment classified as Quality Group D (RG 1.26): (i) **Section VIII, Division 1, of the ASME Code** for vessels and pump casings. (ii) For piping systems (American National Standards Institute, ANSI): B16.5 Steel Pipe Flanges and Flanged Fittings B16.9 Steel Butt Welding Fittings B16.11 Steel Socket Welding Fittings B16.25 Butt Welding Ends	N/A-CLASS	3.9.4 & FSAR Table 3.2-1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B16.34 Steel Valves with Flanged and Butt Welding Ends B31.1 Power Piping MSS-SP-25 Marking for Valves, Fittings, Flanges, and Unions		
	C. For nonpressurized equipment (Non-ASME Code): Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any decreases in design margins should be provided.	N/A-COL	3.9.4
3.9.4-SAC-03	For the various design and service conditions defined in NB-3113 of Section III of the ASME Code, load combination sets are as given in **SRP Section 3.9.3**. The stress limits applicable to pressurized and nonpressurized portions of the control rod drive systems should be as given in SRP Section 3.9.3 for the response to each loading set.	Y	3.9.4
	For BWRs, the CRDS design should adequately consider water hammer loads to assure that system safety functions can be achieved.	N/A-BWR	N/A
3.9.4-SAC-04	The operability assurance program will be acceptable provided the observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements.	Y	3.9.4
SRP 3.9.5	**Reactor Pressure Vessel Internals (R3, 03/2007)**		
3.9.5-AC-01	**GDC 1** and **10 CFR 50.55a** require that reactor internals be designed to quality standards commensurate with the importance of the safety functions performed.	Y	3.9.5
3.9.5-AC-02	**GDC 2** requires that reactor internals be designed to withstand the effects of	Y	3.9.5

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	natural phenomena such as earthquakes without loss of capability to perform safety functions		
3.9.5-AC-03	**GDC 4** requires that reactor internals be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCAs. Dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for piping.	Y	3.9.5
3.9.5-AC-04	**GDC 10** requires that reactor internals be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition normal operation, including the effects of anticipated operational occurrences.	Y	3.9.5
3.9.5-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.5-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has	N/A-COL	N/A

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	been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.9.5-SAC-01	Requirements for loads, loading combinations, and limits applicable to those portions of reactor internals constructed to Subsection NG of the ASME Code are presented in **SRP Section 3.9.3**.	Y	3.9.5
3.9.5-SAC-02	The design and construction of the core support structures should comply with the requirements of **Subsection NG, "Core Support Structures," of the ASME Code** and **SRP Section 3.9.3**.	Y	3.9.5
3.9.5-SAC-03	The design criteria, loading conditions, and analyses that provide the bases for the design of reactor internals other than the core support structures should meet the guidelines of **NG-3000** and be constructed not to affect the integrity of the core support structures adversely (**NG-1122**). If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified.	Y	3.9.5
3.9.5-SAC-04	Deformation limits for reactor internals should be established by the applicant and presented in the safety analysis report. The basis for these limits should be included. The stresses of these displacements should not exceed the specified limits. The requirements for dynamic analysis of these components are addressed in **SRP Section 3.9.2**.	Y	3.9.5
3.9.5-SAC-05	The reactor internals should be designed to accommodate asymmetric blowdown loads from postulated pipe ruptures. The applicant's evaluation of such loads should demonstrate that they do not exceed the limits imposed by the applicable	Y	3.9.5

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	codes and standards. Where double-ended guillotine break of reactor coolant piping is postulated, criteria for evaluating loading transients and structural components are specified in **NUREG-0609**.		
3.9.5-SAC-06	Potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals (including the steam dryer in BWRs) should be adequately addressed in accordance with relevant criteria stated in the **Appendix to this SR Section**.	Y	3.9.5
SRP 3.9.6	**Inservice Testing of Pumps and Valves (R3, 03/2007)**		
3.9.6-AC-01	**10 CFR 50.55a** and **10 CFR Part 50, Appendix A, GDC 1** as they relate to pumps, valves, and dynamic restraints important to safety being designed, fabricated, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed.	Y	3.9.6
3.9.6-AC-02	**GDC 2**, as it relates to pumps, valves, and dynamic restraints important to safety to withstand the effects of natural phenomena combined with the effects of normal and accident conditions.	Y	3.9.6
3.9.6-AC-03	**GDC 4**, as it relates to designing pumps, valves, and dynamic restraints important to safety to accommodate the effects of and to be compatible with the environment conditions associated with normal operation, maintenance, testing, and postulated accidents.	Y	3.9.6
3.9.6-AC-04	**GDC 14**, as it relates to designing pumps, valves, and dynamic restraints that form the reactor coolant boundary so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.	Y	3.9.6
3.9.6-AC-05	**GDC 15**, as it relates to pumps, valves, and dynamic restraints that form the	Y	3.9.6

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	reactor coolant system being designed with sufficient margin to ensure that the design conditions are not exceeded.		
3.9.6-AC-06	**GDC 37**, as it relates to designing the emergency core cooling to permit periodic functional testing to ensure the leak tight integrity and performance of its active components.	Y	3.9.6
3.9.6-AC-07	**GDC 40**, as it relates to designing periodic functional testing of the containment heat removal system to ensure the leak tight integrity and performance of its active components.	Y	3.9.6
3.9.6-AC-08	**GDC 43**, as it relates to designing the containment atmospheric cleanup systems to permit periodic functional testing to ensure the leak tight integrity and the performance of the active components.	Y	3.9.6
3.9.6-AC-09	**GDC 46**, as it relates to designing the cooling water system to permit periodic functional testing to ensure the leak tight integrity and performance of the active components.	Y	3.9.6
3.9.6-AC-10	**GDC 54**, as it relates to designing piping systems penetrating containment with the capability to test periodically the operability of the isolation valves and determine valve leakage acceptability.	Y	3.9.6
3.9.6-AC-11	**Appendix B to 10 CFR Part 50**, as it relates to quality assurance in the design, fabrication, construction, and testing safety-related pumps, valves, and dynamic restraints.	Y	3.9.6
3.9.6-AC-12	**10 CFR 50.55a(c)-(e)**, in so far as it incorporates the ASME Code, Section III, as it relates to qualification of mechanical equipment and supports.	Y	3.9.6
3.9.6-AC-13	**10 CFR 50.55a(f)** for pumps and valves, and **10 CFR 50.55a(g)** for dynamic	Y	3.9.6

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	restraints, whose function is required for safety in the IST program, as it relates to assessing operational readiness.		
3.9.6-AC-14	**10 CFR 50.55a(b)(3)(ii)**, as it relates to requirements for an MOV testing program.	Y	3.9.6
3.9.6-AC-15	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.9.6-AC-16	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.9.6-SAC-01	<u>Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints</u> A. For new plant applications, safety-related pump, valve, and piping designs should include provisions to allow testing of pumps and valves at the maximum flow specified in the plant accident analyses. B. Functional design and qualification of each safety-related pump and valve should be accomplished such that each pump and valve is capable of performing its intended function for a full range of system differential pressure and flow, ambient temperatures, and available voltage (as applicable) under all	Y	3.9.6

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	conditions ranging from normal operating to design-basis accident conditions. C. Acceptance criteria for the design of dynamic restraints (snubbers) are provided in **SRP Section 3.9.3**. D. Acceptance criteria for the design and installation of safety and relief valves are provided in **SRP Section 3.9.3**. E. Acceptance criteria for the seismic and dynamic qualification of mechanical and electrical equipment are provided in **SRP Section 3.10**. F. As required by **GDC 14**, safety-related valves that are part of the RCPB should be designed and tested such that these valves will not experience any abnormal leakage, or increase in leakage, from their loading, as addressed in **SRP Section 3.10**. G. For new plant applications, dynamic restraints in safety-related systems must include provisions to allow access for IST program activities.		
3.9.6-SAC-02	<u>Inservice Testing Program for Pumps</u> A. The scope of the applicant's test program is acceptable if it includes all of the ASME Code Class 1, 2, and 3 pumps described in **10 CFR 50.55a(f)** and **Subsection ISTA-1100 of the OM Code** and, in addition, includes pumps not categorized as ASME Code Class 1, 2, or 3 but which the staff considers to be safety-related. Since the pump test program is based on the detection of changes in the hydraulic and mechanical condition of a pump relative to a reference test specified in **Subsection ISTB-3000 of the OM Code**, the establishment of a set of reference values and a consistent test method are basic criteria of the program. B. The pump test program is acceptable if it meets the requirements for	Y	3.9.6

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	establishing reference values and the periodic testing schedule described in **Subsection ISTB-3000 of the OM Code**. Subsections **ISTB-3000, ISTB-5000, and ISTB-6000 of the OM Code** establish the allowable ranges of IST quantities (e.g., flow rates and pressure differential), corrective actions, and vibration tests. The pump test schedule is required to comply with these rules. C. The frequency of ISTs and test parameters are acceptable if the provisions of **Subsection ISTB-3000 of the OM Code** are met. D. The methods of measurement are acceptable if the test program meets the requirements of **Subsection ISTB-5000 of the OM Code** with regard to instruments, pressure measurements, rotational speed, vibration measurements, and flow measurements. E. The instruments are acceptable if they meet the accuracy and range requirements of **Subsection ISTB-3500 of the OM Code**. F. The duration of the test is acceptable if the provisions of **Subsection ISTB-5000of the OM Code** are met.		
3.9.6-SAC-03	Inservice Testing Program for Valves A. To be acceptable, the SAR valve test list must contain all safety-related ASME Code Class 1, 2, and 3 valves required by **10 CFR 50.55a(f)** and the **OM Code**, except those nonsafety-related valves excepted by **Subsection ISTC-1200 of the OM Code**. It should also include valves not categorized as ASME Code Class 1, 2, or 3 but which are safety related. The SAR valve list must include a valve categorization that complies with the provisions of **Subsection ISTC-1300 of the OM Code**. The SAR should list each specific valve to be tested under the rules of **Subsection ISTA-1100 of the OM Code** by type,	Y N/A-BWR (for 3.9.6-SAC-03.C.vi.(3)) N/A-PAS (for 3.9.6-SAC-03.C.iii.(3)(e))	3.9.6 N/A N/A

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	valve identification number, code class, and valve category. B. The valve test procedures, acceptance criteria, and corrective actions are acceptable if the provisions of **Subsection ISTC of the OM Code**, as incorporated by reference in **10 CFR 50.55a**, are met with regard to preservice and periodic inservice valve testing. C. The following provides additional acceptance criteria for specific valve or actuator types, and leak testing: i. Inservice Testing Program for Motor-Operated Valves (1) In addition to the IST program requirements in the **ASME OM Code** incorporated by reference in **10 CFR 50.55a(f)**, **10 CFR 50.55(b)(3)(ii)** requires establishment of a program to ensure that the safety-related MOVs continue to be capable of performing their design-basis safety functions. **GL 96-05** provides additional guidance for the periodic verification of MOV design-basis capability. Furthermore, **ASME Code Cases OMN-1 and OMN-11**, as accepted by the NRC staff with conditions in **RG 1.192**, provide an alternative method to MOV stroke-time testing that also satisfies the requirement in **10 CFR 50.55a** to supplement the OM Code IST provisions with a program to ensure that safety-related MOVs continue to be capable of performing their safety functions. (2) Periodic testing should be conducted that objectively demonstrates continuing MOV capability to perform its safety functions to open and close, as applicable, under design-basis conditions. Where testing is not conducted under design-basis conditions (e.g., under environmental conditions), an analysis combined with test results should demonstrate		

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	the continued design-basis capability of the MOV. (3) The interval between testing demonstrating continued design-basis capability should not exceed 5 years or three refueling outages, whichever is longer, unless a longer interval can be justified. Longer design-basis verification intervals may be justified through implementation of **ASME Code Case OMN-1**, as accepted in **RG 1.192**. (4) Acceptance criteria for successful completion of the preservice and inservice testing of MOVs should include the following: (a) Consistent with the safety function, the valve should fully open and/or the valve fully close or both. Diagnostic equipment should indicate hard seat contact. (b) The testing should demonstrate adequate margin with respect to the design basis, including consideration of diagnostic equipment inaccuracies, degraded voltage, control switch repeatability, load sensitive MOV behavior, and margin for degradation. (c) The maximum torque and/or thrust (as applicable) achieved by the MOV, allowing sufficient margin for diagnostic equipment inaccuracies and control switch repeatability, should not exceed the allowable structural and undervoltage motor capability limits for the individual parts of the MOV. ii. Inservice Testing Program for Power-Operated Valves Other Than Motor-Operated Valves (1) Safety-related POVs should be qualified to perform their design-basis		

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	functions either before installation or as part of preservice tests. (2) **NRC Regulatory Issue Summary 2000-03** provides guidance for the development of IST programs for POVs that incorporate the lessons learned from MOV analysis and tests in response to **GL 89-10**. (3) Class 1E SOVs are to be verified to function as designed. Each SOV should be verified, to the extent practical, to be capable of performing its safety functions for the electrical power supply amperage and voltage at design basis extremes. iii. Inservice Testing Program for Check Valves (1) Preservice tests should be conducted on each check valve. Each check valve should be tested in the open and closed direction, consistent with the safety function and under normal operating system conditions. Piping system design features should be able to accommodate all applicable check valve testing equipment and procedures. (a) Diagnostic equipment or nonintrusive techniques that monitor internal component conditions or measure such parameters as fluid flow, disk position, disk movement, disk impact forces, leak tightness, leak rates, degradation, and disk stability should be used, if practical, for preoperational testing and later during IST. The equipment and its operating principles should be described and the techniques justified. The operation and accuracy of the diagnostic equipment and techniques should be verified during preoperational testing. (b) To the extent practical, testing should be performed under temperature and flow conditions that would exist during normal		

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	operation as well as cold shutdown. Testing at temperature and flow conditions that may exist in other modes should also be conducted if such conditions are significant. (c) Test results should identify the minimum flow that will open the valve to the full-open position. (d) Testing should include the effects of rapid pump starts and stops, as expected for system operating conditions. The testing should include any other reverse flow conditions that may occur during expected system operating conditions. (2) Nonintrusive (diagnostic) techniques should be used to periodically assess degradation and the performance characteristics of check valves. (3) Acceptance criteria for the successful completion of the preservice and inservice testing of check valves should include the following: (a) During all test modes that simulate expected system operating conditions, the valve disk should fully open or fully close as expected based on the direction of the differential pressure across the valve. (b) Valve disk positions should be determinable without disassembly. (c) Testing should verify that there is free disk movement to and from the seat. (d) The valve disk should be stable in the open position under normal and other minimum system operating fluid flow conditions. (e) For passive plant designs, testing should verify that the valve disk		

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	moves freely off the seat under normal and other minimum expected differential pressure conditions. (4) In **10 CFR 50.55a(b)(3)(iv)**, the regulations specify conditions for the application of **Appendix II to the OM Code**. Those requirements must be satisfied when applying Appendix II to the OM Code. iv. <u>Pressure Isolation Valve Leak Testing</u> (1) Pressure isolation valves (PIVs) are the two normally closed valves, in series, within the RCPB (defined in **10 CFR Part 50**) that isolate the reactor coolant system from an attached low-pressure system. PIVs are classified as A or AC in accordance with the provisions of **Subsection ISTC-1300 of the OM Code**. (2) PIV seat leakage rate tests should be conducted on each individual PIV in accordance with **Subsection ISTC-3630 of the OM Code**. The plant technical specifications or SAR should specify the allowable leak rates and test intervals for each PIV. The maximum allowable leak rate for each PIV at full reactor pressure should be less than 1.9 liters per minute (L/m) (0.5 gallons per minute (gpm)) per nominal inch of valve size, and not to exceed 19 L/m (5 gpm). The test interval should be 18 months or every refueling outage, whichever is longer. (3) The applicant's SAR should provide a list of PIVs that includes the allowable leak rate for each valve. v. <u>Containment Isolation Valve Leak Testing</u> (1) Containment isolation valves (CIVs) should be leak tested in accordance with **Appendix J to 10 CFR Part 50**.		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	(2) The plant SAR should provide a list of CIVs that includes the allowable leak rate for each valve or valve combination. vi. <u>Inservice Testing Program for Safety and Relief Valves</u> (1) Safety and relief valves, including thermal relief valves and vacuum relief valves, are pressure relief devices that protect systems (or portion of systems) that perform a function in shutting down the reactor to the safe-shutdown condition, maintaining the safe-shutdown condition, or mitigating the consequences of an accident. (2) Safety and relief valve tests should be conducted in accordance with **Appendix I to the OM Code**. (3) Stroke tests should be performed for dual-function safety and relief valves (e.g., boiling-water reactor main steam automatic depressurization system safety/relief valves). (4) Power-operated relief valves should be tested in accordance with **Subsection ISTC-5100** for Category B valves and **Subsection ISTC-5240** for Category C valves. (5) The test equipment, including gages, transducers, load cells, and calibration standards, used to determine valve set-pressure is acceptable if the overall combined accuracy does not exceed ±1 percent of the indicated (measured) set pressure. (6) The plant SAR should provide a list of safety and relief valves that includes the set pressure and allowable tolerances for each valve. vii. <u>Inservice Testing Program for Manually Operated Valves</u> (1) The plant SAR should provide a list of manually operated valves,		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	including their safety-related function. (2) In **10 CFR 50.55a(b)(3)(vi)**, the regulations take exception to the 5-year exercise interval for manually operated valves allowed in the OM Code. In accordance with the regulations, manual valves must be exercised on at least a 2-year interval. (3) The valve should exhibit the full range of obturator position set forth in the design bases. viii. <u>Inservice Testing Program for Explosively Actuated Valves</u> (1) At least 20 percent of the charges in explosively actuated valves should be fired and replaced at least every 2 years. (2) If a charge fails to fire, all charges with the same batch number should be removed and replaced with charges from a different batch.		
3.9.6-SAC-04	<u>Inservice Testing Program for Dynamic Restraints</u> A. The IST program for dynamic restraints is acceptable if it meets the requirements of the **ASME Code, Section XI**, or the **ASME OM Code** as incorporated by reference in **10 CFR 50.55a**. The IST program for dynamic restraints must comply with these provisions. B. In **10 CFR 50.55a(b)(3)(v)**, the regulations state that **Subsection ISTD of the ASME OM Code**, 1995 edition through the latest edition and addenda and incorporated by reference in **10 CFR 50.55a(b)(3)**, may be applied in place of the requirements for snubbers in the **ASME Code, Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b)**, by making appropriate changes to technical specifications or licensee-controlled documents. The regulations also state that preservice and inservice examinations must be performed using the VT-3	Y	3.9.6

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	visual examination method prescribed in **IWA-2213**. C. The FSAR should identify and tabulate all safety-related components that use snubbers in their support systems. The tabulation should include the following information: i. Identification of the systems and components in those systems that use snubbers ii. The number of snubbers used in each system and on components in that system iii. The type(s) of snubber (hydraulic or mechanical) and the corresponding supplier iv. Specification whether the snubber was constructed in accordance with the **ASME Code, Section III, Subsection NF** v. Statement whether the snubber is used as a shock, vibration, or dual purpose snubber vi. For snubbers identified as either dual purpose or vibration arrestor type, an indication of whether both snubber and component were evaluated for fatigue strength, the evaluation is performed under **SRP Section 3.9.3 Appendix A**. D. The applicant should provide assurance that all snubbers are properly installed before preoperational piping vibration and plant startup tests. The applicant may use visual observation of piping systems and measurement of thermal movements during plant startup tests to verify that snubbers are operable (not locked up). The piping preoperational vibration and plant startup test programs should discuss the provisions for such examinations and measurements as		

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	described in **SRP Section 3.9.2**. E. The applicant should discuss accessibility provisions for maintenance, inservice inspection and testing, and possible repair or replacement of snubbers consistent with the provisions of the applicable NRC standard technical specifications.		
3.9.6-SAC-05	Relief Requests and Proposed Alternatives A. The applicant should identify the component identified for which it requests relief: i. Name and number as given in SAR ii. Component functions iii. ASME Code, Section III, Code Class iv. Valve category as defined in Subsection ISTC-1300 of the OM Code v. Pump group as defined in Subsection ISTB-2000 of the OM Code B. The applicant should identify the ASME OM Code requirement(s) from which it is requesting relief. C. The applicant should specify the basis under which it is requesting relief and then explain why complying with the OM Code is impractical. D. For alternatives to the OM Code requirements, the applicant should provide sufficient details to demonstrate that (1) the proposed alternative will provide an acceptable level of quality and safety, or (2) compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. E. The applicant should specify a schedule for the implementation of the relief	N/A-OTHER (No Relief Requests or Alternatives)	N/A

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	request or alternative. F. The approval of relief requests or alternatives involves the following: i. Approval of relief for impractical code requirements Pursuant to 10 CFR 50.55a(f)(6)(i) for pumps and valves, and 10 CFR 50.55a(g)(6)(i) for dynamic restraints, the Commission may grant relief from impractical code requirements because of design limitations upon application by the applicant. The NRC will consider the burden on the applicant as a factor in its review and evaluation. ii. Approval of alternatives to the OM Code requirements Pursuant to 10 CFR 50.55a(a)(3), the staff may authorize alternatives to IST program requirements of the OM Code if the applicant has adequately demonstrated either of the following: (1) Proposed alternatives to the Code requirements or portions thereof will provide an acceptable level of quality and safety. (2) Compliance with the Code requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.		
3.9.6-SAC-06	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the preservice testing program, inservice testing program, inservice inspection program, and motor-operated valve testing program are reviewed in accordance with 10 CFR 50.55a(f), 10 CFR 50.55a(g) and 10 CFR 50.55a(b)(3)(ii). The implementation milestones for the specific programs are specified below and included as license conditions for preservice testing and	N/A-COL	N/A

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	motor-operated valve testing programs: A. Preservice testing program Per ASME OM Code, Subsection ISTA-2000, defines the preservice test period as the period of time following the completion of construction activities related to the component and before first electrical generation by nuclear heat. B. IST program Per ASME OM Code, Subsection ISTA-2000, prior to first electrical generation by nuclear heat C. Inservice inspection program related to dynamic restraints Per ASME Code, Section XI, IWA-2430(b), before placement of the plant into commercial service D. MOV program Per ASME OM, Subsection ISTA-2000, prior to first electrical generation by nuclear heat		
SRP 3.9.7	**Risk-Informed Inservice Testing of Pumps and Valves (08/1998)**	N/A-COL	N/A
SRP 3.9.8	**Risk-Informed Inservice Testing of Piping (09/2003)**	N/A-COL	N/A
SRP 3.10	**Seismic and Dynamic Qualification of Mechanical and Electrical Equipment (R3, 03/2007)**		
3.10-AC-01	**GDC 1 and 30** as they relate to qualifying equipment to appropriate quality standards commensurate with the importance of the safety functions to be performed.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-AC-02	**GDC 2** and **Appendix S to 10 CFR Part 50** as they relate to designing equipment	Y (Per AREVA Topical	3.10

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	to withstand the effects of natural phenomena such as earthquakes.	Report ANP-10277)	
3.10-AC-03	**GDC 4** as it relates to qualifying equipment as capable of withstanding the dynamic effects associated with external missiles and internally generated missiles, pipe whip, and jet impingement forces.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-AC-04	**GDC 14** as it relates to qualifying equipment associated with the reactor coolant boundary so that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-AC-05	**10 CFR Part 50, Appendix B**, as it relates to qualifying equipment using the quality assurance criteria provided.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-AC-06	**10 CFR part 50, Appendix B, Criterion III**, as it relates to verifying and checking the adequacy of design, such as by the performance of a suitable test program, among other things, and which specifically requires that a test program used to verify the adequacy of a specific design feature shall include suitable qualifications testing of a prototype unit under the most adverse design conditions.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.10-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed	N/A-COL	N/A

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	inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
3.10-SAC-01	The qualification of electrical equipment and its supports should meet the requirements and recommendations of **American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Std 344-1987** as endorsed by **RG 1.100**. (Subsequent revision to RG 1.100 will provide guidance with exceptions for use of **Appendix QR-A of ASME QME-1-2007** for seismic qualification of active mechanical equipment and other qualifications of mechanical components, and **IEEE Std 344-2004** for seismic qualification for Class 1E equipment.) These documents are generally applicable to all types of equipment and should be used to the extent practicable for the qualification of mechanical equipment. Specifically, conformance to the following criteria should be demonstrated. A. Qualification for Equipment Functionality i. Tests and analyses are required to confirm the functionality of all mechanical and electrical equipment during and after an earthquake of magnitude up to and including the OBE and SSE and for all static and dynamic loads from normal, anticipated operational occurrence, and accident conditions. Before SSE qualification, the applicant should demonstrate that the equipment can withstand the equivalent effect of five OBE excitations without loss of structural integrity. Analyses alone, without	Y (Per AREVA Topical Report ANP-10277)	3.10

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	testing, are acceptable as a basis for qualification only if the necessary function of the equipment is ensured by its structural integrity alone. When complete testing is impractical, a combination of tests and analyses is acceptable. Equipment that has been previously qualified by means of tests and analyses equivalent to those described herein is acceptable provided that the applicant submits proper documentation of such tests and analyses. ii. Equipment should be tested in the operational condition. Functionality should be verified during and/or after the testing, as applicable to the equipment being tested. Loadings simulating those of plant normal operation, such as thermal and flow-induced loading, if any, should be concurrently superimposed upon the seismic and other pertinent dynamic loading to the extent practicable. iii. Response spectrum or time history methods should specify the characteristics of the required seismic and dynamic input motions. These characteristics, derived from the seismic and dynamic analyses of the structures or systems, should be representative of the input motions at the equipment mounting locations, except as noted in subsection II.2 (under SRP Acceptance Criteria) of this SRP Section. iv. For seismic and dynamic loads, the actual test input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated (i.e., the test response spectrum (TRS) should closely resemble and envelop the required response spectrum (RRS) over the critical frequency range).		

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	v. Since seismic and dynamic load excitation generally has a broad frequency content, multi-frequency vibration input motion should be used. However, single-frequency input motion, such as sine beats, is acceptable provided the characteristics of the required input motion indicate that the motion is dominated by one frequency (e.g., by structural filtering effects), or the anticipated response of the equipment is adequately represented by one mode, or in the case of structural integrity assurance, the input has enough intensity and duration to produce sufficiently high levels of stress for such assurance. Components that have been previously tested to **IEEE Std 344-1971** should be reevaluated to justify the appropriateness of the input motion used and requalified if necessary. vi. For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally. Components that have been previously tested to **IEEE Std 344-1971** should be requalified using biaxial test input motions unless the applicant provides justification for using a single-axis test input motion. vii. Dynamic coupling between the equipment and related systems, if any,		

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	such as connected piping and other mechanical components, should be considered. viii. The fixture design should simulate the actual service mounting and should not cause any extraneous dynamic coupling to the test item. ix. For pumps and valves, the loads imposed by the attached piping should be considered. To ensure functionality under combined loadings, the stresses resulting from the applied test loads should envelop the specified service stress limit for the intended function of the component. Stresses in valve bodies and pump casings should be limited to the particular material's elastic limit when the pump or valve is subject to the combination of normal operating loads, SSE, and other applicable dynamic loads. x. If the dynamic testing of a pump or valve assembly proves to be impracticable, static testing of the assembly is acceptable provided that the end loadings are conservatively applied and are equal to or greater than postulated event loads, all dynamic amplification effects are accounted for, the component is in the operating mode during and after the application of loads, and an adequate analysis is made to show the validity of the static application of loads. xi. The in situ application of vibratory devices to simulate the seismic and dynamic vibratory motions on a complex active device is acceptable to confirm the functionality of the device when the applicant shows that a meaningful test can be made in this way. xii. The test program may be based on selective testing of a representative number of components according to type, load level, size, and the like on a prototype basis.		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	xiii. Selection of damping values for equipment to be qualified should be made in accordance with **RG 1.61** and **ANSI/IEEE Std 344-1987**. Higher damping values may be used if justified by documented test data with proper identification of the source and mechanism. xiv. When complete testing is not practicable, the features listed below should be incorporated into a test and analysis functionality assurance program for pumps and valves. Similar programs can be developed for other types of equipment. (1) Simple and passive elements, such as valve and pump bodies and their related piping and supports, may be analyzed to confirm structural integrity under postulated event loadings. However, complex active devices such as pump motors, valve operators and gate or disk assemblies, and other electrical, mechanical, pneumatic, or hydraulic appurtenances which are vital to the pump or valve operation should be tested for functionality. (2) The following analyses are acceptable provided they are correlated to classical problems, elementary laboratory tests, or in situ tests: (a) An analysis is performed to determine the vibratory input to the valve or pump. (b) An analysis is performed to determine the system's natural frequencies and the movement of the pump or valve during the dynamic events. (c) An analysis is performed to determine the pressure differential and the impact energy on a valve disc during a loss-of-coolant accident (LOCA) and to verify the design adequacy of the disc.		

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(d) An analysis is performed to determine the forcing functions of the axial and radial loads imposed on a pump rotor because of a LOCA, such that combined LOCA and vibratory effects on the shaft and rotor assembly can be evaluated. (e) An analysis is performed to determine the speed of the pump shaft as a result of postulated events and to compare it with the design critical speed. (f) An analysis is performed to verify the design adequacy of the wall thickness of valve and pump pressure retaining bodies. (g) An analysis is performed to determine the natural frequencies of a pump shaft and rotor assembly to ascertain whether they are within the frequency range of the vibratory excitations. If the minimum natural frequency of the assembly is beyond the excitation frequencies, a static deflection analysis of the shaft is acceptable to account for dynamic effects. If the assembly's natural frequencies are close to the excitation frequencies, an acceptable dynamic analysis must be performed to determine the structural response of the assembly to the excitation frequencies. (h) When analyses are used for qualification, the combination of multimodal and multidirectional responses should be made in accordance with **RG 1.92**. B. Design Adequacy of Supports i. Analyses or tests should be performed for all supports of mechanical and electrical equipment to ensure their structural capability. ii. The analytical results should include the required input motions to the		

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	mounted equipment as obtained and characterized in the manner stated in subsection II.1.A.iii above, and the combined stresses of the support structures should be in accordance with the criteria specified in **SRP Section 3.9.3**. iii. Supports should be tested with equipment installed or with a dummy simulating the equivalent equipment inertial mass effects and dynamic coupling to the support. If the equipment is installed in a nonoperational mode for the support test, the response in the test at the equipment mounting location should be monitored and characterized in the manner stated in subsection II.1.A.iii above. In such a case, equipment should be tested separately for functionality, and the actual input motion to the equipment in this test should be more conservative in amplitude and frequency content than the monitored response from the support test. iv. The criteria of subsections II.1.A.iii thru II.1.A.xiii above apply when tests are conducted on the equipment supports. C. Verification of Seismic and Dynamic Qualification. The seismic and dynamic qualification testing performed in accordance with **ANSI/IEEE Std 344-1987**, as endorsed by **RG 1.100**, Revision 2, as part of an overall qualification program should be performed in the sequence indicated in **Section 6 of IEEE Std 323-1974** (endorsed with exceptions by **RG 1.89**)		
3.10-SAC-02	Instrumentation described in **RG 1.97**, including associated mountings, should be tested under appropriate seismic and dynamic loadings as described in the regulatory guide, thereby ensuring that the instruments will continue to monitor plant variables and systems after a seismic event and/or accident.	Y (Per AREVA Topical Report ANP-10277)	3.10

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3.10-SAC-03	If the applicant proposes qualification by an experience-based approach, the details of the experience database, including applicable implementation methods and procedures to ensure structural integrity and functionality of the in-scope mechanical and electrical equipment, must meet the functionality of equipment for the defined load condition as presented in paragraphs 1 and 2 above. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and the equivalent effect of five postulated occurrences of OBE in combination with other relevant static and dynamic loads.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-SAC-04	**GDC 1** and **10 CFR Part 50, Appendix B, Criteria XVII** establish requirements for records concerning the qualification of equipment. To satisfy these requirements, complete and auditable records must be available, and the applicant must maintain them, for the life of the plant, at a central location. These files should describe the qualification method used for all equipment in sufficient detail to document the degree of compliance with the criteria of this SRP section. These records should be updated and kept current as equipment is replaced, further tested, or otherwise further qualified. The equipment qualification file should contain a list of all systems, equipment, and the equipment support structures, as defined in the second paragraph of subsection I of this SRP Section. The equipment list should identify which equipment is supplied by the nuclear steam supply system (NSSS) and which equipment is supplied by the balance of plant (BOP). The equipment qualification file should also include qualification summary data sheets for each piece of equipment (i.e., each mechanical and electrical component of each system) which summarize the component's qualification. These data sheets should include the	Y (Per AREVA Topical Report ANP-10277)	3.10

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	following information: A. Identification of equipment, including vendor, model number, and location within each building. Valves that are part of the reactor coolant pressure boundary (RCPB) should be so identified. B. Physical description, including dimensions, weight, and field mounting condition, and identification of whether the equipment is pipe-, floor-, or wall-supported. C. A description of the equipment's function within the system. D. Identification of all design (functional) specifications and qualification reports and their locations. Functional specifications for active valve assemblies should conform to **RG 1.148**. E. Description of the required loads and their intensities for which the equipment must be qualified. F. If qualification by test, identification of the test methods and procedures, important test parameters, and a summary of the test results. G. If qualification by analysis, identification of the analysis methods and assumptions and comparisons between the calculated and allowable stresses and deflections for critical elements. H. If qualification by an experience-based approach, identification of the type of experience and the source of experience database. I. The natural frequency (or frequencies) of the equipment. J. Identification of whether the equipment may be affected by vibration fatigue cycle effects and a description of the methods and criteria used to qualify the equipment for such loading conditions.		

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	K. Indication of whether the equipment has met the qualification requirements. L. Availability for inspection (i.e., statement of whether the equipment is already installed). M. A compilation of the required response spectra (or time history) and corresponding damping for each seismic and dynamic load specified for the equipment together with all other loads considered in the qualification and the method of combining all loads.		
3.10-SAC-05	**GDC 14** requires, in part, that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage. 10 CFR 50, Appendix A, **GDC 30** further requires, in part, that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. As discussed under acceptance criteria in **SRP Section 3.9.6**, to satisfy these requirements, the qualification program for valves that are part of the RCPB should include testing or testing and analyses demonstrating that these valves will not experience any leakage, or increase in leakage, as a result of any loading or combination of loadings for which the valves must be qualified.	Y (Per AREVA Topical Report ANP-10277)	3.10
3.10-SAC-06	The implementation of the qualification program described above should be documented in the following ways:		
	A. The preliminary safety analyses report (PSAR) or DC application should contain the following: i. A detailed description of NSSS and architect/engineer (A/E) practice followed in qualification, including criteria, methods, and procedures used in conducting testing and analysis, which demonstrate the extent of	Y (Per AREVA Topical Report ANP-10277)	3.10

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	compliance with the criteria set forth in subsections II.1 thru II.5 above. ii. If equipment qualification by using earthquake experience data and/or test experience data is proposed, a detailed description of the experience database, including applicable implementation methods and procedures to ensure structural integrity and functionality of the in-scope mechanical and electrical equipment subject to the defined load condition as presented in paragraphs 1 and 2 above. Supporting documentation for equipment identified in the database should confirm that such equipment remained functional during and after an SSE and the equivalent effect of five postulated occurrences of OBE in combination with other relevant static and dynamic loads. Note: For electrical equipment earthquake and/or experience data should not be used without adequate justification. iii. Information regarding administrative control of component qualification, especially a description of the equipment qualification file, the handling of documentation, internal acceptance review procedures, identification of the scope of NSSS and A/E suppliers, and the procedures for interchange of information between NSSS, A/E, equipment vendors, and testing laboratories.		
	B. In addition to the information contained in the PSAR, as revised, the final safety analyses report (FSAR) should contain the following: i. A list of all systems required to perform the functions defined in the second paragraph of subsection I of this SRP section. ii. A description of the results of any in-plant tests, such as in situ impedance	Y (Per AREVA Topical Report ANP-10277)	3.10

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	tests, and any plans for operational tests which will be used to confirm the qualification of any item of equipment.		
	C. The seismic qualification report (SQR) should contain the following: i. The list of systems required to perform the functions defined in the second paragraph of subsection I of this SRP section. ii. The list of equipment, and its supports, associated with each system and any other equipment required in accordance with the second paragraph of Subsection I of this SRP section. iii. The summary data sheets for each piece of equipment (i.e., each component) listed. iv. A detailed description of the experience database similar to item II.6.A.ii above for in-scope equipment not covered in DC.	Y (Per AREVA Topical Report ANP-10277)	3.10
	D. COL applications should include the information described in subsections II.6.A, II.6.B, and II.6.C, as well as the following: i. A description of the environmental parameters applicable to the specific plant and its equipment qualification program. ii. Documentation to demonstrate that properly defined and enveloped seismic and dynamic input response spectra have been applied to the specific plant and its equipment qualification program.	N/A-COL	N/A
SRP 3.11	**Environmental Qualification of Mechanical and Electrical Equipment (R3, 03/2007)**		
3.11-AC-01	**10 CFR 50.49**, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."	Y (Per AREVA Topical	3.11

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		Report ANP-10277)	
3.11-AC-02	**10 CFR 50.67**, "Accident Source Term.”	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-03	**10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1**, "Quality Standards and Records."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-04	**GDC 2**, "Design Bases for Protection Against Natural Phenomena."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-05	**GDC 4**, "Environmental and Dynamic Effects Design Bases."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-06	**GDC 23**, "Protection System Failure Modes."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-07	**10 CFR Part 50, Appendix B,** "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," **Section III**, "Design Control."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-08	**10 CFR Part 50, Appendix B,** "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," **Section XI**, "Test Control."	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-AC-09	**10 CFR Part 50, Appendix B,** "Quality Assurance Criteria for Nuclear Power	Y	3.11

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	Plants and Fuel Reprocessing Plants," **Section XVII**, "Quality Assurance Records."	(Per AREVA Topical Report ANP-10277)	
3.11-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.11-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.11-SAC-01	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment," Revision 1, July 1981 provides staff positions applicable to existing plants for assessing the compliance of an environmental qualification program with 10 CFR 50.49. For future plants, **Regulatory Guide 1.89** provides the principal guidance for implementing the requirements and criteria of **10 CFR 50.49** for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. However, certain	Y (Per AREVA Topical Report ANP-10277)	3.11

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	NUREG-0588 Category I guidance may be used if relevant guidance is not provided in Regulatory Guide 1.89. NUREG-0588 includes two sets of qualification criteria, Category I and Category II. Category I refers to IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Category I applies to plants whose CP SERs were dated after July 1, 1974. Category II refers to **IEEE Std 323-1971**, and is not applicable to any future plants.		
3.11-SAC-02	**IEEE Std 323** contains the principles and criteria that are generic to the environmental qualification process. The following clarification related to the criteria in IEEE Std 323 should be considered. IEEE Std 323 requires that the service environment, including the installed configuration of the equipment, be considered as part of the qualification process. In meeting this requirement, the potential for flooding of electrical equipment that are installed above the flood level, but are subject to water and moisture intrusion, should be considered as part of environmental qualification. Operating experience (e.g., **Information Notice 89-63**) shows that electrical enclosures that are located above the flood level and are subject to water and moisture intrusion could result in submergence of electrical components inside the enclosures, if the enclosures do not have drainage holes. The reviewer should confirm that equipment in such locations, whose design is such that water accumulation is possible, should have measures to preclude such accumulation (e.g., enclosure drain holes) or the affected equipment should be qualified for the anticipated submergence.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-03	**Regulatory Guide 1.40**, “Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants,” endorses **IEEE Std 334**, "IEEE Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors	Y (Per AREVA Topical Report ANP-10277)	3.11

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	Installed Inside the Containment of Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1E motors, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental design and qualification of Continuous-Duty Class 1E Motors.		
3.11-SAC-04	**Regulatory Guide 1.63**, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants," endorses **IEEE Std 317**, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." These documents contain general guidance that is acceptable to the staff for the environmental design and qualification of electrical penetration assemblies, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental design and qualification of electrical penetration assemblies.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-05	**Regulatory Guide 1.73**, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants," endorses **IEEE Std 382**, "IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental design and qualification of Class 1E electric valve operators, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental design and qualification of Class 1E electric valve operators.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-06	**Regulatory Guide 1.89**, "Environmental Qualification of Certain Electric Equipment Important to Safety in Nuclear Power Plants," provides guidance for implementing the requirements and criteria of **10 CFR 50.49** for environmental qualification of electrical equipment that is important to safety and located in a	Y (Per AREVA Topical Report ANP-10277)	3.11

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	harsh environment. **Regulatory Guide 1.89** endorses the provisions of **IEEE Std 323** as being acceptable to the staff, and provides supplementary guidance for satisfying the Commission's regulations regarding the environmental qualification of electrical equipment located in a harsh environment.		
3.11-SAC-07	**Regulatory Guide 1.97**, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," provides guidance acceptable to the staff for the environmental qualification of the post-accident monitoring equipment described in Subsection I, Item 1(f), of this SRP section, as well as instruments and controls for the equipment described in Subsection I, Items 1(a) to 1(e), of this SRP section. These criteria, as supplemented by those of **Regulatory Guide 1.89**, should be used to evaluate the environmental qualification of the I&C equipment.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-08	Draft **Regulatory Guide 1.131**, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants,” endorses **IEEE Std 383**, "Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E electric cables and field splices, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental qualification of Class 1E electric cables and field splices. Pending issuance of the “Final” version, the Draft version of **RG 1.131** may be used as guidance.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-09	**Regulatory Guide 1.156**, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants,” endorses **IEEE Std 572**, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the	Y (Per AREVA Topical Report ANP-10277)	3.11

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	environmental qualification of Class 1E connection assemblies, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental qualification of Class 1E connection assemblies.		
3.11-SAC-10	**Regulatory Guide 1.158**, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants," endorses **IEEE Std 535**, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations." These documents contain guidance acceptable to the staff for the environmental qualification of Class 1E lead storage batteries, and should be used in conjunction with **NUREG-0588** and **Regulatory Guide 1.89**, as appropriate, for evaluating the environmental qualification of lead storage batteries.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-11	**Regulatory Guide 1.180**, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," provides guidance acceptable to the staff for determining electromagnetic compatibility for I&C equipment during service. These criteria, as supplemented by those of **Regulatory Guide 1.89**, should be used to evaluate the environmental design and qualification of safety-related I&C equipment. New digital systems and new advanced analog systems may require susceptibility testing for electromagnetic interference/radio-frequency interference (EMI/RFI) and power surges, if the environments are significant to the equipment being qualified. The functional descriptions of I&C equipment are provided in SRP Chapter 7.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-12	**Regulatory Guide 1.183**, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions. These criteria, as supplemented	Y (Per AREVA Topical Report ANP-10277)	3.11

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	by those of **Regulatory Guide 1.89**, should be used to evaluate the accident source term used in the environmental design and qualification of equipment important to safety. **10 CFR 50.67**, “Accident Source Term,” provides the requirements for licensees to revise the accident source term used in design basis radiological analyses for plants licensed prior to January 10, 1997. Radiation dose and dose rate used to determine the radiation environment for qualification of electrical and mechanical equipment must be based on an NRC staff approved source term and methodology, as discussed in **NUREG-0588** and as supplemented by **Section II.B.2 of NUREG-0737**, “Clarification of TMI Action Plan Requirements,” and **NUREG-0718**, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License,” or as discussed in **NUREG-1465**, "Accident Source Terms for Light-Water Nuclear Power Plants." The radiation environment must be based on the integrated effects of the normally expected radiation environment over the equipment's installed life, plus the effects associated with the most severe design basis event during or following which the equipment is required to remain functional. The effects of beta radiation must also be considered in the qualification process. The effects of radiation exposure due to recirculatory fluid must be considered for equipment located outside the containment. The staff's definition of what constitutes a mild radiation environment for electronic components, such as semiconductors or electronic components containing organic material, differs from that for other equipment and is as stated in **NUREG-1503**, “Final SER ABWR, Chapter 3, Design of Structures, Components, Equipment, and Systems,” and **NUREG-1793**, “Final SER AP1000, Chapter 3, Design of		

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	Structures, Components, Equipment, and Systems," is that a mild radiation environment for electronic equipment is a total integrated dose less than 10 Gy (1E3 rad), and a mild radiation environment for other equipment is less than 100 Gy (1E4 rad). Environmental qualification for electrical equipment located in a ?Radiation harsh" environment (i.e., locations where radiation is the only harsh environmental condition) can be accomplished in accordance with **10 CFR 50.49(f)(4)** using analysis of test data (from identical materials) combined with radiation test information (i.e., partial test data), and appropriate consideration of margin and aging effects for nonmetallic components/materials when sufficient documentation is available to preclude the need for a type test.		
3.11-SAC-13	The effects of chemical exposure must be addressed in the environmental qualification process. The concentration of chemicals used for qualification must be equivalent to, or more severe than that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling system initiation, or recirculation phase). If the chemical composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical environment that results from a single failure in the spray system must be assumed. If only demineralized water spray is used, then the effect of the demineralized water spray must be included in the equipment qualification.	Y (Per AREVA Topical Report ANP-10277)	3.11
3.11-SAC-14	Mechanical components must be designed to be compatible with postulated environmental conditions, including those associated with loss-of-coolant accidents (LOCAs). A process must be established to determine the suitability of materials, parts, and equipment needed for safety-related functions, and to verify that the design of such materials, parts, and equipment is adequate. Also,	Y (Per AREVA Topical Report ANP-10277)	3.11

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	equipment records must be maintained, and these records must include the results of tests and material analyses used as part of the environmental design and qualification process for each component. For mechanical equipment, the staff concentrates its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). The reviewer confirms that the applicant has (1) identified safety-related mechanical equipment located in harsh environment areas, including its required operating time; (2) identified nonmetallic subcomponents of such equipment; (3) identified the environmental conditions and process parameters for which this equipment must be qualified; (4) identified nonmetallic material capabilities; and (5) evaluated environmental effects.		
3.11-SAC-15	For electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design/purchase" specifications for the equipment. The specifications must contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. A well-supported maintenance/surveillance program, in conjunction with a good preventive maintenance program, is sufficient to ensure that equipment that meets the design/purchase specifications is qualified for the designed life. Compliance with **10 CFR 50.65**, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and associated guidance in **Regulatory Guide 1.160** are sufficient to provide reasonable assurance that environmental considerations established during design are reviewed every refueling outage and maintained on a continuing basis to ensure that the qualified design life has not been reduced by thermal, radiation, and/or cyclic degradation resulting from	Y (Per AREVA Topical Report ANP-10277)	3.11

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	unanticipated operational occurrences or service conditions. Modification to the replacement program and/or replacement of equipment should be based on the review of maintenance/surveillance data.		
3.11-SAC-16	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the environmental qualification program are reviewed in accordance with **10 CFR 50.49**. The implementation milestone for the environmental qualification program is to have all qualification requirements met prior to the loading of fuel. Implementation is required by a license condition.	N/A-COL	3.11
SRP 3.12	**ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports (03/2007)**		
3.12-AC-01	**10 CFR 50.55a** and **GDC 1** as they relate to piping systems, pipe supports, and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	Y (Per AREVA Topical Report ANP-10264)	3.12
3.12-AC-02	**GDC 2** and **10 CFR Part 50, Appendix S** with regard to design transients and resulting load combinations for piping and pipe supports necessary to withstand the effects of earthquakes combined with the effects of normal or accident conditions.	Y (Per AREVA Topical Report ANP-10264)	3.12
3.12-AC-03	**GDC 4**, with regard to piping systems and pipe supports important to safety, being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal as well as postulated events, such as Loss-of-Coolant Accident (LOCA) and dynamic effects.	Y (Per AREVA Topical Report ANP-10264)	3.12
3.12-AC-04	**GDC 14**, with regard to the RCPB of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of	Y	3.12

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	abnormal leakage, of rapidly propagating failure, and of gross rupture.		
3.12-AC-05	**GDC 15**, with regard to the reactor coolant systems and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design condition of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Y	3.9.2
3.12-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.12-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.12-SAC-A	<u>Piping Analysis Methods</u>		
	i. <u>Experimental Stress Analysis Methods</u> If experimental stress analysis methods are used in lieu of analytical methods for Seismic Category I ASME Code and non-Code piping system designs, the applicant should provide sufficient information to show the validity of the	N/A-OPT (See SRP 3.12, 3.12-SAC_A.ii, 3.12-SAC_A.iii,	3.12

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	design. It is recommended, prior to use of the experimental stress analysis methods, that details of the method as well as the scope and extent of its application, be submitted for approval. The experimental stress analysis methods provided in **Appendix II to ASME Code, Section III, Division 1** are applicable.	3.12-SAC_A.iv, 3.12-SAC_A.v, 3.12-SAC_A.vi, 3.12-SAC_A.vii, & 3.12-SAC_A.viii)	
	ii. Modal Response Spectrum Method The SRP acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2** are Applicable	Y (Per AREVA Topical Report ANP-10264)	3.9.2
	iii. Response Spectra Method- Independent Support Motion Method This method may be used in lieu of the response spectra method when there is more than one supporting structure. The acceptance criteria provided in **NUREG-1061, Volume 4** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	iv. Time History Method The SRP acceptance criteria provided in **SRP Section 3.7.2, Subsection II.6** are Applicable	Y (Per AREVA Topical Report ANP-10264)	3.12
	v. Inelastic Analysis Method If inelastic analysis methods are used for the piping design, the applicant will provide sufficient information to show the validity of the analysis. It is recommended, prior to use of the inelastic analysis method that details of the method, as well as the scope and extent of its application and acceptance criteria, be submitted for approval. The inelastic analysis methods provided in **SRP Section 3.9.1, Subsection II.4** are applicable.	N/A-OPT (See SRP 3.12, 3.12-SAC_A.ii, 3.12-SAC_A.iii, 3.12-SAC_A.iv, 3.12-SAC_A.v, 3.12-SAC_A.vi, 3.12-SAC_A.vii, &	3.12

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		3.12-SAC_A.viii)	
	vi. Small Bore Piping Method The SRP acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2(A)** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	vii. Nonseismic/Seismic Interaction (II/I) The acceptance criteria provided in **Section 3.9.2, Subsection II.2.(K)** are Applicable	Y (Per AREVA Topical Report ANP-10264)	3.12
	viii. Category I Buried Piping, Conduits, and Tunnels The acceptance criteria provided in **SRP Section 3.7.3, Subsection II.12** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
3.12-SAC-B	Piping Modeling Techniques		
	i. Computer Codes The acceptance criteria provided in **SRP Section 3.9.1, Subsection II.2** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	ii. Dynamic Piping Model The acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	iii. Piping Benchmark Program The computer programs are benchmarked with the appropriate NRC benchmarks.	Y (Per AREVA Topical Report ANP-10264)	3.12
	iv. Decoupling Criteria The acceptance criteria provided in **SRP Section 3.7.2, Subsection II.3(b)** are	Y (Per AREVA Topical	3.12

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	applicable.	Report ANP-10264)	
3.12-SAC-C	<u>Piping Stress Analysis Criteria</u>		
	i. <u>Seismic Input</u> The acceptance criteria provided in **SRP Section 3.7.2 Subsection II.5** are applicable.	Y	3.7.2
	ii. <u>Design Transients</u> The acceptance criteria provided in **SRP Section 3.9.1, Subsection II.1** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.9.1
	iii. <u>Loadings and Load Combinations</u> The acceptance criteria provided in **SRP Section 3.9.3, Subsection II.1** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	iv. <u>Damping Values</u> The acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2(L)** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	v. <u>Combination of Modal Responses</u> The acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2(E)** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	vi. <u>High-Frequency Modes</u> The acceptance criteria provided in **SRP Section 3.9.3, Subsection II.2** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	vii. <u>Fatigue Evaluation for ASME Code Class 1 Piping</u>	Y	3.12

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	The acceptance criteria in **Section III of the ASME Code** are applicable.	(Per AREVA Topical Report ANP-10264)	
	viii. Fatigue Evaluation of ASME Code Class 2 and 3 Piping The acceptance criteria for provided in **Section III of the ASME Code** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	ix. Thermal Oscillations in Piping Connected to the RCS The operating experience insights contained in **NRC Bulletin (BL) 88-08 and Supplements** are applicable for the identification and evaluation of piping systems susceptible to thermal stratification, cycling, and striping.	Y	3.12
	x. Thermal Stratification The operating experience insights contained in **NRC BL 79-13** and **BL 88-11** are applicable for the identification and evaluation of long runs of horizontal piping susceptible to thermal stratification.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xi. Safety Relief Valve Design, Installation, and Testing The acceptance criteria provided in **SRP Section 3.9.3, Subsection II.2** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xii. Functional Capability The acceptance criteria provided in **NUREG-1367**, “Functional Capability of Piping Systems,” may be used to ensure piping functionality under level D loading conditions. Alternative criteria will be reviewed on a case by case basis.	Y (Per AREVA Topical Report ANP-10264)	3.12

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	xiii. <u>Combination of Inertial and SAM Effects</u> The acceptance criteria provided in **SRP Section 3.9.2, Subsection II.2(G)** are applicable for enveloped support motion analysis. The acceptance criteria provided in **NUREG-1061, Volume 4** are applicable for independent support motion analysis.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xiv. <u>OBE as a Design Load</u> **Appendix S to 10 CFR Part 50**, "Earthquake Engineering Criteria for Nuclear Power Plants," allows the use of operating basis earthquake ground motion. The criteria is provided in **paragraph IV.(a)(2)**. The detail criteria for use of such an option was provided in NUREG-1503, “Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design, Section 3.1.1.2.”	Y (Per AREVA Topical Report ANP-10264)	3.12
	xv. <u>Welded Attachments</u> Support members, connections, or attachments welded to piping should be designed such that their failure under unanticipated loads does not cause failure at the pipe pressure boundary. The applicant may use Code Cases for the design of the welded attachments. Acceptable Code Cases are listed in **RG 1.84**.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xvi. <u>Modal Damping for Composite Structures</u> The acceptance criteria provided in **SRP Section 3.7.2, Subsection II.13** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xvii. <u>Temperature for Thermal Analyses</u> The applicant should perform thermal expansion analyses for piping systems	Y (Per AREVA Topical	3.12

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	that operate at temperatures above or below the stress-free reference temperature. The stress-free reference temperature for a piping system is typically defined as a temperature of 70°F. The applicant should provide justification if thermal expansion analyses are not performed. The justification will be reviewed on a case by case basis.	Report ANP-10264)	
	xviii. <u>Intersystem LOCA</u> The acceptance criteria for the design of the piping system should be such that over pressurization of low-pressure piping systems due to RCPB isolation failure will not result in rupture of the low-pressure piping outside containment. The criteria provided in **Staff Requirements Memoranda (SRM)** dated June 26, 1990 **in response to Commission Papers (SECY)-90-016** dated January 12, 1990 are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xix. <u>Effects of Environment on Fatigue Design</u> The guidance provided in **Regulatory Guide 1.207** is applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
3.12-SAC-D	<u>Piping Support Design</u>		
	i. <u>Applicable Codes</u> The design of ASME Code, Section III, Class 1, 2, and 3, piping supports should comply with the design criteria requirements of **ASME Code, Section III, Subsection NF**.	Y (Per AREVA Topical Report ANP-10264)	3.12
	ii. <u>Jurisdictional Boundaries</u> The jurisdictional boundaries between pipe supports and interface attachment points should comply with **ASME Code, Section III, Subsection NF**.	Y (Per AREVA Topical Report ANP-10264)	3.12

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	iii. Loads and Load Combinations The criteria provided in **SRP Section 3.9.3, Subsection II.1** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	iv. Pipe Support Baseplate and Anchor Bolt Design The design of the pipe support baseplates and anchor bolts should comply with guidance provided in **NRC BL 79-02, Revision 2**.	Y (Per AREVA Topical Report ANP-10264)	3.12
	v. Use of Energy Absorbers and Limit Stops The evaluation typically consists of iterative response spectra analyses of the piping and support system. The analyses will be reviewed on a case by case basis.	Y (Per AREVA Topical Report ANP-10264)	3.12
	vi. Use of Snubbers The acceptance criteria provided in **SRP Section 3.9.3, Subsection II.3** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	vii. Pipe Support Stiffness The acceptance criteria provided in **SRP Section 3.9.3, Subsection II.3** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	viii. Seismic Self-Weight Excitation The acceptance criteria provided in **SRP Section 3.9.3**, are applicable for loads caused by the seismic excitation of the pipe support.	Y (Per AREVA Topical Report ANP-10264)	3.12
	ix. Design of Supplementary Steel The design of structural steel for use as pipe supports should comply with the	Y (Per AREVA Topical	3.12

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	ASME Code, Section III, Subsection NF.	Report ANP-10264)	
	x. Consideration of Friction Forces The design of sliding type supports, such as guides or box supports, should include evaluation of the friction loads induced by the pipe on the support. The applicant should provide the friction coefficients used in the evaluation. The proposed friction coefficient will be reviewed on a case by case basis.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xi. Pipe Support Gaps and Clearances Small gaps are generally provided for frame type supports. The gap allows for radial thermal expansion of the pipe and for pipe rotation. This gap must account for the diametrical expansion of the pipe due to temperature and pressure. The acceptance criteria for the minimum gap (total of opposing sides) between the pipe and the support and will be reviewed on a case by case basis.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xii. Instrumentation Line Support Criteria The acceptance criteria provided in **ASME Code, Section III, Subsection NF** are applicable.	Y (Per AREVA Topical Report ANP-10264)	3.12
	xiii. Pipe Deflection Limits The allowable deflections of the piping at support locations resulting from design loadings should be controlled to ensure that the pipe deflections do not cause the failure of the supports. This criteria will be reviewed on a case by case basis. This criteria applies to following type of pipe supports: limit stops, snubbers, rods, hangers, and sway struts.	Y (Per AREVA Topical Report ANP-10264)	3.12
SRP 3.13	**Threaded Fasteners – ASME Code Class 1, 2, and 3 (03/2007)**		

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3.13-AC-01	**10 CFR 50, Appendix A, General Design Criteria (GDC) 1 and 30**, as they relate to the requirement that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed;	Y	3.13
3.13-AC-02	**GDC 4,** as it relates to the compatibility of components with environmental conditions;	Y	3.13
3.13-AC-03	**GDC 14**, as it relates to the requirement that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture;	Y	3.13
3.13-AC-04	**GDC 31**, as it relates to the requirement that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized;	Y	3.13
3.13-AC-05	**10 CFR Part 50, Appendix B**, as it relates to controlling the cleaning of material and equipment to prevent damage or deterioration;	Y	3.13
3.13-AC-06	**10 CFR Part 50, Appendix G**, as it relates to materials testing and acceptance criteria for fracture toughness of reactor pressure boundary components;	Y	3.13
3.13-AC-07	**10 CFR 50.55a** incorporates by reference the design criteria of **ASME Code, Section III**, Class 1, 2, and 3 components. The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints are acceptable if they meet the criteria of the ASME Code, Section III, Class 1, 2, and 3 components. However, **10 CFR 50.55a(b)(4)** permits	Y	3.13

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	use of code cases that have been adopted by the staff in **Regulatory Guide (RG) 1.84** in lieu of applicable criteria of ASME Code, Section III, Class 1, 2, and 3 components;		
3.13-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
3.13-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
3.13-SAC-01	Design Aspects		
	A. Materials Selection The selection of materials used for the design of threaded fasteners is acceptable if the **ASME Code, Section III criteria shown in Table 3.13-1** of this SRP section are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems.	Y	3.13

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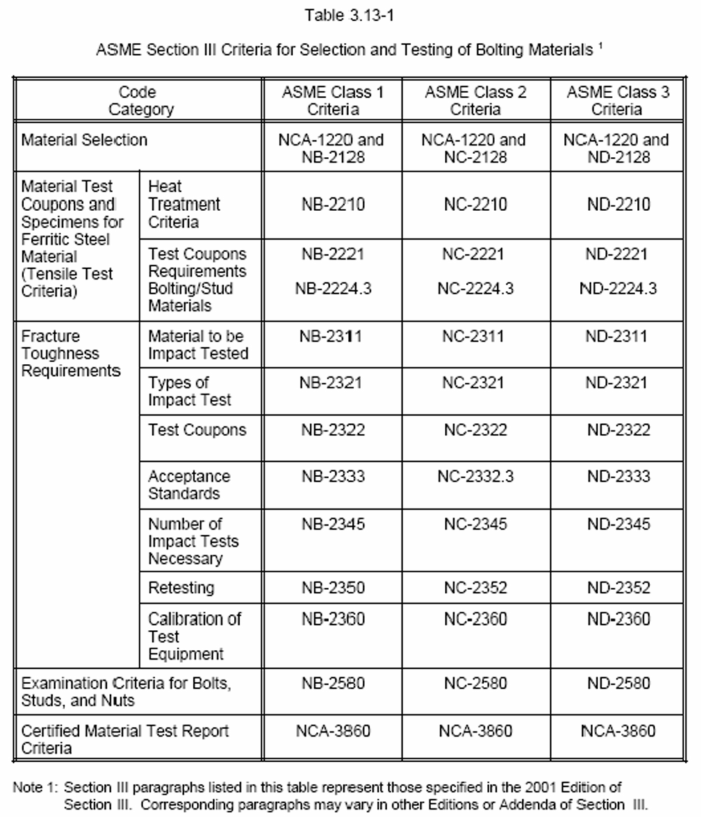
SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
			

Table 3.13-1

ASME Section III Criteria for Selection and Testing of Bolting Materials [1]

Code Category		ASME Class 1 Criteria	ASME Class 2 Criteria	ASME Class 3 Criteria
Material Selection		NCA-1220 and NB-2128	NCA-1220 and NC-2128	NCA-1220 and ND-2128
Material Test Coupons and Specimens for Ferritic Steel Material (Tensile Test Criteria)	Heat Treatment Criteria	NB-2210	NC-2210	ND-2210
	Test Coupons Requirements Bolting/Stud Materials	NB-2221 NB-2224.3	NC-2221 NC-2224.3	ND-2221 ND-2224.3
Fracture Toughness Requirements	Material to be Impact Tested	NB-2311	NC-2311	ND-2311
	Types of Impact Test	NB-2321	NC-2321	ND-2321
	Test Coupons	NB-2322	NC-2322	ND-2322
	Acceptance Standards	NB-2333	NC-2332.3	ND-2333
	Number of Impact Tests Necessary	NB-2345	NC-2345	ND-2345
	Retesting	NB-2350	NC-2352	ND-2352
	Calibration of Test Equipment	NB-2360	NC-2360	ND-2360
Examination Criteria for Bolts, Studs, and Nuts		NB-2580	NC-2580	ND-2580
Certified Material Test Report Criteria		NCA-3860	NCA-3860	NCA-3860

Note 1: Section III paragraphs listed in this table represent those specified in the 2001 Edition of Section III. Corresponding paragraphs may vary in other Editions or Addenda of Section III.

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B. Mechanical Testing, Special Process and Controls The criteria for mechanical property testing of threaded fastener materials are provided in the particular ASME Code Section II, Part A, specification under which the material was procured. The material heat treatment and tensile test coupon preparation criteria for threaded fasteners that are fabricated from ferritic materials (i.e., carbon steel, low-alloy steel, quenched and tempered steel) are acceptable if the **ASME Code, Section III criteria shown in Table 3.13-1** are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. The applicant should apply criteria of ASME Code Section III Subparagraphs NB-2200, NC-2200, ND-2200 rather than the criteria of the material specification applicable to the mechanical testing if there is a conflict between the two sets of criteria. Lubricants and sealants in mechanical connections secured by threaded fasteners should be specified to ensure they are compatible with the threaded fasteners. Any mechanical joint using threaded fasteners should be designed to preclude galvanic corrosion.	Y	3.13
	C. Fracture Toughness Requirements for Ferritic Materials The fracture toughness of ferritic bolts, studs, and nuts (i.e., made from either low-alloy steel or carbon steel materials) is acceptable if the **ASME Code, Section III criteria shown in Table 3.13-1** are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. Ferritic bolts, studs, and nuts (i.e., bolts, studs, and nuts made from either low-alloy steel or carbon steel materials) used in RCPB applications must also meet the fracture	Y	3.13

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	toughness requirements of **10 CFR Part 50, Appendix G**.		
	D. Fabrication Inspection The examination criteria for threaded fasteners are acceptable if the **ASME Code, Section III criteria shown in Table 3.13-1** are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems.	Y	3.13
	E. Quality Records The applicant should provide assurance that the CMTRs will be retained in accordance with the requirements of **10 CFR 50.70**. The CMTR should identify the material specification for which the material was procured along with the associated material properties tests (including fracture toughness tests) and inspections that apply to the particular material specification.	Y	3.13
3.13-SAC-02	Preservice and Inservice Inspection Requirements The preservice and inservice inspection provisions for mechanical joints are acceptable if the **ASME Code, Section XI criteria shown in Table 3.13-2** are appropriately specified by the applicant for ASME Code Class 1, 2, and 3 systems. For system pressure testing, the requirements of **10 CFR 50.55a(b)(2)(xxvii)** for visual examination of certain insulated bolting or studs during system pressure testing should also be identified	Y	3.13

<table>
<tr><th colspan="4">CHAPTER 3
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<tr><th>SRP Criterion</th><th>Description
(AC = Acceptance Criteria Requirement,
SAC = Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
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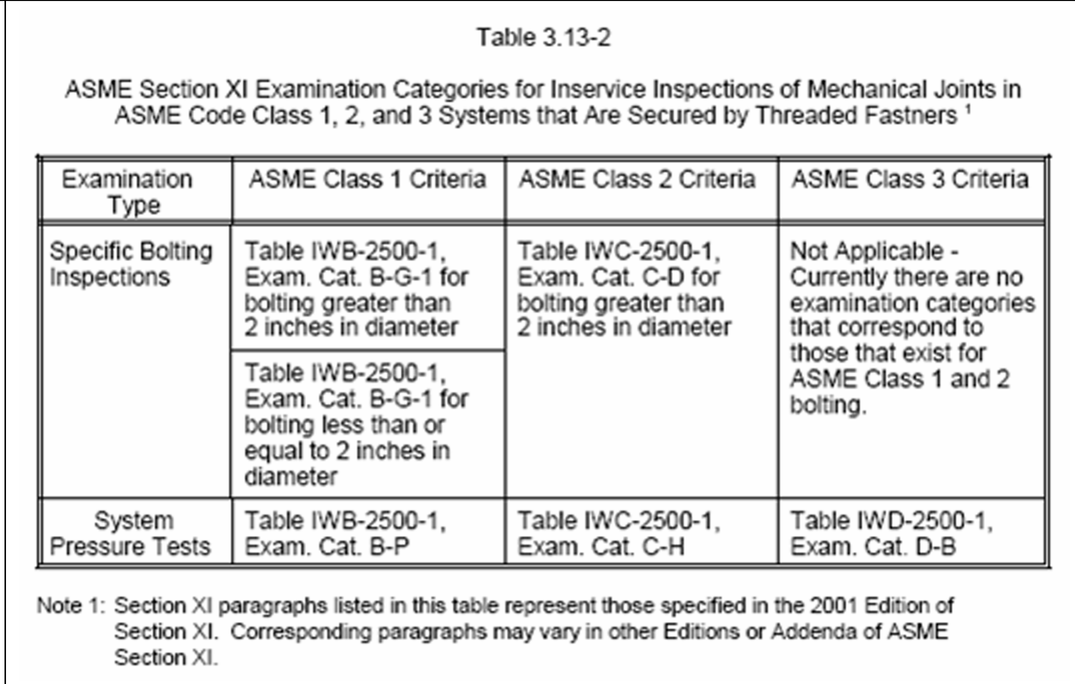

Table 3.13-2

ASME Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME Code Class 1, 2, and 3 Systems that Are Secured by Threaded Fastners [1]

<table>
<tr><th>Examination Type</th><th>ASME Class 1 Criteria</th><th>ASME Class 2 Criteria</th><th>ASME Class 3 Criteria</th></tr>
<tr><td rowspan="2">Specific Bolting Inspections</td><td>Table IWB-2500-1, Exam. Cat. B-G-1 for bolting greater than 2 inches in diameter</td><td rowspan="2">Table IWC-2500-1, Exam. Cat. C-D for bolting greater than 2 inches in diameter</td><td rowspan="2">Not Applicable - Currently there are no examination categories that correspond to those that exist for ASME Class 1 and 2 bolting.</td></tr>
<tr><td>Table IWB-2500-1, Exam. Cat. B-G-1 for bolting less than or equal to 2 inches in diameter</td></tr>
<tr><td>System Pressure Tests</td><td>Table IWB-2500-1, Exam. Cat. B-P</td><td>Table IWC-2500-1, Exam. Cat. C-H</td><td>Table IWD-2500-1, Exam. Cat. D-B</td></tr>
</table>

Note 1: Section XI paragraphs listed in this table represent those specified in the 2001 Edition of Section XI. Corresponding paragraphs may vary in other Editions or Addenda of ASME Section XI.
</td><td></td><td></td></tr>
<tr><td>BTP 3-1</td><td>Classification of Main Steam Components Other Than the Reactor Coolant Pressure Boundary for BWR Plants (R2, 03/2007)</td><td>N/A-BWR</td><td>N/A</td></tr>
<tr><td>BTP 3-2</td><td>Classification of BWR/6 Main Steam Components Other Than the Reactor Coolant Pressure Boundary (R2, 03/2007)</td><td>N/A-BWR</td><td>N/A</td></tr>
</table>

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
BTP 3-3	**Protection Against Postulated Piping Failures in Fluid Systems Outside Containment (R3, 03/2007)**	See SRP 3.6.1; SRP 9.22, 9.2.2-AC-04; SRP10.4.9, 10.4.9-SAC-02; & SRP 15.2.8, 15.2.8-SAC-07	
BTP 3-4	**Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment (R2, 03/2007)**	See SRP 3.6.2 & SRP 15.2.8, 15.2.8-SAC-07	

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 4.2	**Fuel System Design (R3, 03/2007)**		
4.2-AC-01	**10 CFR 50.46**, **10 CFR 50.34**, and **10 CFR 50.67**, as they relate to the cooling performance analysis of the ECCS using an acceptable evaluation model and establishing acceptance criteria for light-water nuclear power reactor ECCSs.	Y	4.2 Chapter 15
4.2-AC-02	**10 CFR Part 100** and **10 CFR 50.67**, as they relate to determining the acceptability of a reactor site based on calculating the exposure to an individual as a result of fission product releases to the environment following a major accident scenario.	Y	Chapter 15
4.2-AC-03	**GDC 10**, as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOO	Y	4.2
4.2-AC-04	**GDC 27**, as it relates to the reactivity control system being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions.	Y	4.2 Chapter 15
4.2-AC-05	**GDC 35**, as it relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.	Y	4.2
4.2-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and	ITAAC	Tier 1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
4.2-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
4.2-SAC-01	Design Bases The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following paragraphs:	N/A-INFO	N/A
	A. Fuel System Damage This subsection applies to normal operation, and Section 4.2 of the safety analysis report should contain the information to be reviewed. To meet the requirements of **GDC 10**, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms. Fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. Complete		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	damage criteria should address the following:		
	i. Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in **Section III of the Boiler and Pressure Vessel Code** of the American Society of Mechanical Engineers (ASME) are acceptable. Other proposed limits must be justified.	Y	4.2.1.1.2 4.2.1.5.2 4.2.1.6.3 4.2.3.4 4.2.3.5.1
	ii. The cumulative number of strain fatigue cycles on the structural members mentioned in item (i) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits must be justified.	Y	4.2.1.1.3 4.2.1.6.3 4.2.3.5.1
	iii. Fretting wear at contact points on the structural members mentioned in item (i) above should be limited. Fretting wear tests and analyses that demonstrate compliance with this design basis should account for grid spacer spring relaxation. The allowable fretting wear should be stated in the safety analysis report, and the stress and fatigue limits in items (i) and (ii) above should presume the existence of this wear.	Y	4.2.1.1.3 4.2.1.4.2 4.2.1.5.8 4.2.3.5.7
		EXCEPTION (Allowable fretting wear not addressed per AREVA Topical Report BAW-10227P-A)	4.2.3.5.7
	iv. Oxidation, hydriding, and the buildup of corrosion products (crud)	Y	4.2.1.1.4

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SRP Criterion		Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
		should be limited, with a limit specified for each fuel system component. These limits should be established based on mechanical testing to demonstrate that each component maintains acceptable strength and ductility. The safety analysis report should discuss allowable oxidation, hydriding, and crud levels and demonstrate their acceptability. These levels should be presumed to exist in items (i) and (ii) above. The effect of crud on thermal-hydraulic considerations and neutronic (AOA) considerations are reviewed as described in **SRP Sections 4.3 and 4.4**.		4.2.1.5.15 4.2.1.6.2
	v.	Dimensional changes, such as rod bowing or irradiation growth of fuel rods, fuel assemblies, control rods, and guide tubes, should be limited to prevent fuel failures or a situation in which the thermal-hydraulic limits established in **Section 4.4** are exceeded. Irradiation growth can result in a significant interference fit between the rod upper end cap and the tie plate (in a boiling-water reactor (BWR)) or the upper nozzle (in a pressurized-water reactor (PWR)), resulting in rod bowing.	Y	4.2.1.5.9 4.2.1.5.4 4.2.1.5.6 4.2.1.6.3 4.2.1.6.4 4.2.3.5.4 4.2.3.5.4.1 4.2.3.5.4.2 4.2.3.5.4.3 4.2.3.5.6
		Control blade/rod, channel, and guide tube bow as a result of (1) differential irradiation growth (from fluence gradients), (2) shadow corrosion (hydrogen uptake results in swelling), and (3) stress relaxation, which can impact control blade/rod insertability from interference problems between these components. For BWRs,	Y	4.2.1.5.10 4.2.1.6 4.2.1.7 4.2.3.5.4

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	the effects of shadow corrosion should be considered for new control blade or channel designs, dimensions (e.g., the distance between control blade and channel is important), or materials. The effects of channel bulge should also be considered for interference problems for BWRs. Design changes can alter the pressure drop across the channel wall, thus necessitating an evaluation of such changes. Channel material changes can also impact the differential growth, stress relaxation, and the amount of bulge and therefore must be evaluated. If interference is determined to be possible, tests are needed to demonstrate control blade/rod insertability consistent with assumptions in safety analyses. Additional in-reactor surveillance (e.g., insertion times) may also be necessary for new designs, dimensions, and materials to demonstrate satisfactory performance.		4.2.3.5.6 4.2.4.5
		N/A-BWR	N/A
	vi. Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation or other limits must be justified based on, but not limited to, the following minimum criteria. (1) No cladding liftoff during normal operation (2) No reorientation of the hydrides in the radial direction in the cladding (3) A description of any additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents (see Subsection II, item 1.B.vii)	Y	4.2.1.3.2 4.2.1.6.4
	vii. Because unseating a fuel bundle may challenge control	Y	4.2.1.5.5

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	rod/blade insertion, an evaluation of worst-case hydraulic loads should be performed for normal operation, AOOs, and accidents. These worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in **SRP Section 4.4**.		4.2.3.5.5
	viii. Control rod reactivity and insertability must be maintained. This requires that, at a minimum, the following may need to be reviewed: (1) Changes in control rod configuration (2) Introduction of new materials (3) Changes in neutronics and mechanical lifetime (4) Changes in mechanical design (5) The ability to exclude water/coolant if water-soluble or leachable materials (e.g., B_4C) are used Changes in mechanical and neutronics lifetimes need to be calculated using acceptable methods. Safety analyses must specifically account for the reduction in neutron-absorbing capabilities with time in-reactor.	Y	4.2.1.5.12 4.2.1.6 4.2.3.5.1 4.3
	B. <u>Fuel Rod Failure</u> This subsection applies to normal operation, AOOs, and postulated accidents. Items 1.B.i through 1.B.iii below address failure mechanisms that are more limiting during normal operation; Section 4.2 of the safety analysis report should contain the information to be reviewed. Items 1.B.iv through 1.B.viii below address failure	N/A-INFO	N/A

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	mechanisms that are more limiting during AOOs and postulated accidents; Chapter 15 of the safety analysis report usually contains the information to be reviewed. To meet the requirements of (1) **GDC 10** as it relates to SAFDLs for normal operation, including AOOs and (2) **10 CFR Part 100** as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although the staff recognizes that it is impossible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, the review must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis. Fuel rod failures can be caused by overheating, PCI, hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. When applicable, the fuel rod failure criteria should consider high burnup effects based on irradiated material properties data. Complete fuel failure criteria should address the following:		
	i. Hydriding. Both internal and external sources of hydriding can cause a zirconium alloy component to fail. To prevent failure from internal hydriding (i.e., primary hydriding), the level of moisture and other hydrogenous impurities within the fuel is kept very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 micrograms per gram	Y	4.2.1.1.4

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(µg/g) (20 parts per million (ppm)). Current specifications of the **American Society for Testing and Materials (ASTM),1989 edition, Standard C776-89, Part 45**, for uranium oxide fuel pellets state an equivalent limit of 2 µg/g (2 ppm) of hydrogen from all sources. For other materials clad in Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 milligrams of water per cubic centimeter of hot void volume within the Zircaloy cladding has been shown to be insufficient for primary hydride formation. External hydriding is caused by waterside corrosion in which the water reaction with the zirconium alloy results in zirconium hydrides as well as zirconium dioxide.		
	ii. Cladding Collapse. If axial gaps in the fuel pellet column result from densification, the cladding has the potential to collapse into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.	Y	4.2.1.3.1 4.2.1.3.2 4.2.3.1.11
	iii. Overheating of Cladding. Traditional practice assumes that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. **SRP Section 4.4** details the review of these criteria. Violation of the thermal margin criteria is not permitted for normal operation and AOOs. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes. Although a thermal margin criterion is sufficient to demonstrate that overheating from	Y	4.2.1.3.1 4.2.3.3.1 4.2.3.3.5 4.2.3.3.6 4.4 Chapter 15

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	a deficient cooling mechanism can be avoided, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. At present, there is little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.		
	iv. <u>Overheating of Fuel Pellets</u>. Traditional practice has also assumed that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and AOOs, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.	Y	4.2.3.2.3 4.4 Chapter 15
	v. <u>Excessive Fuel Enthalpy</u>. The sudden increase in fuel enthalpy from a reactivity initiated accident (RIA) below fuel melting can result in fuel failure due to pellet/cladding mechanical interaction (PCMI) (see Subsection II, item 1.B.vii). Exceeding the DNBR for a PWR or the CPR for a	Y	4.2.3.1.13 Chapter 15

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	BWR may result in cladding failure during an RIA. See Appendix B for criteria.		
	vi. Pellet/Cladding Interaction. No criterion currently exists for fuel failure resulting from PCI or PCMI. The difference between PCI and PCMI is subtle, and it is sometimes difficult to differentiate the two types of failures from visual observation of the failure. PCI is generally caused by stress-corrosion cracking due to fission product (iodine) embrittlement of the cladding, while PCMI is primarily a stress-driven failure. The design basis for PCI and PCMI can only be generally stated. Two related criteria should be applied, but they are not sufficient to preclude PCI or PCMI failures. The first criterion limits uniform strain of the cladding to no more than 1 percent. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing must demonstrate that the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1-percent strain criterion. Although observing this strain limit may preclude some PCI and PCMI failures, it will neither preclude the corrosion-assisted failures that occur at low strains nor the highly localized overstrain failures introduced by pellet chips on the outer fuel diameter. The second criterion states that fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the	N/A-OTHER	4.2.3.1.13 Chapter 15

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SRP Criterion	**Description** **(AC = Acceptance Criteria Requirement,** **SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	cladding. Avoiding fuel melting can preclude such a PCI. Note that item 1.B.iv above invoked this same criterion to ensure that overheating of the cladding would not occur. Fuel vendors have introduced fuel design limits on power maneuvering and rate of power ascension to prevent PCI or PCMI. These design limits have primarily been based on power ramp data from test reactors for a specific fuel design. Recently, however, fuel vendors have been relying more on their predictions of cladding strain and less on their power ramp data to verify that PCMI will not occur. Convincing evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.		
	vii. Bursting To meet the requirements of **10 CFR 50.46**, as it relates to ECCS performance evaluation, the ECCS evaluation model should include a calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding. **Regulatory Guide (RG) 1.157** provides guidelines for performing a realistic (i.e., best estimate) model to calculate the degree of cladding swelling and rupture. Alternatively, **Appendix K to 10 CFR Part 50** presents the acceptable features of an evaluation model for predicting the degree of swelling and	Y	4.2.3.1.2 Chapter 15

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	rupture in the Zircaloy cladding. Although fuel suppliers may use different rupture-temperature vs. differential-pressure curves, an acceptable curve should be similar to the one described in **NUREG-0630** based on similar data for a specific material. Cladding burst from non-LOCA accidents also needs to be evaluated and addressed in terms of impact on cladding temperatures and radiological consequences.		
	viii. Mechanical Fracturing A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90 percent of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.	Y	4.2.3.5.1 Chapter 15
	C. Fuel Coolability This subsection applies to postulated accidents, and Chapter 15 of the safety analysis report will contain most of the information to be reviewed. Item 1.C.v below addresses the combined effects of two accidents, and Section 4.2 of the safety analysis report should include that information. To meet the requirements of **GDC 27 and 35** as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. Coolability, or coolable geometry, has	N/A-INFO	N/A

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. This subsection also addresses control rod insertability criteria. Complete criteria should address the following:		
	i. <u>Cladding Embrittlement</u>. The ECCS performance analysis must satisfy the fuel design criteria specified within **10 CFR 50.46(b)**. These criteria ensure a coolable core geometry by preserving adequate postquench ductility in the fuel rod cladding. The current criteria require that (1) the peak cladding temperature remains below 2200 °F and (2) the peak cladding oxidation remains below 17 percent ECR. These criteria were originally developed on the basis of unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the postquench characteristics of the fuel cladding material. Rulemaking pursuant to **10 CFR 50.46** is planned to implement a performance-based test program that will dictate postquench performance requirements and provide an acceptable means to establish specific limits for new cladding materials. Future cladding alloys must comply with the postquench performance requirements specified by the new rule and provide the empirical database to support any limits assigned to the new alloy.	Y	Chapter 15

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	ii. <u>Violent Expulsion of Fuel</u>. In severe RIAs, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and produce pressure pulses in the primary system. (See Appendix B for criteria.)	Y	Chapter 15
	iii. <u>Generalized Cladding Melting</u>. Generalized (i.e., nonlocal) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in item 1.C.i above are more stringent than melting criteria. Therefore, additional specific criteria are not used. However, this may not always be the case for newer alloys or reactor types.	Y	Chapter 15
	iv. <u>Fuel Rod Ballooning</u>. To meet the requirements of 10 CFR 50.46 as it relates to ECCS performance during accidents, the analysis of the core flow distribution must account for burst strain and flow blockage caused by ballooning (swelling) of the cladding. **RG 1.157** describes acceptable models, correlations, data, and methods that can be used to meet the requirements for a realistic calculation of ECCS performance during a LOCA. Alternatively, **Appendix K to 10 CFR Part 50** outlines the acceptable features of a conservative evaluation model to consider burst strain and flow blockage. Burst strain and flow blockage models must be	Y	Chapter 15

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	based on applicable data to (1) properly estimate the temperature and differential pressure at which the cladding will rupture (see item 1.B.vii above), (2) avoid underestimating the resultant degree of cladding swelling, and (3) avoid underestimating the associated reduction in assembly flow area. The flow blockage model evaluation is provided to the organization responsible for the review of transient and accident analyses for incorporation in the comprehensive ECCS evaluation model to demonstrate that the criteria in **10 CFR 50.46(b)** are not exceeded. The reviewer also determines whether the analysis of AOOs and other accidents should include fuel rod ballooning. The possibility of ballooning during an AOO transient or accident increases as the fuel rod pressure exceeds the system pressure. Those non-LOCA accidents that result in clad ballooning should examine the possibility of DNB propagation resulting from ballooning. The impact of ballooning on non-LOCA accidents should not be underestimated. A limit on ballooning (circumferential strain) may be required to prevent DNB propagation for these accidents.		
	v. <u>Structural Deformation</u>. Appendix A discusses the applicable analytical procedures.	Y	4.2.1.4.1 4.2.1.5.3 4.2.1.5.5 4.2.1.5.6 4.2.1.6 4.2.3.5.1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
4.2-SAC-02	Description and Design Drawings The reviewer determines that the fuel system description and design drawings provide an accurate representation and supply the information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description: • Type and metallurgical state of the cladding • Cladding outside diameter • Cladding inside diameter • Cladding inside roughness • Pellet outside diameter • Pellet roughness • Pellet density • Pellet resintering data • Pellet length • Pellet dish dimensions • Pellet grain size and open porosity • Burnable poison content • Insulator pellet parameters • Fuel column length • Overall rod length • Rod internal void volume • Fill gas type and pressure • Sorbed gas composition and content	Y	4.2.2

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	• Spring and plug dimensions • Fissile enrichment • Equivalent hydraulic diameter • Coolant pressure • Design-specific burnup limit • Control blade/rod descriptions, dimensions, and lifetime limits • Fit of control blade/rod interference with surrounding structure (e.g., channel box or guide tube) The following design drawings and dimensions are also necessary for an acceptable fuel system description: • Fuel assembly cross section • Fuel assembly outline • Fuel rod schematic • Spacer grid cross section • Guide tube and nozzle joint • Guide tube with respect to control rod dimensions • Control blade/rod assembly cross section • Control rod assembly outline • Control rod schematic • Burnable poison rod assembly cross section • Burnable poison rod assembly outline • Burnable poison rod schematic • Orifice and source assembly outline		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
4.2-SAC-03	<u>Design Evaluation</u> The reviewer will evaluate the methods for demonstrating that the design bases are met. Methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the safety analysis report by reference.	Y	4.2.3
	A. <u>Operating Experience</u> Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed before gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.	Y	4.2.4.1 4.2.4.1.4 4.2.4.1.5 4.2.3.5.7
	B. <u>Prototype Testing</u> When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed, when practical, to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested before irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer: • Spacer grid structural tests • Control rod structural and performance tests	Y	4.2.3.5.7 4.2.4.2 4.2.4.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	• Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping) • Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, fuel rod fretting (should account for spacer spring relaxation), and assembly wear and life)		
	In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The maximum burnup or fluence experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup or fluence limit for the new design. The following phenomena have been tested in this manner in new designs and will serve as a guide to the reviewer: • Fuel and burnable poison rod growth • Fuel rod bowing • Fuel rod, spacer grid, and channel box oxidation and hydride levels • Fuel rod fretting • Fuel assembly growth • Fuel assembly bowing • Channel box wear and distortion • Fuel rod ridging (PCI) • Crud formation • Fuel rod integrity • Holddown spring relaxation • Spacer grid spring relaxation	N/A-OTHER (No LTA program)	N/A

CHAPTER 4 Reactor			
SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	• Guide tube wear characteristics In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished before operation of the design's full core. The inability to perform in-reactor testing may result from an incompatibility of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see Subsection II.4 below).		
	C. Analytical Predictions Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.	N/A-INFO	N/A
	i. Fuel Temperatures (Stored Energy). Fuel temperatures and stored energy during normal operation serve as input to ECCS performance calculations. Temperature calculations require complex computer codes that model many different phenomena. **RG 1.157** describes models, correlations, data, and methods to realistically calculate ECCS performance during a LOCA and to estimate the uncertainty in that calculation. Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of **Appendix K to 10 CFR Part 50**. Phenomenological models that should be reviewed include the following: • Radial power distribution	Y	4.2.3.1.3 4.2.3.2 4.2.3.3 4.2.3.3.1 4.4 Chapter 15

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	• Fuel and cladding temperature distribution • Burnup distribution in the fuel • Thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers • Densification of the fuel • Thermal expansion of the fuel and cladding • Fission gas production and release • Solid and gaseous fission product swelling • Fuel restructuring and relocation • Fuel and cladding dimensional changes • Fuel-to-cladding heat transfer coefficient • Thermal conductivity of the gas mixture • Thermal conductivity in the Knudsen domain • Fuel-to-cladding contact pressure • Heat capacity of the fuel and cladding • Growth and creep of the cladding • Rod internal gas pressure and composition • Sorption of helium and other fill gases • Cladding oxide and crud layer thickness • Cladding-to-coolant heat transfer coefficient • Cladding hydriding Because of the strong interaction between these models, overall code behavior should be checked against data (standard problems or benchmarks) and the NRC audit codes.		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	NUREG/CR-6534 (PNNL-11513) Vol. 2, December 1997, FRAPCON-3 NUREG/CR-6534 (PNNL-11513) Vol. 4, May 2005, Babcox & Wilcox Report **BAW-10087A**, Rev. 1, August 1977, CENPD-139-A, July 1974, Supplement 1 to Technical Report on General Electric Reactor Fuels, December 14, 1973, Technical Report on Exxon Nuclear PWR Fuels, February 27, 1975, and the Letter on Westinghouse Safety Evaluation of WCAP-8720, February 9, 1979, provide examples of previous fuel performance code reviews.		
	ii. <u>Densification Effects</u>. In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking - see **SRP Section 4.3**), (2) the fuel linear heat generation rate (LHGR) - see **SRP Section 4.4**, and (3) the potential for cladding collapse. **NUREG-0085** and **RG 1.126** discuss densification magnitudes for power spike and LHGR analyses. To be acceptable, densification models should follow the guidelines of RG 1.126. Models for cladding collapse times should also be reviewed. The memorandums on Evaluation of Westinghouse Report, WCAP-8377, January 14, 1975 and on CEPAN-Method of Analyzing Creep Collapse of Oval Cladding, February 5, 1976, provide previous review examples.	Y	4.2.3.1.11 4.2.3.3.1 4.4
	iii. <u>Fuel Rod Bowing</u>. The memorandum on Request for Revised Rod Bowing Topical Reports, May 30, 1978, includes guidance for the analysis of fuel rod bowing. The memorandum on Revised Interim Safety	Y	4.2.3.5.6 4.4

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, February 16, 1977, presents interim methods that may be used. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing have not been approved.		
	iv. Cladding Collapse. Approved analytical models/methods are used to demonstrate that cladding collapse is not possible within the fuel lifetime. A change in cladding or fuel material (additives or significant changes in fabrication) and/or a reduction in the as-fabricated fuel cladding gap can impact the approved analytical model/methods used for this analysis. A change in fuel material can impact fuel densification and a change in cladding material can impact cladding creep, both of which can impact cladding collapse. If any of these parameters change, they must be evaluated in terms of their impact on the approved analytical models and methods for evaluating cladding collapse.	Y	4.2.3.1.11 4.2.3.6.3
	v. Structural Deformation. Appendix A discusses the acceptance criteria.	Y	4.2.3.4 4.2.3.5 4.2.3.5.2
	vi. Rupture and Flow Blockage (Ballooning). The ECCS evaluation model includes Zircaloy rupture and flow blockage models, which should be reviewed by the organization responsible for reactor systems. The models are empirical and should be compared with relevant data. **NUREG-0630**, **NUREG/CR-1883**, and the publication on Burst Criterion of	Y	Chapter 15

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SRP Criterion	**Description** **(AC = Acceptance Criteria Requirement,** **SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Zircaloy Fuel Cladding in a LOCA, August 4-7, 1980, provide examples of such data and previous reviews. These models should account for the phase transformation in the cladding at high temperatures.		
	vii. Fuel Rod Pressure. The thermal performance code for calculating temperatures discussed in item 3.C.i above should be used to calculate fuel rod pressures in conformance with the fuel damage criteria of item 1.A.vi in Subsection II. This calculation should account for uncertainties in the estimated rod powers, code models, and fuel rod fabrication. The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.	Y (Per AREVA Topical Report BAW-10183P-A)	4.2.3.3.1 4.4 Chapter 15
	viii. Metal/Water Reaction Rate. To meet the requirements of **10 CFR 50.46(b)** as it relates to the performance of the ECCS during accidents, the rate of energy release, hydrogen generation, and cladding oxidation resulting from the reaction of the Zircaloy cladding with steam should be calculated. Currently this can be calculated in two ways. **RG 1.157** allows the use of a best-estimate model, provided its technical basis is demonstrated with appropriate data and analyses. Alternatively, **Appendix K to 10 CFR Part 50** specifies that the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation **(Argonne National Laboratory Report ANL-6548**, May 1962). For non-LOCA	Y	Chapter 15

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	applications, other correlations may be used if justified. These reaction rate models were originally developed based upon unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the reaction rate characteristics of the fuel cladding material.		
	Rulemaking pursuant to **10 CFR 50.46** is planned to implement a **performance-based test program** to provide an acceptable means for establishing specific reaction rate models for new cladding materials. Future cladding alloys must comply with the new rules and the need to provide an empirical database to support applicable reaction rate models.	EXEMPTION (10 CFR 50.46 regarding the use of M5^{TM} per AREVA Topical Report BAW-10227P-A)	4.2
	ix. **Fission Product Inventory**. The assumptions in RG 1.3, RG 1.4, RG 1.5, RG 1.25, RG 1.77, RG 1.195, and RG 1.196, as they relate to fission product release for existing reactors (i.e., DC applications before January 10, 1997), currently specify the available radioactive fission product inventory in fuel rods (i.e., the gap inventory). **RG 1.195** and **RG 1.196** can be used in place of RG 1.3, RG 1.4, RG 1.5, RG 1.25, and RG 1.77. **RG 1.183** and the requirements of **10 CFR 50.34** apply to fission product release for new reactors. An alternate source term (AST), specified in **10 CFR 50.67**, can be applied to existing reactors as an alternative to **10 CFR Part 100** as defined in these documents. **American Nuclear Society (ANS) 5.4** presents an approved method for release during non-LOCAs and situations that do not involve accidents in which the fuel temperature exceeds the temperature experienced during	Y	4.4 15.0.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	normal operation and AOOs. ANS 5.4 also provides an acceptable analytical model for calculating the release of volatile fission products from oxide fuel pellets during normal steady-state conditions.		
	When used with nuclide yields, this model will define the inventory of **volatile fission products** that could be available for release from the fuel rod if the cladding were breached, sometimes referred to as gap inventory. Recent experimental data from RIA tests in Nuclear Safety Research Reactor (NSRR) and Cabri (Publication on NSRR/RIA Experiments with High Burnup PWR Duels, March 2-6, 1997, Publication on High-Burnup BWR Fuel Behavior Under Simulated Reactivity-Initiated Accident Conditions, Nuclear Technology Vol. 38, June 2002, and Publication on The Role of Grain Boundary Fission Gases in High Burn-Up Fuel Under Reactivity Initiated Accident Conditions, September 2000) suggest that the gap inventory for a BWR rod drop accident specified in RG 1.183 and for a PWR control rod ejection accident may need modification. The NRC has plans to issue new guidelines for gap inventory (fission product release) from these accidents.	Y	4.4 15.0.3
4.2-SAC-04	Testing, Inspection, and Surveillance Plans Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.		
	A. Testing and Inspection of New Fuel Testing and inspection plans for new fuel should verify cladding integrity, fuel system dimensions, fuel enrichment, burnable poison	Y	4.2.4.4

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	concentration, and absorber composition. Quality control reports should document the details of the manufacturer's testing and inspection programs and should be referenced and summarized in the safety analysis report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. When the overall testing and inspection programs are essentially the same as those for previously approved plants, a statement to that effect should be made. In that case, the safety analysis report need not include program details, but an appropriate reference should be cited and a summary (tabular) should be presented.		
	B. Online Fuel System Monitoring The applicant's online fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. NUREG-0401 and NUREG/CR-1380 evaluate several common detection methods and should be used in this review. Surveillance is also needed to assure that B4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as those described in NUREG-0308 are acceptable.	Y	4.2.4.5 9.3.2
	C. Postirradiation Surveillance A postirradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered (i.e., whether the proposed fuel design	Y	4.2.4.6

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SRP Criterion	**Description** **(AC = Acceptance Criteria Requirement,** **SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	is the same as current operating fuel or incorporates new design features). For a fuel design similar to that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. The program should also commit to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel. In addition to the plant-specific surveillance program, a continuing fuel surveillance effort should exist for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program. For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with the prototype testing discussed in Subsection II.3.B. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.		

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 4.3	**Nuclear Design (R3, 03/2007)**		
4.3-AC-01	**GDC 10** requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.	Y	4.3.1.1 4.3.1.6
4.3-AC-02	**GDC 11** requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.	Y	4.3.1.2
4.3-AC-03	**GDC 12** requires that power oscillations that could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.	Y	4.3.1.9 4.3.2.7
4.3-AC-04	**GDC 13** requires provision of instrumentation and controls (I&C) to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges.	Y	4.3.2.2.6 4.3.2.2.9
4.3-AC-05	**GDC 20** requires automatic initiation of the reactivity control systems (RCSs) to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety occurs under accident conditions. There are usually primary and secondary independent RCSs.	Y	4.3.1 4.3.1.8 4.3.2.2.6
4.3-AC-06	**GDC 25** requires that no single malfunction of the RCSs (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.	Y	4.3.1.7

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
4.3-AC-07	**GDC 26** requires that two independent RCSs of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.	Y	4.3.1.8
4.3-AC-08	**GDC 27** requires that the RCSs have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.	Y	4.3.1.9
4.3-AC-09	**GDC 28** requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.	Y	4.3.1.7
4.3-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
4.3-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the	N/A-COL	N/A

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
4.3-SAC-01	There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady-state or load-following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7), and transient and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204°C (2200°F) peak cladding temperature allowed for loss-of-coolant accidents (LOCAs), are not exceeded during design-basis accidents. Consideration must also be made to the effect of coolant temperatures and enthalpy on the fuel and cladding temperatures. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed. The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that:	Y	4.3.1.6 4.3.2.2.9
	A. A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into	Y	4.3.1.1 4.3.1.6

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.		4.3.2.1 4.3.2.2 4.3.2.7 4.3.3
	B. A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).	Y	4.3.1.1 4.3.1.6 4.3.2.2.7 4.3.2.2.8 4.3.2.2.9 4.3.3
	Criteria for acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram frequency and extent of power distribution measurements, and use of ex-core and in-core instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review. The organization responsible for the review/assessment of nuclear design has enunciated Branch Technical Position 4-1 for	N/A-VEN	N/A

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Westinghouse reactors that employ constant axial offset control.		
	Acceptance criteria for power spike models can be found in a NUREG report on fuel densification, and are discussed in Regulatory Guide (RG) 1.126. Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.	Y	4.3.2.2.5
4.3-SAC-02	The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states “...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity,” and is considered to be satisfied in light water reactors (LWRs) by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient (MTC) such as may exist in PWRs at beginning of core life. The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgement to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of	Y	4.3.1.2 4.3.2.3 4.3.2.4 4.3.3

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.		
4.3-SAC-03	Acceptance criteria relative to control rod patterns and reactivity worths include:		
	A. The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may occur in the reactor. These values are used in the transient and accident analyses and judgment as to the adequacy of the uncertainty allowances are made in the review of the transient and accident analyses.	Y	4.3.2.4.12 4.3.2.5
	B. Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a position of the organization responsible for the review/assessment of nuclear design to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.	Y	4.3.1.9 4.3.2.2.6 4.3.2.2.7 4.3.2.2.9
4.3-SAC-04	There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and database should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and encompass a sufficient range of variables and operating conditions.	Y	4.3.3

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 4.4	**Thermal and Hydraulic Design (R2, 03/2007)**		
4.4-AC-01	**General Design Criterion (GDC) 10**, as it relates to whether the design of the reactor core includes appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation or AOOs.	Y (Per AREVA Topical Reports BAW-10231P-A & ANP-10263	4.4.1.2.1
4.4-AC-02	**GDC 12**, as it relates to whether the design of the reactor core and associated coolant, control, and protection systems assures that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.	Y (Per AREVA Topical Report ANP-10287)	4.4.6
4.4-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
4.4-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
4.4-SAC-01	**SRP Section 4.2** specifies the acceptance criteria for the evaluation of fuel design limits. One criterion provides assurance that there be at least a 95-percent probability at the 95-percent confidence level that the hot fuel rod in the core does not experience a DNB or transition condition during normal operation or AOOs.	Y (Per AREVA Topical Reports ANP-10269 & BAW-10199)	4.4.4.3
	Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least a 95-percent probability at the 95-percent confidence level. The assessment of thermal margin should also consider the uncertainties in instrumentation. The origin of each uncertainty parameter, such as fabrication uncertainty, computational uncertainty, or measurement uncertainty e.g., reactor power, coolant temperature, flow), should be identified. Each uncertainty parameter should be identified as statistical or deterministic and should clearly describe the methodologies used to combine uncertainties.	Y (Per AREVA Topical Reports ANP-10269 & BAW-10199)	4.4.2.9.2 4.4.2.9.3 4.4.2.9.4 4.4.2.9.6
	Core design and operating changes for extended power uprates (EPUs) should be performed in a manner that ensures adequate safety margin. At a minimum, there should be a 95-percent probability at the 95-percent confidence level that a hot fuel rod in the reactor core will not experience a DNB or a transition condition during normal operation or AOOs. Specifically, this safety criterion should be satisfied while accounting for changes in radial and bundle power distribution, including any changes in critical heat flux ratio (CHFR) and CPR. The reviewer should confirm the adequacy of the flow-based average power range monitor flux trip and	Y	4.4.4 4.4.5 4.4.6

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	safety limit minimum critical power ratio at the uprated conditions (Review Standard RS-001). The reviewer should also ensure that the correlations used in the EPU analysis do not exceed their validation range under uprated normal operation and AOO conditions.		
	The following are two examples of acceptable approaches to meeting this criterion:		
	A. For departure from nucleate boiling ratio (DNBR), CHFR or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs.	Y (Per AREVA Topical Reports ANP-10269 & BAW-10199)	4.4.4.1
	B. The limiting (minimum) value of DNBR, CHFR, or CPR correlations is to be established such that at least 99.9 percent of the fuel rods in the core will not experience a DNB or boiling transition during normal operation or AOOs. Correlations of critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculational techniques involving coolant mixing and the effect of axial power distributions.	Y	4.4.1.1
4.4-SAC-02	Problems affecting DNBR or CPR limits, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty which is determined experimentally or analytically. Subchannel hydraulic analysis codes, such as those described in “TEMPThermal Enthalpy Mixing Program,” BAW-10021, Babcock and Wilcox Company, April 1970 and “THINC-IC-An Improved Program for Thermal-Hydraulic Analysis Of Rod	Y (Per AREVA Topical Reports BAW-10147, BAW-10156, & BAW-10183)	4.4.4.1.5 4.4.4.5.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Bundle Cores," WCAP-7956, Westinghouse Electric Corporation, June 1973, should be used to calculate local fluid conditions within fuel assemblies for use in PWR DNB correlations. The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores. The review should include the effects of radial pressure gradients in the core flow distribution. The reviewer should also confirm that calculations of BWR fluid conditions for use in CHF correlations have been made in accordance with the models specified in "Loss of Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors," NEDO-10329, Appendix C, General Electric Company, April 1971 and "General Electric Company Analytical Model for Loss of Coolant Accident Analysis in Accordance with 10 CFR Part 50, Appendix K," NEDO-20566, General Electric Company, November 1975.		
4.4-SAC-03	The design should address core oscillations and thermal-hydraulic instabilities as described in **SRP Section 15.9**.	N/A-BWR	N/A
4.4-SAC-04	Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations. For components of unusual geometry, such as those listed below, these relationships should be confirmed empirically using representative databases from approved reports: A. Reactor vessel ("Reactor Vessel Model Flow Tests," BAW-10037 (nonproprietary version of BAW-10012), Rev. 2, Babcock and Wilcox Company, September 1968).	Y	4.4.2.4
			4.4.2.7.1
		Y (Per AREVA Topical Report BAW-10156-A)	4.4.2.7.2
			4.4.2.7.3
	B. Jet pump ("Design and Performance of General Electric Boilng Water	N/A-BWR	N/A

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Reactor Jet Pumps," APED-5460, General Electric Company, September 1968).		
	C. Core flow distribution (BAW-10037 and "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299, General Electric Company, January 1971, DRAFT Rev. 2, April 1996).	N/A-BWR	N/A
	D. Void fraction distribution for BWRs.	N/A-BWR	N/A
4.4-SAC-05	The proposed technical specifications should ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters. The safety limits and limiting safety settings must be established for each parameter, or combinations of parameters, to satisfy specific acceptance criterion 1, above.	Y	4.4.6.4 4.4.6.5
4.4-SAC-06	Preoperational and initial startup test programs should follow the recommendations of **Regulatory Guide 1.68**, as it relates to measurements and the confirmation of thermal-hydraulic design aspects.	Y	4.4.5.1 4.4.5.2
4.4-SAC-07	The design description and proposed procedures for use of the loose parts monitoring system should be consistent with the requirements of **Regulatory Guide 1.133**.	Y	4.4.6.6
4.4-SAC-08	The thermal-hydraulic design should account for the effects of crud in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the capability to detect a 3-percent drop in the reactor coolant flow. The flow should be monitored every 24 hours.	EXCEPTION (T-H methodology and CHF analyses addressed AREVA Topical Reports BAW-10220P & BAW-10178P)	4.4.4.1 4.4.4.5

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
4.4-SAC-09	Instrumentation provided for an unambiguous indication of ICC, such as primary coolant saturation meters in PWRs, reactor vessel measurement systems, and core exit thermocouples, should meet the design requirements of TMI Action Plan Item II.F.2 of **NUREG-0737**. Applicants subject to 10 CFR 50.34(f) should meet the requirements of **10 CFR 50.34(f)(2)(xviii)**. Procedures for detection and recovery from conditions of ICC must be consistent with technical guidelines, including applicable EPGs developed pursuant to the TMI action plan, that incorporate response predictions based on appropriate analyses.	Y (Per AREVA Topical Report ANP-10287)	4.4.6.1 4.4.6.2
4.4-SAC-10	Thermal-hydraulic stability performance of the core during an ATWS event should not exceed acceptable fuel design limits. **SRP Sections 15.8** and 15.9 describe an acceptable method for performing such an analysis for BWR and PWR cores.	Y	4.4.4.5.4 15.8
SRP 4.5.1	**Control Rod Drive Structural Materials (R3, 03/2007)**		
4.5.1-AC-01	**GDC 1**, as it relates to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.	Y	4.5.1
4.5.1-AC-02	**GDC 14**, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.	Y	4.5.1
4.5.1-AC-03	**GDC 26**, as it relates to control rods being capable of reliable control of reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin of malfunctions, specified acceptable fuel design limits are not exceeded.	Y	4.5.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
4.5.1-AC-04	**10 CFR 50.55a**, as it relates to SSCs being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	Y	4.5.1
4.5.1-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
4.5.1-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
4.5.1-SAC-01	Materials Specifications. The properties of the materials selected for the CRDM should be equivalent to those of **Section III, Appendix I, Division 1 of the ASME Code** or **Section II, Parts A, B, C, and D of the ASME Code**. Cold-worked austenitic stainless steels should have a 0.2 percent offset yield strength no greater than 620 MPa (90,000 psi), to reduce the probability of stress corrosion cracking in these systems. **Regulatory Guide (RG)**	Y	4.5.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	1.85 describes the acceptable code cases that may be used with these specifications.		
4.5.1-SAC-02	Austenitic Stainless Steel Components. Acceptance criteria include criteria described in **SRP Section 5.2.3**, Subsections II.4.D and E, and the criteria described below. **RG 1.44** describes accepted methods for preventing intergranular corrosion of stainless steel components. Furnace-sensitized material should not be allowed, and methods described in this guide should be followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding. The controls for abrasive work on austenitic stainless steel surfaces should be adequate for preventing contamination that promotes stress corrosion cracking. The final surfaces should meet the acceptance standards specified in **ASME NQA-1-1994** Edition, “Quality Assurance Requirements for Nuclear Facilities.” Tools that contain materials that could contribute to stress-corrosion cracking or that, from previous usage, may be contaminated with such materials should not be used on austenitic stainless steel surfaces.	Y	4.5.1
4.5.1-SAC-03	Other Materials. All materials for use in this system should be selected for their compatibility with the reactor coolant as described in **Articles NB-2160 and NB-3120 of the ASME Code**. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardening stainless steels should be specified for assurance that these materials will not deteriorate from stress corrosion cracking in service.	Y	4.5.1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Acceptable heat treatment temperatures include aging at 565E - 595°C (1050° - 1100°F) for Type 17-4 PH and 565°C (1050°F) for Type 410 stainless steel.		
4.5.1-SAC-04	Cleaning and Cleanliness Control. Onsite cleaning and cleanliness control should be in accordance with **ASME NQA-1-1994** edition. The oxygen content of the water in vented tanks need not be controlled. Vented tanks with deionized or demineralized water are normal sources of water for final cleaning or flushing of finished surfaces. Halogenated hydrocarbon cleaning agents should not be used.	Y	4.5.1
SRP 4.5.2	**Reactor Internal and Core Support Structure Materials (R3, 03/2007)**		
4.5.2-AC-01	**10 CFR 50.55a**, "Codes and Standards," which requires that SSCs shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	Y	4.5.1
4.5.2-AC-02	**10 CFR Part 50, Appendix A, GDC 1**, "Quality Standards and Records," which requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. **GDC 1** also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	Y	4.5.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
4.5.2-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
4.5.2-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
4.5.2-SAC-01	Materials. For core support structures and reactor internals, the permitted material specifications are those given in the **ASME Code, Section III, Division 1, Sub-subarticle NG-2120**. The properties of these materials are specified in **Tables 2A, 2B and 4 of Section II of the Code**. Additional permitted materials and their applications are identified in ASME Code Cases approved for use as described in **Regulatory Guide 1.84**, “Design, Fabrication, and Material Code Case Acceptability, ASME, Section III.”	Y	4.5.1

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SRP Criterion	Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
4.5.2-SAC-02	Controls on Welding. Methods and controls for core support structures and reactor internals welds shall be in accordance with **ASME Code, Section III, Division 1, Article NG-4000**. The examination requirements and acceptance criteria for these welds are specified in **Article NG-5000**.	Y	4.5.1
4.5.2-SAC-03	Nondestructive Examination. Nondestructive examinations shall be in accordance with the requirements of **ASME Code, Section III, Division 1, Subarticle NG-2500**. The nondestructive examination acceptance criteria shall be in accordance with the requirements of **ASME Code, Section III, Division 1, Subarticle NG-5300**.	Y	4.5.1
4.5.2-SAC-04	Austenitic Stainless Steels. The acceptance criteria for this area of review are given in **SRP Section 5.2.3**, subsections II.2 and II.4.a, b, d, and e. **Regulatory Guide 1.44** provides acceptance criteria for preventing intergranular corrosion of stainless steel components. In conformance with this guide, furnace sensitized material should not be allowed. Methods described in this guide should be followed for cleaning and protecting austenitic stainless steel from contamination during handling, storage, testing, and fabrication, and for determining the degree of sensitization that occurs during welding.	Y	4.5.1
4.5.2-SAC-05	Other Materials. All materials used for reactor internals and core support structures must be selected for compatibility with the reactor coolant, as specified in **Subsubarticles NG-2160 and NG-3120 of Section III, Division 1 of the**	Y	4.5.1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	ASME Code. The tempering temperature of martensitic stainless steels and the aging temperature of precipitation-hardened stainless steels should be specified to provide assurance that these materials will not deteriorate in service. Acceptable heat treatment temperatures are 565°C - 595°C (1050°F - 1100°F) for aging of Type 17-4 PH and 565°C (1050°F) for tempering of Type 410 stainless steel. Other materials shall have similar appropriate heat treat and fabrication controls in accordance with strength and compatibility requirements.		
SRP 4.6	**Functional Design of Control Rod Drive System (R2, 03/2007)**		
4.6-AC-01	**GDC 4** found in Appendix A to 10 CFR Part 50, as it relates to the structures, systems, and components important to safety that shall be designed to accommodate the effects of and to compatible with the environmental conditions during normal plant operation as well as during postulated accidents.	Y	4.6
4.6-AC-02	**GDC 23**, as it relates to the protection system failure modes such that the system shall fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of system, loss of energy, or postulated adverse environment are experienced.	Y	3.9.4 4.6.2
4.6-AC-03	**GDC 25**, as it relates to the fuel design such that the specified limits are not exceeded for any single malfunction of the reactivity control system.	Y	4.6.5 7.1
4.6-AC-04	**GDC 26**, as it relates to the reactivity control system redundancy and capability such that two independent reactivity control systems of different design principles shall be provided and capable of reliably controlling reactivity changes under conditions of normal operation,	Y	4.6

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	including anticipated operational occurrences to assure acceptable fuel design limits are not exceeded. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.		
4.6-AC-05	**GDC 27**, as it relates to the combined reactivity control systems capability such that the reactivity control system design shall have a combined capability, in conjunction with poison addition by the emergency core cooling system to reliably control reactivity changes to assure that under postulated accident conditions the capability to cool the core is maintained.	Y	4.6
4.6-AC-06	**GDC 28**, as it relates to reactivity limits such that reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant boundary nor disturb the core and its supports structures to impair significant capability to cool the core.	Y	4.6.5 7.2
4.6-AC-07	**GDC 29**, as it relates to protecting system against anticipated operational occurrences such that the design of the protection and reactor control systems should assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.	Y	4.6
4.6-AC-08	**10 CFR 50.62(c)(3)**, as it relates to those requirements that impact the CRDS functional design. Specifically for BWRs, the alternate rod injection system must be diverse and independent (from the reactor trip system) and must have redundant scram air header exhaust valves.	N/A-BWR	N/A
4.6-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC)	ITAAC	Tier 1

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
4.6-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
4.6-SAC-01	To meet the requirements of **GDC 4**, the CRDS should remain functional and provide reactor shutdown capabilities under adverse environmental conditions and after postulated accidents.	Y	4.6
4.6-SAC-02	To meet the requirements of **GDC 23**, the CRDS should fail in an acceptable condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure.	Y	3.9.4 4.6.2
4.6-SAC-03	To meet the requirements of **GDC 25**, the design of the reactivity control systems should assure that a single malfunction of the CRDS will not result in exceeding acceptable fuel design limits.	Y	4.6.5 7.1
4.6-SAC-04	To meet the requirements of **GDC 26**, the CRDS should be capable of providing sufficient operational control and reliability during reactivity	Y	4.6

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SRP Criterion	**Description (AC = Acceptance Criteria Requirement, SAC = Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	changes during normal operation and anticipated operational occurrences.		
4.6-SAC-05	To meet the requirements of **GDC 27**, the combined capability of CRDS and emergency core cooling system should reliably control the reactivity changes to assure the capability to cool the core under accident conditions.	Y	4.6
4.6-SAC-06	To meet the requirements of **GDC 28**, the CRDS should be designed to assure that reactivity accidents do not result in damage to the reactor coolant pressure boundary, or result in sufficient damage to the core or support structures so as to significantly impair coolability.	Y	4.6.5 7.2
4.6-SAC-07	The CRDS should be designed to ensure an extremely high probability of functioning during anticipated operational occurrences to in conformance is **GDC, 29**.	Y	4.6
4.6-SAC-08	To meet the requirements of **10 CFR 50.62(c)(3)**, BWR plants should have an alternate rod injection system that is diverse and independent from the reactor trip system and should have redundant scram air header exhaust valves.	N/A-BWR	N/A
BTP 4-1	**Westinghouse Constant Axial Offset Control (CAOC) (R3, 03/2007)**	N/A-VEN	N/A

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Reactor Coolant System and Connected Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 5.2.1.1	**Compliance with the Codes and Standards Rule, 10 CFR 50.55a (R3, 03/2007)**		
5.2.1.1-AC-01	**10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1** as to the requirement that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function performed.	Y	5.2.1
5.2.1.1-AC-02	**10 CFR 50.55a** as to the establishment of minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of RCPB components and other safety-related fluid systems of boiling- and pressurized-water reactor nuclear power plants by compliance with appropriate editions of published industry codes and standards.	Y	5.2.1
5.2.1.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.2.1.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 5.2.1.2	**Applicable Code Cases (R3, 03/2007)**		
5.2.1.2-AC-01	**10 CFR Part 50, Appendix A, General Design Criterion 1**, as it relates to the requirement that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.	Refer to SRP 5.2.1.1	5.2.1
5.2.1.2-AC-02	**10 CFR 50.55a**, as it relates to the rule that establishes minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain components of boiling and pressurized water reactor nuclear power plants by requiring conformance with appropriate editions of specified published industry codes and standards.	Refer to SRP 5.2.1.1	5.2.1
5.2.1.2-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.2.1.2-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
5.2.1.2-SAC-01	To meet the requirements of **General Design Criterion 1** and **10 CFR 50.55a**, the following regulatory guides are used: 1. **Regulatory Guide 1.84**, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1." This guide lists those Section III, Division 1, ASME Code Cases oriented to design, fabrication, materials, and testing, which are acceptable to the staff for implementation in the licensing of nuclear power plants. 2. **Regulatory Guide 1.147**, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." This guide lists those Section XI ASME Code Cases which are acceptable to the staff for use in the inservice inspection of components and their supports, as described in the first paragraph of subsection I, of this SRP. 3. **Regulatory Guide 1.192**, "Operation and Maintenance Code Case Acceptability, ASME OM Code." This guide lists ASME OM Code Cases oriented to operation and maintenance for nuclear power plant components which are acceptable to the staff for implementation in the licensing of nuclear power plants.	Y	5.2.1
5.2.1.2-SAC-02	Code Cases pertaining to ASME Code Section III, Division 2, as well as Code Cases alternatives to **Regulatory Guides 1.84, 1.147, or 1.192**, or for those not covered in Regulatory Guides 1.84, 1.147, or 1.192 may be acceptable in either of the following cases: 1. If the proposed Code Cases provide an acceptable level of quality and safety; or 2. If compliance with the specified requirements of **10 CFR 50.55a** would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.	Y	5.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 5.2.2	**Overpressure Protection (R3, 03/2007)**		
5.2.2-AC-01	**General Design Criterion (GDC) 15**, as it relates to designing the RCS and associated auxiliary, control, and protection systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.	Y	5.2.2
5.2.2-AC-02	**GDC 31**, as it relates to designing the RCPB with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fractures is minimized.	Y	5.2.2
5.2.2-AC-03	**10 CFR 50.34(f)(2)(x)** and **10 CFR 50.34(f)(2)(xi)** require that RCS SRVs meet Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of **NUREG-0737**.	Y	5.2.2
5.2.2-AC-04	**10 CFR 52.47(a)(8)**provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in **10 CFR 50.34(f)**.	Y	5.2.2
5.2.2-AC-05	**10 CFR 52.79(a)(17)** provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the TMI-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).	Y	3.9.6 5.2.2 5.4.13 14.2
5.2.2-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	ITAAC	Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the Atomic Energy Act, and the NRC's regulations;		
5.2.2-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.2.2-SAC-01	<u>Material Specifications</u>. The requirements of **GDC 1**, **GDC 30**, and **10 CFR 50.55a** regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of material code cases, as described in **Regulatory Guide 1.84**. The specifications for permitted materials are identified in Appendix I to Section III of the ASME Code or described in detail in Parts A, B, and C of Section II of the ASME Code. Regulatory Guide 1.84 describes acceptable material code cases and guidelines for application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications.	Y	5.2.2
5.2.2-SAC-02	<u>Design Requirements for BWRs Operating at Power</u>	N/A-BWR	N/A
5.2.2-SAC-03	<u>Design Requirements for PWRs Operating at Power</u>		
	A. For overpressure protection during power operation of the PWR reactor, the design of the PORVs or the pressurizer should have	Y (U.S. EPR design	5.2.2

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Reactor Coolant System and Connected Systems

SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	sufficient capacity to preclude actuation of safety valves during normal operational transients, when assuming the following conditions at the plant: i. The reactor is operating at the licensed core thermal power level. ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure. iii. All components, instrumentation, and controls function normally.	does not utilize PORVs or associated block valves)	
	B. The designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by **ASME Code Article NB-7000**. Also, sufficient available margin should account for uncertainties in the design and operation of the plant assuming: i. The reactor is operating at a power level that will produce the most severe overpressurization transient. ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure. iii. The second safety-grade signal from the reactor protection system initiates the reactor scram. iv. The discharge flow is based on the rated capacities specified in ASME Code Article NB-7000 for each type of valve. In addition, the designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000.	Y	5.2.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power. Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of **ASME Code Article NB-7511.1**.	Y	5.2.2
5.2.2-SAC-04	Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown). The design of the low-temperature overpressure protection (LTOP) system or the cold overpressure mitigation system (COMS) should be in accordance with the requirements of **Branch Technical Position (BTP) 5-2**. The LTOP system or COMS should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP 5-2.	Y	5.2.2.9
5.2.2-SAC-05	Testing and Inspections. The performance of tests and inspections should occur before operation and during startup to functionally demonstrate that the overpressure protection system, as installed, meets all design requirements.	Y	5.2.2.10
5.2.2-SAC-06	Technical Specifications. The technical specifications should specify appropriate limiting conditions of operation and inservice surveillance to ensure continued system reliability, including, for PWRs, specific limiting conditions of operation and testing of the LTOP system as specified in **NUREG-1430 through NUREG-1434**, **Generic Letters No. 82-16, 83-02, and 90-06**.	Y	5.2.4.1.5 and Chapter 16

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
5.2.2-SAC-07	TMI Action Plan Requirements. Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief valve (RV) and safety valve (SV) testing. Section II.D.3 of the TMI Action Plan requires that RVs and SVs be provided with direct valve position indication. **Generic Letters No. 82-16 and 83-02** requires sections II.D.1 and II.D.3 be covered by technical specifications while **NUREG-0737** section II.K.3.3 specifies reporting for section II.D.1 and II.D.3.	Y	3.9.6
SRP 5.2.3	**Reactor Coolant Pressure Boundary Materials (R3, 03/2007)**		
5.2.3-AC-01	**General Design Criteria (GDC) 1 and 30** found in Appendix A to Part 50, as they relate to quality standards for design, fabrication, erection and testing;	Y	5.2.1.1 5.2.2.1
5.2.3-AC-02	**GDC 4**, as it relates to the compatibility of components with environmental conditions;	Y	5.2.3
5.2.3-AC-03	**GDC 14 and 31**, as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB;	Y	5.2.3
5.2.3-AC-04	**Appendix B to Part 50, Criterion XIII**, as it relates to onsite material cleaning control;	Y	5.2.3
5.2.3-AC-05	**Appendix G to Part 50**, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB;	Y	5.2.3
5.2.3-AC-06	**Section 50.55a**, as it relates to quality standards applicable to the RCPB;	Y	5.2.3
5.2.3-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if	ITAAC	Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
5.2.3-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.2.3-SAC-01	<u>Material Specifications</u>. The requirements of **GDC 1**, **GDC 30**, and **§ 50.55a** regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of materials Code Cases as described in **Regulatory Guide 1.84**, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” The specifications for permitted materials are those identified in the ASME Code, Section III, Appendix I, or described in detail in the ASME Code, Section II, “Materials, Parts A, B, and C. Regulatory Guide 1.84 describes acceptable materials Code Cases and guidelines for their application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications. Staff positions related to BWR piping materials and materials processing are described in Attachment A to Generic Letter 88-01. The technical	Y	5.2.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313.		
5.2.3-SAC-02	Compatibility of Materials with the Reactor Coolant. The requirements of **GDC 4** relative to compatibility of components with environmental conditions are met by compliance with the applicable provisions of the **ASME Code** and by compliance with the positions of **Regulatory Guide 1.44**, “Control of the Use of Sensitized Stainless Steel.” Ferritic low alloy steels and carbon steels, which are used in many principal pressure-retaining components, are clad with a layer of austenitic stainless steel. If cladding is not used, conservative corrosion allowances must be indicated for all exposed surfaces of carbon and low alloy steels, as indicated in the **ASME Code, Section III, NB-3121**, “Corrosion.” Regulatory Guide 1.44 contains staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB. Positions related to BWR piping materials, including verification of nonsensitization of the material by an approved test, are described in Attachment A to Generic Letter 88-01. The technical bases for the positions provided in Generic Letter 88-01 and similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are detailed in NUREG-0313, Revision 2.	Y	5.2.3
5.2.3-SAC-03	Fabrication and Processing of Ferritic Materials		
	A. The acceptance criteria for fracture toughness are the requirements of **Appendix G**, “Fracture Toughness Requirements,” **of 10 CFR Part**	Y	5.2.3.3

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	U.S. EPR Assessment	FSAR Section(s)
	50. These criteria satisfy the requirements of **GDC 14 and GDC 31** regarding prevention of fracture of the RCPB. Appendix G requires that the pressure-retaining components of the RCPB that are made of ferritic materials shall meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. With respect to absorbed energy in J (ft-lbs) and lateral expansion as shown by Charpy V-notch (C_v) impact tests, all materials shall meet the acceptance standards of **Article NB-2300 of the Code, Section III**, and the requirements of **Sections IV of Appendix G, 10 CFR Part 50**, as follows: (1) Materials for piping (i.e., pipes, tubes, and fittings), pumps, and valves, excluding bolting materials, shall meet the requirements of the **Code, Section III, Paragraph NB-2331 or NB-2332** (as applicable based upon thickness), and **Appendix G, Paragraph G-3100 to the Code, Section III**. The required C_v values for piping, pumps, and valves are specified in **Table NB-2332(a)-1 of the Code, Section III**. (2) Materials for bolting for which impact tests are required shall meet the requirements of the **Code, Section III, Paragraph NB-2333**. (3) Calibration of instruments and equipment shall meet the requirements of the **Code, Section III, Paragraph NB-2360**. The special acceptance requirements and staff positions for fracture toughness of reactor vessels are covered by **SRP Section 5.3.1**.		
	B. The acceptance criteria for control of ferritic steel welding are based upon the following regulatory guides and ASME Code provisions to	Y	5.2.3.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	satisfy the quality standards requirements of **GDC 1**, **GDC 30**, and **§ 50.55a**: (1) The amount of specified preheat must be in accordance with the requirements of the **Code, Section III, Appendix D, Paragraph D-1210**. These requirements are supplemented by positions described in **Regulatory Guide 1.50**, “Control of Preheat Temperature for Welding of Low Alloy Steel.” The supplemental acceptance criteria for control of preheat temperature are as follows: (a) According to the welding procedure qualification minimum preheat and maximum interpass temperatures should be specified and the welding procedure should be qualified at the minimum preheat temperature. For production welds, the preheat temperature should be maintained until a post-weld heat treatment has been performed. (b) Production welding should be monitored to verify that the limits on preheat and interpass temperatures are maintained. In the event that the above criteria are not met, the weld is subject to rejection. (2) The acceptance criteria for electroslag welds are presented in **Regulatory Guide 1.34**, “Control of Electroslag Weld Properties.” These criteria specify acceptable solidification patterns and impact test limits (for qualification of welds in Class 1 and Class 2 components) and the criteria for verifying conformance during production welding. (3) **Regulatory Guide 1.71**, “Welder Qualification for Areas of Limited Accessibility,” provides the following criteria for		

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	requalification of welders: the performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint; and requalification is required for different restricted accessibility conditions or when any of the essential variables listed in the Code, Section IX, "Welding and Brazing Qualifications" are changed. Qualification of the welder or welding operators for limited accessibility may be waived provided that 100% radiographic and/or ultrasonic examination of the completed welded joint is performed. Examination procedures and acceptance standards should meet the requirements of the **ASME Section III of the Code**. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance Documentation for the completed weld. (4) **Regulatory Guide 1.43**, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," provides criteria to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel. According to these criteria, material known to have susceptibility to underclad cracking should not be weld clad by high-heat-input welding processes and should be qualified for use to demonstrate that underclad cracking is not induced.		
	C. For nondestructive examination of ferritic steel tubular products, the requirements of **GDC 1**, **GDC 30**, and **§ 50.55a** regarding quality standards are met by compliance with the applicable provisions of the ASME Code. The acceptance criteria are given in **Section III of the**	Y	5.2.3.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Code, Paragraphs NB-2550 through NB-2570.		
5.2.3-SAC-04	<u>Fabrication and Processing of Austenitic Stainless Steel</u>		
	A. The requirements of **GDC 4** relative to compatibility of components with environmental conditions are met with measures to avoid sensitization in austenitic stainless steels. The acceptance criteria for testing, alloy compositions, and heat treatment, to avoid sensitization in austenitic stainless steels, are covered in **Regulatory Guide 1.44** and additional criteria for BWRs are specified in Attachment A to Generic Letter 88-01 based upon the technical information provided in NUREG-0313, Revision 2. Similar recommendations related to minimizing stress corrosion cracking in susceptible piping of BWRs are described in NUREG-0313, Revision 2. Regulatory Guide 1.44 also identifies acceptable methods for verification of non-sensitization of austenitic stainless steel materials and qualification of welding processes employed in production including testing using **ASTM A-262 Practice A or E** or another method which can be demonstrated to show non-sensitization. Alternative tests that have been previously accepted, based upon the adequacy of justifications presented and circumstances of proposed use, include the use of **ASTM A-708**.	Y	5.2.3.4
	B. The requirements of **GDC 4** relative to compatibility of components with environmental conditions are met with additional controls to avoid stress corrosion cracking in austenitic stainless steels. These controls consist of acceptance criteria on prevention of contamination, cleaning, and upper limit on yield strength. Additional controls for avoiding stress corrosion cracking are applied to BWRs as described	Y	5.2.3.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	below. Controls to avoid stress corrosion cracking in austenitic stainless steels are also covered in **Regulatory Guide 1.44**. This guide provides acceptance criteria on the cleaning and protection of the material against contaminants capable of causing stress corrosion cracking. Acid pickling is to be avoided on fabricated stainless steels. Necessary pickling is to be done only with appropriate controls. Pickling should not be performed upon sensitized stainless steels. The quality of water used for final cleaning or flushing of finished surfaces during installation should be in accordance with **Regulatory Guide 1.37**, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants.” Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water need not be controlled. The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in **Regulatory Guide 1.37 position C.5** to prevent contamination which promotes stress corrosion cracking. Tools which contain materials that could contribute to intergranular or stress corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces. Laboratory stress corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	on the yield strength of 620 MPa (90,000 psi). Additional controls, beyond those described above, are warranted to avoid intergranular stress corrosion cracking (IGSCC) in and near welds in BWR austenitic stainless steel piping. The affected piping and the additional controls are described in Attachment A to Generic Letter 88-01 or NUREG-0313. These controls include material and weldment specifications for IGSCC resistant materials, processing techniques, categorization of the IGSCC resistance of installations based upon material properties, treatment history, and post-weld treatments. The technical bases for these controls are described in NUREG-0313.		
	C. The acceptance criteria for compatibility of austenitic stainless steel with thermal insulation are based on **Regulatory Guide 1.36**, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel,” to satisfy **GDC 14 and 31** relative to prevention of failure of the RCPB. The compatibility of austenitic stainless steel materials with thermal insulation is dependent upon the type of insulation. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation which meets the criteria of Regulatory Guide 1.36 is used. The acceptance criteria for nonmetallic insulation for stainless steel are based on the levels of leachable contaminants in the material and are presented in position C.2.b and Figure 1 of the guide.	Y	5.2.3.4
	D. The acceptance criteria for control of welding of austenitic stainless steels are based on **NUREG-0313** as described below and on Regulatory Guides 1.31, 1.34, and 1.71, to satisfy the quality	Y	5.2.3.4

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	standards requirements of **GDC 1**, **GDC 30**, and **10 CFR 50.55a**. The acceptance criteria for delta ferrite in austenitic stainless steel welds are given in **Regulatory Guide 1.31**, “Control of Ferrite Content in Stainless Steel Weld Metal.” These acceptance criteria cover (1) verification of delta ferrite content of filler metals, (2) ferrite measurement, (3) instrumentation, (4) acceptability of test results, and (5) documentation of weld pad verification tests. For the BWR austenitic stainless steel RCPB piping specified in Generic Letter 88-01, the weld metal ferrite content should be controlled as described in the positions of Attachment A to Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2. The acceptance criteria for electroslag welds in austenitic stainless steel are given in **Regulatory Guide 1.34**, “Control of Electroslag Weld Properties.” These criteria specify acceptable solidification patterns for qualification of austenitic stainless steel welds and the basis for verifying conformance during production welding. **Regulatory Guide 1.71** provides the following criteria for requalification of welders: (1) The performance qualification should require testing of the welder when conditions of accessibility to a production weld are less than 30 to 35 cm (12-14 inches) in any direction from the joint. (2) Requalification should be required for different restricted accessibility conditions or when other essential variables listed in the Code, Section IX, are changed. An alternate acceptance criterion is as stated in Subsection II.3.B of this SRP section.		
	E. For nondestructive examination of austenitic stainless steel tubular	Y	5.2.3.4

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	products, the quality standards requirements of **GDC 1**, **GDC 30**, and **§ 50.55a** are met by compliance with the applicable provisions of the ASME Code. The acceptance criteria are given in Section **III of the Code, Paragraphs NB-2550 through NB-2570**.		
	F. Not Used		
	G. Operational Programs.	Y	5.2.3
	For COL reviews, the description of the operational program and proposed implementation milestones for the Inservice Inspection and Inservice Testing Programs are reviewed in accordance with **10 CFR 50.55a(g)** and **10 CFR 50, Appendix A**. The implementation milestones in the Inservice Inspection and Inservice Testing Programs are identified under **SRP Section 5.2.4**.	N/A - COL	5.2.4
SRP 5.2.4	**Reactor Coolant Pressure Boundary Inservice Inspection and Testing (R2, 03/2007)**		
5.2.4-AC-01	**General Design Criterion (GDC) 32** found in Appendix A to Part 50, as it relates to periodic inspection and testing of the RCPB;	Y	5.2.4
5.2.4-AC-02	**10 CFR 50.55a**, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code;	Y	5.2.4
5.2.4-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	ITAAC	Tier 1

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	the Atomic Energy Act, and the NRC's regulations;		
5.2.4-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.2.4-SAC-01	System Boundary Subject to Inspection. The applicant's or licensee's definition of the RCPB is acceptable if it is in agreement with the following criteria: for pressurized water reactor (PWR) and boiling water reactor (BWR) nuclear power systems, the inspection requirements of **10 CFR 50.55a**, as detailed in **Section XI of the ASME Code**, must be met for all Class 1 pressure-containing components (and their supports). The system boundary, as defined in **10 CFR 50.2**, includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:		
	A. The outermost containment isolation valve in system piping that penetrates the primary reactor containment.	Y	5.2
	B. The second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary reactor containment.	Y	5.2
	C. The reactor coolant system safety and relief valves.	Y	5.2

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5.2.4-SAC-02	Accessibility. The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with **Subarticle IWA-1500**, “Accessibility,” **of the ASME Code, Section XI**.	Y	5.2.4.1.1
5.2.4-SAC-03	Examination Categories and Methods. The examination categories and methods specified in the SAR are acceptable if they are in agreement with the criteria in **Article IWB-2000**, “Examination and Inspection,” **of Section XI of the ASME Code**. Every area subject to examination should fall within one or more of the examination categories in Article IWB-2000 and should be examined at least to the extent specified. The methods of examination for the components and parts of the pressure retaining boundaries are also listed in the requirements of Article IWB-2000 of Section XI of the ASME Code. The applicant’s or licensee’s examination techniques and procedures used for preservice examination or inservice inspection of the system are acceptable if they are in agreement with the following criteria: A. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with **Article IWA-2000**, “Examination and Inspection,” **and Article IWB-2000**, “Examination and Inspection,” **of Section XI of the ASME Code**. B. The acceptance standards for the examination results required by 3.A above are given in **Section XI, Article IWB-3000**, “Acceptance Standards.” C. The methods, procedures, and requirements for qualification of personnel performing ultrasonic examination are in accordance with	Y	5.2.4.1.2

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	the requirements of **Appendix VII to Division 1 of Section XI of the ASME Code**. D. Performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws is in accordance with the requirements of **Appendix VIII of Section XI of the ASME Code**. E. The methods, procedures, and requirements for ultrasonic examination of reactor-vessel-to-flange welds, closure-head-to-flange welds, and integral attachment welds incorporate the regulatory positions provided in **Regulatory Guide 1.150**, unless qualified by performance demonstration in accordance with the requirements of **Appendix VIII of Section XI of the ASME Code**.		
5.2.4-SAC-04	<u>Inspection Intervals</u>. The required examinations and pressure tests must be completed during each ten-year interval of service, hereinafter designated as the inspection interval. In addition, the scheduling of the program must comply with the provisions of **Article IWA-2000**, “Examination and Inspection,” concerning inspection intervals **of Section XI of the ASME Code**.	Y	5.2.4.1.3
5.2.4-SAC-05	<u>Evaluation of Examination Results</u>. A. The standards for evaluation of examination results are acceptable if they are in accordance with the requirements of **Section XI, Article IWB-3000**, “Acceptance Standards.” B. The proposed program regarding repair or replacement of components containing defects is acceptable if the program is in accordance with the requirements of **Section XI, Article IWA-4000**, “Repair/Replacement Activities.” The criteria that establish the need	Y	5.2.4.1.4

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	for repair or replacement are described in **Section XI, Article IWB-3000**, "Acceptance Standards." C. The standards for evaluation of examination results should be in accordance with the requirements of **Sections XI, Article IWB-3000**, "Acceptance Standards," if **Regulatory Guide 1.150** is used.		
5.2.4-SAC-06	System Pressure Tests. The pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program is acceptable if the program is in accordance with the requirements of **Section XI, Article IWB-5000**, "System Pressure Tests," and the technical specification requirements for operating limitations during heatup, cooldown, and system hydrostatic pressure testing. In some cases, these limitations may be more severe than those in Article IWB-5000.	Y	5.2.4.1.5
5.2.4-SAC-07	Code Exemptions. Exemptions from Code examinations should be permitted if the criteria in **Subsubarticle IWB-1220**, "Components Exempt from Examination," are met. The applicant's or licensee's program should list the exemptions taken in accordance with the ASME Code.	Y	5.2.4.1.6
5.2.4-SAC-08	Code Cases**.** ASME code cases referenced by the COL application are reviewed for acceptability and compliance with **Regulatory Guide 1.147**. Code cases not specifically referenced in Regulatory Guide 1.147 will be reviewed and accepted on a case-by-case basis.	Y	5.2.4.1.8

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5.2.4-SAC-09	<u>Augmented ISI to Protect Against Postulated Piping Failures</u>. The reviewer verifies that the high-energy system piping between containment isolation valves should receive an augmented ISI as follows: A. Protective measures, pipe whip restraints, structures, supports and guard pipes should not prevent access required to conduct the inservice examinations specified in the **ASME Code, Section XI, Division 1**. B. For those portions of high-energy fluid system piping between containment isolation valves, the inservice examination completed during each inspection interval should provide 100% volumetric examination of circumferential and longitudinal pipe welds. C. For those portions of high-energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in the portion of the guard pipe passing through the annulus of dual-barrier containment structures. D. The areas subject to examination should be defined in accordance with the Examination Category for Class 1 piping welds specified in **Article IWB-2000**.	Y	5.2.4.1.9 6.6
5.2.4-SAC-10	<u>Other Inspection Programs</u>. A. For BWR plants, the reviewer ascertains that the ISI program addresses the staff positions concerning augmented inspections for intergranular stress corrosion cracking (IGSCC) provided in Generic Letter 88-01, Supplement 1 to Generic Letter 88-01, and NUREG-0313, Revision 2.	N/A-BWR	N/A

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	B. For BWR plants, the reviewer ascertains that the ISI program adequately addresses the augmented inspections of feedwater and control rod drive nozzles as discussed in NUREG-0619. The staff may approve alternatives to the inspection guidelines in NUREG-0619.	N/A-BWR	N/A
	C. For PWR plants, the reviewer verifies that the applicant or licensee has established a program to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in **Generic Letter 88-05**.	Y	5.2.3.4.3 5.2.4.1.10
	D. For Westinghouse PWR plants, the reviewer verifies that the applicant or licensee has established an inspection program to periodically confirm the integrity of incore neutron-monitoring system thimble tubes, as described in NRC Bulletin 88-09.	N/A-VEN	N/A
5.2.4-SAC-11	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Preservice Inspection, Inservice Inspection and Inservice Testing Programs are reviewed in accordance with **10 CFR 50.55a(g)** and **10 CFR Part 50, Appendix A**.	N/A-COL	N/A
SRP 5.2.5	**Reactor Coolant Pressure Boundary Leakage Detection (R2, 03/2007)**		
5.2.5-AC-01	**GDC 2**, as it relates to SSC being designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, seiches, and tsunami without loss of capability to perform their safety functions.	Y	5.2.5

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5.2.5-AC-02	**GDC 30**, as it relates the components which are part of the RCPB being designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	Y	5.2.5
5.2.5-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.2.5-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.2.5-SAC-01	For **GDC 2**, acceptance is based on the guidelines of **RG 1.29, Positions C.1 and C.2**.	Y	5.2.5
5.2.5-SAC-02	For **GDC 30**, acceptance is based on meeting the guidelines of **RG 1.45**.	Y	5.2.5
SRP 5.3.1	**Reactor Vessel Materials (R2, 03/2007)**		
5.3.1-AC-01	**General Design Criteria (GDC) 1 and 30** found in Appendix A to Part 50, as they relate to quality standards for design, fabrication, erection,	Y	5.3.1

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	and testing of structures, systems and components;		
5.3.1-AC-02	**GDC 4**, as it relates to the compatibility of components with environmental conditions;	Y	5.3.1
5.3.1-AC-03	**GDC 14**, as it relates to prevention of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB);	Y	5.3.1
5.3.1-AC-04	**GDC 31**, as it relates to material fracture toughness;	Y	5.3.1
5.3.1-AC-05	**GDC 32**, as it relates to the requirements for a materials surveillance program;	Y	5.3.1
5.3.1-AC-06	**10 CFR 50.55a**, as it relates to quality standards for design, and determination and monitoring of fracture toughness;	Y	5.3.1
5.3.1-AC-07	**10 CFR 50.60**, “Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation,” as it relates to RCPB fracture toughness and material surveillance requirements of **10 CFR Part 50, Appendix G and Appendix H**;	Y	5.3.1
5.3.1-AC-08	**10 CFR Part 50, Appendix B, Criterion XIII**, as it relates to onsite material cleaning control;	Y	5.3.1
5.3.1-AC-09	**10 CFR Part 50, Appendix G**, as it relates to materials testing and acceptance criteria for fracture toughness,	Y	5.3.1
5.3.1-AC-10	**10 CFR Part 50, Appendix H**, as it relates to the determination and monitoring of fracture toughness,	Y	5.3.1
5.3.1-AC-11	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance	ITAAC	Tier 1

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	criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
5.3.1-AC-12	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.3.1-SAC-01	**1.** <u>Materials</u>. The requirements of **GDC 1 and 30** and **10 CFR 50.55a** regarding quality standards are met by compliance with the provisions of the **ASME Code, Section III**, for materials, as detailed below: A. Acceptable materials for the reactor vessel and its appurtenances and attachments are those identified in the **Code, Section III, Appendix I**. The materials must also meet the requirements of **10 CFR Part 50, Appendix G**. B. The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of **Code Section III, Appendix IV-1000** and **10 CFR Part 50, Appendix G**. These data must include information on mechanical properties, weldability, and physical changes of the material.	Y	5.3.1

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5.3.1-SAC-02	Special Processes Used for Manufacture and Fabrication of Components. The requirements of **GDC 1 and 30** and **10 CFR 50.55a** regarding quality standards are met by compliance with the provisions of the **ASME Code, Section III**, for fabrication of components. The reactor vessel and its appurtenances are fabricated and installed in accordance with **Code Section III, Paragraph NB-4100**. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with **Code Section III, Article NCA-8000**, that the materials used comply with the requirements of **NB-2000**, and that the fabrication or installation comply with the requirements of **NB-4000**.	Y	5.3.1
5.3.1-SAC-03	Special Methods for Nondestructive Examination. The requirements of **GDC 1 and 30** and **10 CFR 50.55a** regarding quality standards are met by compliance with the **ASME Code, Section III**, for fabrication nondestructive testing. The acceptance criteria for examination of the reactor vessel and its appurtenances by nondestructive examination are those specified in **Code Section III, NB-5000**.	Y	5.3.1
5.3.1-SAC-04	Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels. The acceptance criteria for special controls and processes in welding austenitic or ferritic steel components are based upon the following regulatory guides, ASME Code provisions, and other regulatory documents necessary to satisfy the relevant requirements of **GDC 1, 4, 14, and 30**; **Appendix B**; and **10 CFR 50.55a**.		

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	A. Only those welding processes capable of producing welds in accordance with the welding procedure qualification requirements of **Code Sections III and IX** may be used. Any process used shall be such that the records required by **NB-4300 of Section III** can be made, with the exception of stud welding, which is acceptable only for minor nonpressure attachments.	Y	5.3.1
	B. **ASME Code Sections III and IX** criteria for welding ferritic steel are supplemented by the regulatory positions in **Regulatory Guides (RGs) 1.50**, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” **and 1.34**, “Control of Electroslag Weld Properties.”	Y	5.3.1
	C. The regulatory positions of **RG 1.43**, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” provide the acceptance criteria to avoid underclad cracking of stainless steel clad ferritic components.	Y	5.3.1
	D. **ASME Code Sections III and IX** criteria for welding austenitic stainless steels are supplemented by the regulatory positions in **RG 1.31**, “Control of Ferrite Content in Stainless Steel Weld Metal,” and **RG 1.34**. For the BWR austenitic stainless steel reactor vessel attachments and appurtenances specified in Generic Letter (GL) 88-01, the weld metal ferrite content should be controlled as described in the positions of Attachment A to GL 88-01 or the recommendations of NUREG-0313, Revision 2.	Y	5.3.1
	E. The regulatory positions of **RGs 1.44**, “Control of the Use of Sensitized Stainless Steel,” **and 1.37**, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” provide the	Y	5.3.1

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	acceptance criteria to avoid sensitization and contamination of stainless steel. RG 1.44 states that non-sensitization should be verified using **ASTM A-262** Practices A or E, or another method that can be demonstrated to show nonsensitization of austenitic stainless steel. Alternative tests to those in ASTM A-262 that have been previously accepted include **ASTM A 708**. For BWRs, the control of sensitized steel per RG 1.44 should be modified as necessary to conform with the positions in Attachment A to GL 88-01 or the recommendations of NUREG-0313. The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in **RG 1.37 position C.5** to prevent contamination which promotes stress corrosion cracking. Tools which contain materials that could contribute to intergranular or stress-corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces.		
	F. Additional controls, beyond those described above, are considered necessary to avoid intergranular stress corrosion cracking (IGSCC) in and near welds in BWR austenitic stainless steel reactor vessel attachments and appurtenances. The additional controls are described in Attachment A to GL 88-01 and in NUREG-0313, Revision 2. These controls include material and weldment specifications for IGSCC resistant materials, processing techniques, categorization of the IGSCC resistance of installations based upon material properties, treatment history, and post-weld treatments. The technical bases for these controls are described in NUREG-0313,	N/A-BWR	N/A

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	Revision 2.		
5.3.1-SAC-05	Fracture Toughness. The acceptance criteria for this area of review are the requirements **of Appendix G of 10 CFR Part 50**. These criteria satisfy the requirements of **GDC 31** and **10 CFR 50.60** regarding materials testing and acceptance standards for fracture toughness. Appendix G requires that the reactor vessel and appurtenances thereto which are made of ferritic materials shall meet the following minimum requirements for fracture toughness during system hydrostatic tests, conditions of normal operation, and anticipated operational occurrences:	Y	5.3.1
	A. The ferritic materials shall be tested in accordance with the **ASME Code paragraph NB-2300** including: i. T_{NDT} shall be determined for each material by means of a drop weight test. ii. The materials shall meet the acceptance standards of paragraph NB-2330 of the Code, which states that at a temperature not greater than (T_{NDT} + 33°C)[(T_{NDT} + 60°F)] each Charpy C_v specimen tested shall exhibit at least 0.89 mm (35 mils) lateral expansion and not less than 68 J (50 ft-lbs) of absorbed energy. When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}. iii. In the event that the above requirements are not met, additional C_v notch impact tests are performed (in groups of three specimens) to determine the temperature T_{cv} at which they are met. In this case the reference temperature $RT_{NDT} = T_{cv}$ - 33°C ($RT_{NDT} = T_{cv}$- 60°F). Thus the reference temperature RT_{NDT} is the	Y	5.3.1

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	higher of T_{NDT} and (T_{cv} - 33°C) [(T_{cv} - 60°F)] iv. When a C_v impact test has not been performed at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)], or when the C_v impact test at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)] does not exhibit a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion, a temperature representing a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v impact tests performed.		
	B. In addition to the above criteria, the requirements of **paragraphs IV.A.1, IV.A.2, and IV.B of Appendix G of 10 CFR Part 50** and **10 CFR 50.61(b)(2)** (for PWRs) shall be met. i. **SRP Section 5.3.2** discusses the requirements of **paragraphs IV.A.2 and of Appendix G** in detail. ii. The acceptance criteria discussed in **paragraph IV.A.1 of Appendix G** states that reactor vessel belt-line materials shall have a minimum upper shelf energy of 102 J (75 ft-lbs) as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with **paragraph NB-2331(a) of the Code, Section III**. Reactor vessel belt-line materials must also maintain an upper shelf energy no less than 68 J (50 ft-lb) throughout the life of the vessel. These two requirements do not apply, however, if it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper shelf fracture energy are adequate.	Y	5.3.1
	C. The neutron radiation embrittlement effects on reactor vessel materials shall be determined in accordance with **10 CFR Part 50,**	Y	5.3.1

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	Appendix G, Section III, and **RG 1.99**, “Radiation Embrittlement of Reactor Vessel Materials.”		
5.3.1-SAC-06	Material Surveillance. The material surveillance acceptance criteria are the requirements of **Section III of Appendix H of 10 CFR Part 50**. Complying with the acceptance criteria satisfies the requirements of **GDC 32** regarding an appropriate material surveillance program for the reactor vessel.	Y	5.3.1
	A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1 MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.	N/A-OTHER	5.3.1 5.3.2.1
	B. Reactor vessels constructed of ferritic materials which do not meet the conditions in paragraph a. shall have their belt-line regions monitored by a surveillance program complying with the **American Society for Testing and Materials (ASTM) standard ASTM E-185**, except as modified by **Appendix H to 10 CFR Part 50**.	Y	5.3.1
	C. The surveillance program shall meet the following requirements: i. Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by **Section III of Appendix G of 10 CFR Part 50**. The specimen types shall comply with the requirements of **Section III.B of Appendix H**, except that drop-weight specimens are not required. ii. Surveillance capsules containing the surveillance specimens shall be located near the inside vessel wall in the belt-line region, so	Y	5.3.1

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	that the neutron flux received by the specimens approximates that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. If the capsule holders are attached to the vessel wall or cladding, inspection shall be done according to the requirements for permanent structural attachments as given in **ASME Code Sections III and XI**. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in **paragraph III.B.1 of Appendix H**. iii. The required number of capsules, which will vary from three to five depending upon the adjusted reference temperature at the end of the service lifetime of the reactor vessel, and their withdrawal schedules, shall be in accordance with the requirements of **paragraph III.B.2 of Appendix H**. iv. For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis in accordance with the requirements of **paragraph III.C of Appendix H**.		
	The material surveillance program criteria of **ASTM E-185** cited in **10 CFR Part 50, Appendix H**, is predicated on an assumed 40-year reactor vessel design life. For those applicants proposing a facility with greater than a 40-year design life, the criteria of ASTM E-185 must be supplemented to provide for monitoring of the reactor vessel materials for the entire reactor vessel design life.		

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	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Reactor Vessel Material Surveillance Program are reviewed in accordance with **10 CFR 50.60** and **10 CFR 50, Appendix H**. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.	N/A-COL	N/A
5.3.1-SAC-07	Reactor Vessel Fasteners. The acceptance criteria for the reactor vessel bolting material are given by **paragraph IV.A of Appendix G to 10 CFR Part 50** and by the recommendations of **RG 1.65**, “Materials and Inspections for Reactor Vessel Closure Studs.” These acceptance criteria satisfy the quality standards requirements of **GDC 1**, **GDC 30**, and **10 CFR 50.55a**, and meet the requirements of **GDC 31** regarding prevention of fracture of the RCPB.	Y	5.3.1
	A. Materials for reactor vessel studs (and other fasteners) that are considered suitable are SA-540 Grades B-23 and B-24, SA-193 Grade B-7, SA-194 Grade 7, and SA-320 Grade L-43, as presented in **Section II of the ASME Code**.	Y	5.3.1
	B. The fastener material should not have an ultimate tensile strength over 1170 MPa (170 ksi), and the fracture toughness tests and acceptance levels of NB-2333 of Section III of the Code must be met as required by **paragraph IV.A of Appendix G to 10 CFR Part 50**.	Y	5.3.1
	C. Surface treatments, plating, or thread lubricants used should be shown to be compatible with the materials, and stable at operating temperatures.	Y	5.3.1

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	D. Nondestructive examination should be performed according to **Section III of the Code, subsubarticle NB-2580** including additional recommendations given in **Regulatory Position C.2 of RG 1.65**.	Y	5.3.1
SRP 5.3.2	**Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock (R2, 03/2007)**		
5.3.2-AC-01	**10 CFR 50.55a**, as it relates to quality standards for the design, fabrication, erection, and testing of SSCs important to safety.	Y	5.3.2
5.3.2-AC-02	**10 CFR 50.60**, as it relates to compliance with the requirements of **Appendix G to 10 CFR Part 50**.	Y	5.3.2
5.3.2-AC-03	**10 CFR 50.61**, as it relates to fracture toughness criteria for PWRs relevant to PTS events.	Y	5.3.2
5.3.2-AC-04	**General Design Criterion (GDC) 1**, found in Appendix A to 10 CFR Part 50, as it relates to quality standards for design, fabrication, erection, and testing.	Y	5.3.2
5.3.2-AC-05	**GDC 14**, as it relates to ensuring an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture of the RCPB.	Y	5.3.2
5.3.2-AC-06	**GDC 31**, as it relates to ensuring that the RCPB will behave in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.	Y	5.3.2
5.3.2-AC-07	**GDC 32**, as it relates to the reactor vessel materials surveillance program.	Y	5.3.2
5.3.2-AC-08	**Appendix G to 10 CFR Part 50**, as it relates to material testing and fracture toughness.	Y	5.3.2

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
5.3.2-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.3.2-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.3.2-SAC-01	Pressure-Temperature Limits		
	A. Applicable Regulations, Codes, and Basis Documents. The regulations in **10 CFR 50.60** and associated **Appendix G to 10 CFR Part 50** describe the conditions that require P-T limits and provide the general basis for these limits. Appendix G specifically requires that P-T limits must be at least as conservative as limits obtained by following **Appendix G to Section XI of the ASME Code** during heatup, cooldown, and test conditions. Appendix G to 10 CFR Part 50 also requires additional safety margins when the reactor core is critical. Since the regulations may not have included specific fracture	Y	5.3.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	toughness testing requirements for the ferritic materials in the pressure-retaining components at the time some of the reactor facilities were designed and constructed, **Branch Technical Position (BTP 5-3)** describes procedures for making estimates and assumptions concerning the fracture toughness properties of materials in the older plants. Although Appendix G to Section III of the ASME Code is usually referenced with regard to facility design and construction, the reviewer should instead apply the provisions of **Appendix G to Section XI of the ASME Code** when using this SRP. The following provide the rationale for using Appendix G to Section XI of the ASME Code instead of Appendix G to Section III of the ASME Code: i. **Appendix G to 10 CFR Part 50** specifically references **Appendix G to Section XI to the ASME Code**, and **Appendix G to Section III to the ASME Code** contains similar provisions. ii. The differences between **Appendix G to Section XI of the ASME Code** and **Appendix G to Section III of the ASME Code** have resulted from a series of ASME code cases, including N-588, N-640, and N-641. Appendix G to Section III of the ASME Code has not been updated since those code cases were developed. However, the staff expects that Appendix G of Section III of the ASME Code will be updated to be consistent with Appendix G to Section XI of the ASME Code.		
	B. <u>Pressure-Temperature Requirements</u>. **Appendix G to 10 CFR Part 50** requires that the pressure-temperature (P-T) limits defined in that Appendix be at least as	Y	5.3.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	conservative as limits obtained by following the methods of analysis and the margins of safety of **Appendix G of Section XI of the ASME Code**, as stated below: i. Pressure-Temperature Limits for Preservice Hydrostatic Tests During preservice hydrostatic tests (if fuel is not in the vessel), a material's lower bound static crack initiation fracture toughness, K_{Ic}, must be greater than the K_I caused by pressure stresses acting on a defined, conservative hypothetical flaw, as shown in the following expression: $K_{applied} = K_I(pressure) < K_{Ic}$ ii. Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests During performance of inservice leak and hydrostatic tests, a material's K_{Ic} must be greater than 1.5 times the K_I caused by pressure, as shown in the following expression: $K_{applied} = 1.5 K_I(pressure) < K_{Ic}$ iii. Pressure-Temperature Limits for Heatup and Cooldown Operations At all times during heatup and cooldown operations, a material's K_{Ic} must be greater than the sum of 2 times the K_I caused by pressure and the K_I caused by thermal gradients, as shown in the following expression: $K_{applied} = 2K_I(pressure) + K_I(thermal) < K_{Ic}$ iv. Pressure-Temperature Limits for Core Critical Operation At all times that the reactor core is critical (except for low-power physics tests), the temperature must be higher than that required for		

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	inservice hydrostatic testing. In addition, the P-T relationship must provide at least a 22 °C (40 °F) margin over that required for heatup and cooldown operations.		
5.3.2-SAC-02	Upper-Shelf Energy		
	A. Applicable Regulations, Codes, and Basis Documents. **Appendix G to 10 CFR Part 50** requires that reactor vessel beltline materials have a Charpy USE value in the transverse direction for base material and along the weld for weld material according to the **ASME Code** of no less than 102 J (75 ft-lb) initially and must maintain a Charpy USE value throughout the life of the vessel of no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by **Appendix G of Section XI of the ASME Code**.	Y	5.3.2
	B. Upper-Shelf Energy Requirements. **Appendix G to 10 CFR Part 50** contains the following USE requirements: i. Initially, the USE value in the transverse direction for base material and along the weld must not be less than 102 J (75 ft-lb). ii. Charpy USE throughout the life of the vessel must be maintained at no less than 68 J (50 ft-lb), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by **Appendix G of Section XI of the ASME Code**.	Y	5.3.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
5.3.2-SAC-03	Pressurized Thermal Shock		
	A. Applicable Regulations, Codes, and Basis Documents. Projected values of RT_{PTS} must be determined for PWR reactor vessel beltline materials in accordance with **10 CFR 50.61**. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed.	Y	5.3.2
	B. Pressurized Thermal Shock Requirements. In accordance with **10 CFR 50.61**, values of RT_{PTS} projected using the methods of 10 CFR 50.61 for the time of the initial application submittal and for the projected expiration date of the operating license must not exceed the screening criteria of 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials, throughout the facility's licensed operating permit. This assessment must be updated whenever projected values of RT_{PTS} change significantly, or upon request for a change in the expiration date for operation of the facility. For RT_{PTS} values projected to exceed the screening criteria, safety analyses must be provided that include proposed flux reduction programs or other corrective actions to prevent potential PTS-related failure of the reactor vessel if continued plant operation beyond the screening criterion is allowed.	Y	5.3.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 5.3.3	**Reactor Vessel Integrity (R2, 03/2007)**		
5.3.3-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.3.3-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.3.3-SAC-01	Design. With regard to compatibility of design with material properties and fabrication methods, the quality standards requirements of **GDC 1**, **GDC 30**, and **§ 50.55a** are met by compliance with the provisions of the ASME boiler and pressure vessel code. The basic acceptance criteria for the design of the vessel are the requirements of **Section III of the ASME Boiler and Pressure Vessel Code** (hereafter “the Code”). The design of the reactor vessel must be compatible with the properties of the materials used, and must permit construction by the use of standard and well proven fabrication methods. The design details should not include new or	Y	5.3.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	novel concepts unless they are substantiated by a comprehensive justification showing that no aspects of the design will compromise the overall integrity of the vessel in any manner. The design details must be adequate to permit all required inspections and to provide required access to all areas requiring inservice inspection in conformance with **Section XI of the Code**, as detailed in **SRP Section 5.2.4**. This satisfies the requirements of **GDC 32** and **§ 50.55a** regarding inservice inspection. If the procedures of **Section IV.A of Appendix G**, “Fracture Toughness Requirements,” to **10 CFR Part 50** do not indicate the existence of an equivalent safety margin, then **Section IV.B** allows the reactor vessel beltline to be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of **10 CFR 50.66**, “Requirements for thermal annealing of the reactor pressure vessel.” Annealing of the reactor vessel provides assurance that fracture toughness properties can be restored to satisfy the fracture toughness requirements of **GDC 31**.		
5.3.3-SAC-02	<u>Materials of Construction</u>. The basic acceptance criteria for the materials used in the construction of the reactor vessel, and the regulations that they satisfy, are detailed in SRP **Sections 5.2.3 and 5.3.1**. These criteria are the requirements of **Appendix G, 10 CFR Part 50**, as augmented by **Sections III and IX of the Code**. The materials must be compatible with the design requirements in the GDC. Acceptability is based on standard practice and engineering judgement, with consideration being given to such factors as material	Y	5.3.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	form, size-related variations in properties, and nonisotropic characteristics. Although many materials are acceptable for reactor vessels according to **Section III of the Code**, the special considerations relating to fracture toughness and radiation effects effectively limit the basic materials that are currently acceptable for most parts of reactor vessels to SA 533 Gr B C1 1, SA 508 C1 2, and SA 508 C1 3. Acceptability criteria for other grades will have to be developed before they can be used. Material compositions and expected neutron fluence must be compatible with the requirements for the material surveillance program. The reviewer uses published data to ensure that the predicted shift in toughness properties (RT_{NDT} and upper shelf energy) is conservative, based on actual material composition and predicted fluence. The predicted shift in toughness properties should be at least as conservative as that obtained by use of the most recent revision of **Regulatory Guide (RG) 1.99**. Acceptability of the material surveillance program, as specified in **Appendix H**, "Reactor Vessel Material Surveillance Program Requirements," **of 10 CFR Part 50**, depends on these relationships.		
5.3.3-SAC-03	Fabrication Methods. Acceptance criteria for the basic fabrication processes and their qualification and control requirements, and the regulations satisfied by these criteria, are detailed in **SRP Section 5.3.1**. These criteria are given in **Sections III and IX of the Code**. Although a particular fabrication process (such as multiple wire-high heat input welding) may be generally acceptable, it may not be suitable for reactor vessel fabrication for some materials without further justification	Y	5.3.3

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	or qualification. The reviewer uses “state-of-the-art” criteria and past practice to evaluate the acceptability of materials process combinations. Because fabrication methods, materials, and the effectiveness of nondestructive evaluation methods are interrelated, the reviewer should rely on state-of-the-art knowledge and past practice to determine whether the proposed combinations are compatible and acceptable.		
5.3.3-SAC-04	<u>Inspection Requirements</u>. The basic requirements for performing nondestructive inspections, the quality assurance criteria for the reactor vessel, and the regulations that all of these criteria satisfy, are detailed in **SRP Section 5.3.1**. These requirements and criteria are contained in **Section III of the Code**. Additional criteria are contained in **Section V of the Code**. Acceptance criteria for compatibility with materials and fabrication areas are discussed in previous sections. Very important relationships are those among in-process and final shop inspections, and the inservice inspection requirements of **Section XI of the Code**. The reviewer should determine whether the methods of inspection, the sensitivity levels, and flaw evaluation criteria are compatible with Section XI, and whether the results of the preservice baseline inspection can be correlated with the results of later inservice inspections.	Y	5.3.3
5.3.3-SAC-05	<u>Shipment and Installation</u>. The basic acceptance criteria for procedures and care to maintain proper cleanliness and freedom from contamination during all stages of shipping, storage, and installation of the reactor vessel, and the regulations that these criteria satisfy, are given in **SRP Section 5.2.3**.	Y	5.3.3

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	The purpose of this area of review is to verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. Acceptability in these areas is assured for current designs and materials by compliance with the basic acceptance criteria. If nonstandard materials or designs are used, the reviewer should determine whether criteria will be adequate, based on current technology. If the basic criteria are not followed, either intentionally or through error, the reviewer should evaluate, on a case basis, whether the integrity of the reactor vessel is compromised, using current technology, past practice, and experience as applicable.		
5.3.3-SAC-06	<u>Operating Conditions</u>. Acceptance criteria for operating limits for the reactor vessel, and the regulations that they satisfy, are detailed in **SRP Section 5.3.2**. These acceptance criteria are given in **Appendix G**, “Fracture Toughness Requirements,” to **10 CFR Part 50** and for PWRs, **10 CFR 50.61**, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.” The criterion for acceptable behavior is that the vessel remains leaktight enough to support adequate core cooling. The generally accepted principles and procedures of linear elastic fracture mechanics provide the basis for acceptance of analyses that support conformance with this criterion.	Y	5.3.3
5.3.3-SAC-07	<u>Inservice Surveillance</u>. The acceptance criteria for adequacy of the reactor vessel materials surveillance program, and the regulations satisfied by the criteria, are detailed in **SRP Section 5.3.1**. The criteria are based on the	Y	5.3.3

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	requirements of **Appendix H**, “Reactor Vessel Material Surveillance Program Requirements,” **to 10 CFR Part 50**. The SAR also provides information regarding the inservice inspections to be performed on the reactor vessel. The acceptance criteria for accessibility and inspection plan details, and the regulations that they satisfy, are detailed in **SRP Section 5.2.4**. These criteria are those of **Section XI of the Code**.		
5.3.3-SAC-08	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Inservice Inspection and Reactor Vessel Material Surveillance Programs are reviewed under SRP Section 5.2.4 and 5.3.1 respectfully, in accordance with 10 CFR 50.55a(g), 10 CFR 50.60 and 10 CFR 50, Appendix H. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.	Y	5.2.4 5.3.1
		N/A - COL	5.2.4 5.3.1
SRP 5.4	**Reactor Coolant System Component and Subsystem Design (R2, 03/2007)**		
5.4-AC-01	Specific requirements are identified in the applicable SRP sections.	Y	5.4
5.4-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1

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5.4-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
SRP 5.4.1.1	**Pump Flywheel Integrity (PWR) (R2, 03/2007)**		
5.4.1.1-AC-01	**GDC 1** and **10 CFR Part 50.55a(a)(1)**, as they relate to pump flywheel design, materials selection, fracture toughness, preservice and inservice inspection programs, and overspeed test procedures to determine their adequacy to assure a quality product commensurate with the importance of the safety function to be performed.	Y	5.4.1
5.4.1.1-AC-02	**GDC 4**, as it relates to protecting safety-related structures, systems, and components of nuclear power plants from the effects of missiles that might result from reactor coolant pump failure.	Y	5.4.1
5.4.1.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.4.1.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the	N/A-COL	N/A

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	proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
5.4.1.1-SAC-01	Materials Selection and Fabrication. The applicant's materials selection and fabrication are acceptable if they comply with the following criteria, which are derived from **Subsections C.1.a and C.1.c of RG 1.14**. The flywheel material is acceptable if it is produced by a process (such as vacuum melting or degassing) that minimizes flaws in the material and improves its fracture toughness properties. If the flywheel is flame cut from a plate or forging, at least 1.3 cm (1/2 inch) of material should be left on the outer and bore radii for machining to final dimensions.	Y	5.4.1
5.4.1.1-SAC-02	Fracture Toughness. The pump flywheel fracture toughness properties are acceptable if they comply with the following criteria, which are derived from **Subsection C.1.b and** supplemented by **Subsection B of RG 1.14** and the **ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Appendix G**, Protection Against Nonductile Failure. The material should be examined and tested to establish its fracture toughness property. The minimum K_{IC} of the material at the normal operating temperature of the flywheel should be 165 MPa $\sqrt{m}$ (150 ksi $\sqrt{in}$). Use of the direct test method to obtain K_{IC} is encouraged.	Y	5.4.1

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	Direct Test. The plane-strain fracture toughness, K_{IC}, should be obtained in accordance with **ASTM E 399-05** if linear elastic fracture mechanics is used in the fracture mechanics analysis. The J-resistance curve should be obtained in accordance with **ASTM E 1820-05a** if elastic-plastic fracture mechanics is used. Either test should be conducted at or below the operating temperature of the pump flywheel. Indirect Tests for Certain Steel. For flywheel materials made of ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, and ASME SA-516 Grade 65 steel, the fracture toughness values can be found in the **ASME Code, Section XI, Appendix A** as a function of the difference between operating temperature (T) and the RT_{NDT} of the flywheel material, i.e., $T\text{-}RT_{NDT}$. The RT_{NDT} of the flywheel material should be determined in accordance with **NB-2320 and NB-2330 of the ASME Code, Section III** based on the nil-ductility transition temperature (T_{NDT}) determined by dropweight tests (DWT) and the impact energy determined by Charpy V-notch (C_v) tests. NB-2320 specifies ASTM E-208-95a as the Standard for DWT tests and ASTM A-370 as the Standard for C_v tests. If this indirect approach is applied to flywheel materials other than ASME SA-533-B Class 1, ASME SA-508 Class 2, ASME SA-508 Class 3, or ASME SA-516 Grade 65 steel, justification should be given to establish equivalence of fracture toughness between the proposed flywheel material and those mentioned here.		
5.4.1.1-SAC-03	**3.** Preservice Inspection. The applicant's preservice inspection program, including finish machining	Y	5.4.1

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	and ultrasonic and surface inspections, is acceptable if it complies with the following criteria, which are derived from **Subsection C.4.a of RG 1.14**. A. Each finished flywheel should be subjected to a 100% volumetric examination by ultrasonic methods using procedures and acceptance criteria specified in **ASME Code, Section III, NB-2530** for plates, **and NB-2540** for forgings. B. If the flywheel is flame cut from a plate or forging, at least 1.3 cm (1/2 inch) of material should be left on the outer and bore radii for machining to final dimensions. C. Finish machined bores, keyways, splines, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. D. The inspection results should be appropriately documented to establish initial flywheel conditions, accessibility, and practicality of the program to be used as baseline information for future inservice inspections.		
5.4.1.1-SAC-04	Flywheel Design. The applicant's flywheel design is acceptable if it complies with the following criteria, which are derived from **Subsection C.2 of RG 1.14**. The flywheel should be designed to withstand normal conditions, anticipated transients, the design basis loss of coolant accident, and the safe shutdown earthquake without loss of structural integrity. The design of the pump flywheel should also meet the following criteria: A. The combined stresses at the normal operating speed due to centrifugal forces and the interference fit of the wheel on the shaft, should not exceed 1/3 of the minimum specified yield strength or 1/3	Y	5.4.1

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	of the measured yield strength in the weak direction of the material if appropriate tensile tests have been performed on the actual material of the flywheel. B. The design overspeed of a flywheel should be at least ten percent above the highest anticipated overspeed. The anticipated overspeed should include consideration of the maximum rotational speed of the flywheel if a break occurs in the reactor coolant piping in either the suction or discharge side of the pump. An acceptable basis for the assumed design overspeed, addressing pipe breaks consistent with the design basis for reactor coolant piping, should be submitted to the staff for review. C. The combined stresses at the design overspeed, due to centrifugal forces and the interference fit, should not exceed 2/3 of the minimum specified yield strength, or 2/3 of the measured yield strength in the weak direction if appropriate tensile tests have been performed on the actual material of the flywheel. D. The shaft and the bearings supporting the flywheel should be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake. E. A fracture mechanics analysis should be conducted for the life time of the flywheel, including extended operation, to predict the critical speed for fracture of the flywheel. The ratio of K_{IC} to the maximum tangential stress at speeds from normal to design overspeed should be at least 2 √in (consistent with **SRP 10.2.3**, “Turbine Disk Integrity”), or alternatively, the ratio of K_{IC} to the applied K should be 3.16 for normal and upset conditions and 1.41 for emergency and faulted		

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	conditions (consistent with the ASME Code approach). This fracture mechanics analysis should consider crack growth due to identified degradation mechanisms for the largest flaw which could be missed by inspection (use the NRC accepted value of 0.25 inch for Westinghouse Owners Group [WOG] and ABB Combustion Engineering Owners Group [CEOG] flywheels if a smaller value can not be justified). The analysis should be submitted as a topical report to the NRC staff for evaluation.		
5.4.1.1-SAC-05	Overspeed Test. The applicant's commitment to perform an overspeed test is acceptable if each flywheel assembly is tested at the design overspeed of the flywheel. This criterion is taken from **Subsection C.3 of RG 1.14**.	Y	5.4.1
5.4.1.1-SAC-06	Inservice Inspection (ISI). The applicant's ISI program is acceptable if it complies with the following criteria, which are derived from **Subsection C.4.b of RG 1.14**, operating experience, and staff's evaluation of WOG's and CEOG's fracture mechanics analyses on reactor coolant pump flywheels of operating plants. A. A volumetric examination by ultrasonic methods of the areas of higher stress concentration at the bore and keyway extending to half of the flywheel radius, or a surface examination by liquid penetrant or magnetic particle methods of all exposed surfaces, at approximately 10 operating year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the **ASME Code, Section XI**. Removal of the flywheel is not required.	Y	5.4.1

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	B. Examination procedures and acceptance criteria should be in conformance with the requirements specified in Subsection II.3.A of this SRP section.		
5.4.1.1-SAC-07	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Pre-Service Inspection, Inservice Inspection, and Inservice Testing Programs are reviewed in accordance with 10 CFR Part 50, Appendix A, 10 CFR 50.55a(a)(1) and 10 CFR 50.55a(f). The implementation milestone are completion prior to initial plant start-up, prior to commercial service and after generator on-line on nuclear heat.	Y N/A-COL	14.2 N/A
SRP 5.4.2.1	**Steam Generator Materials (R3, 03/2007)**		
5.4.2.1-AC-01	**General Design Criterion (GDC) 1** of Appendix A to 10 CFR Part 50 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. If generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented to provide adequate assurance that these SSCs will perform their safety functions and that records will be maintained.	Y	5.4.2
5.4.2.1-AC-02	**GDC 4** requires, in part, that SSCs important to safety should be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.	Y	5.4.2
5.4.2.1-AC-03	**GDC 14** requires that the RCPB should be designed, fabricated, erected,	Y	5.4.2

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	and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.		
5.4.2.1-AC-04	**GDC 15** requires that the reactor coolant system and associated auxiliary control and protection systems should be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Y	5.4.2
5.4.2.1-AC-05	**GDC 30** requires, in part, that components that are part of the RCPB should be designed, fabricated, erected, and tested to the highest quality standards practical.	Y	5.4.2
5.4.2.1-AC-06	**GDC 31** requires, in part, that the RCPB should be designed with sufficient margin to ensure that-when stressed under operating, maintenance, testing, and postulated accident conditions-the boundary behaves in a nonbrittle manner, thereby minimizing the probability of rapidly propagating fracture.	Y	5.4.2
5.4.2.1-AC-07	**10 CFR 50.55a(c)**, **10 CFR 50.55a(d)**, and **10 CFR 50.55a(e)** generally require certain grouping of components, including those compromising the pressure boundary, to meet the requirements of **Section III of the ASME Code**.	Y	5.4.2
5.4.2.1-AC-08	**Appendix B to 10 CFR Part 50** applies to the steam generator materials. Of particular note is Criterion XIII, which requires, in part, that measures shall be established to control the cleaning of material and equipment in accordance with work and inspection procedures to prevent damage or deterioration.	Y	5.4.2
5.4.2.1-AC-09	**Appendix G to10 CFR Part 50** requires that RCPB pressure-retaining	Y	5.4.2

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	components that are made of ferritic materials meet ASME Code requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences.		
5.4.2.1-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.4.2.1-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.4.2.1-SAC-01	<u>Selection, Processing, Testing, and Inspection of Materials</u> The materials selected for the steam generator form portions of the primary and secondary system pressure boundary. In addition, certain materials used for nonpressure- retaining components (including tube supports) can have a direct impact on the integrity of the pressure boundary (e.g., denting of the steam generator tubes from corrosion of a tube support or mechanical damage to the tubes from the generation of	Y	5.4.2

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	loose parts). As a result, the materials selected for the steam generator must be fabricated and tested to quality standards, as required by **GDC 1**. In addition, the materials selected for the RCPB must be fabricated and tested to the highest quality standards, as required by **GDC 30**. The materials selected for use in fabricating the steam generator are acceptable from a fabrication/manufacturing standpoint if they comply with 10 CFR 50.55a. In general, this regulation - specifically **10 CFR 50.55a(c)**, **10 CFR 50.55a(d)**, and **10 CFR 50.55a(e)** - requires that the components satisfy the requirements of Section III of the ASME Code. Provisions in **10 CFR 50.55a(b)** permit ASME Code cases, as discussed in **Regulatory Guide 1.84**, to be used to select, fabricate, and test materials for the steam generator. **Section III of the ASME Code** establishes - through articles such as **NCA-1000, NB-2000** (for Class 1 components), **and NC-2000** (for Class 2 components) - requirements for selecting, processing, testing, inspecting (during fabrication/ manufacturing), and certifying materials. In general, **Section III of the ASME Code** references **Parts C and D of Section II of the ASME Code** for permitted material specifications (e.g., in **Articles NB-2120 and NC-2120**). Examples of materials that are currently used for Class 1 components in the steam generator include the following: Tubing: ASME SB-163, N06690, Thermally Treated Pressure Plates: ASME SA-533, Grade B, Class 1 Pressure Forgings: (including nozzles and tubesheet): ASME SA-508, Grade 3, Class 2 (formerly referred to as class 3a)		

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	Nozzle Safe Ends: ASME SA-182, F316LN Channel Heads: ASME SA-508, Grade 3, Class 2 (formerly referred to as Class 3a) Cladding, Buttering, and Welds: ASME SFA 5.4 (308L, 309L), 5.9 (308L, 309L) 5.11 (ENiCrFe-7), and 5.14 (ERNiCrFe-7) Pressure Boundary Welds: Low Alloy Steel, SFA 5.5, 5.23, 5.28 Manway Studs: ASME SA-193, Grade B7 Manway Nuts: ASME SA-194 Examples of materials that are currently used for Class 2 components in the steam generator include the following: Pressure Plates: ASME SA-533, Grade B, Class 1 Bolting: ASME SA-193, Grade B7 Tube Support Structures (including antivibration bars/fan-bars): ASME SA-240, Type 405 and Type 410S In summary, for the purposes of satisfying **GDC 1** and **GDC 30**, the materials used in fabricating the steam generator are acceptable if they are selected, fabricated, tested, and inspected (during fabrication/manufacturing) in accordance with the ASME Code.		
5.4.2.1-SAC-02	<u>Steam Generator Design</u> The design of the steam generator should limit the potential for	Y	5.4.2

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	degradation so that the integrity of the steam generator, including the tubes, is maintained during the operating interval between inspections. Degradation of the steam generator tubes and other secondary side components that could affect tube integrity should be manageable through the steam generator program (reviewed under **SRP Section 5.4.2.2**). Degradation of other steam generator pressure boundary materials should be manageable through the inservice inspection program (the RCPB inservice inspection program is reviewed under **SRP Section 5.2.4**). The steam generator design is acceptable from a degradation standpoint if it accomplishes the following: A. Limits the crevice between the tube and the tube supports. This can be accomplished by using openings of various shapes (e.g., trifoil or quatrefoil) in tube support plates or by using lattice grid (eggcrate) tube supports. The design of the tube supports should promote high-velocity flow along the tubes. Limiting the crevices will limit the buildup of corrosion product and sludge, which can lead to corrosion of the tubes and the supports. B. Uses appropriate corrosion-resistant materials or employs cladding for materials susceptible to corrosion. To limit the potential for denting the tubes, the tube support structures should use a corrosion-resistant material. Tube denting is a phenomenon associated with corrosion of the tube support structures, creating a hard corrosion product that fills the crevice between the tube and the tube support. Denting of tubes can result in the restriction of primary coolant flow and stress-corrosion cracking of the tubes. To limit the steam generator tube's susceptibility to corrosion, the tubes should be heat-		

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	treated (e.g., thermally treated), as needed, to optimize their microstructure from a corrosion resistance standpoint. To reduce residual stresses in the U-bend region of short-radius tubes (and therefore the material's susceptibility to corrosion), the U-bend region of short-radius tubes should be stress-relieved after bending. The materials that support the tubes and other materials on the secondary side should be sufficiently resistant to degradation to ensure that the tubes will remain adequately supported and to reduce the potential for the generation of loose parts, which can result in a loss of tube integrity. The corrosion-resistant cladding on the tubesheet and on other primary side components should be weld-deposited, fabricated, and inspected according to the requirements in **Part QW of Section IX of the ASME Code**. C. Limits the crevice and residual stresses in the tubesheet region. The extent of the tube-to-tubesheet crevice should be limited. This can be accomplished by expanding the tube throughout the tubesheet region, if practical (given other design considerations such as the desired preload in the tube for once-through steam generators). The choice of the method for expansion should consider limiting the stresses in the tube. Limiting the crevices will restrict the buildup of corrosion product and sludge that can lead to corrosion. Limiting the stresses will diminish the potential for stress-corrosion cracking. D. Includes an appropriate allowance for deterioration (including corrosion) of the steam generator materials. This is accomplished through compliance with **Section III of the ASME Code** (**Articles NB-2160 and NB-3121** for Class 1 components **and Articles NC-2160 and NC-3121** for Class 2 components).		

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	E. Uses bolting material that will perform adequately under the expected service conditions and that is not subject to stress-corrosion cracking. This can be accomplished by following the regulatory positions in **Regulatory Guide 1.65**. Although Regulatory Guide 1.65 provides guidance for the design of reactor vessel closure studs, it is also appropriate for the selection of suitable steam generator bolting material. The integrity of bolting and threaded fasteners is also reviewed under **SRP Section 3.13**. The above criteria, in conjunction with the acceptance criteria for interfacing reviews and appropriately performed inservice inspections, as discussed above, provide assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture will be extremely low, (2) the design conditions of the RCPB are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of **GDC 14, 15, and 31**.		
5.4.2.1-SAC-03	Fabrication and Processing of Ferritic Materials		
	A. Fracture Toughness The steam generator is part of the primary and secondary system pressure boundary. As a result, the materials selected should be sufficient to avoid rapidly propagating failure and to ensure that the design conditions will not be exceeded during operation, consistent with the requirements of **GDC 14, 15, and 31**. The pressure-retaining ferritic materials selected for use in steam generators are acceptable from a fracture toughness standpoint if they (1) comply with **Appendix G to 10 CFR Part 50** and with **10 CFR Part 50, 10 CFR**	Y	5.4.2

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	50.55a(c), **10 CFR 50.55a(d)**, and **10 CFR 50.55a(e)** and (2) follow the provisions of **Appendix G to Section III of the ASME Code**. In general, the regulations cited above require the use of Section III of the ASME Code. **Articles NB-2300 and NC-2300 of Section III of the ASME Code** address fracture toughness requirements for Class 1 and 2, respectively. **Appendix G to Section III of the ASME Code** includes additional fracture toughness criteria.		
	B. Welding The joining of the materials used to fabricate a steam generator is critical to ensuring that it can properly function. Consistent with the requirements of **GDC 1** and **GDC 30** (for RCPB materials), the welding qualification, weld fabrication processes, and inspection during fabrication and assembly of the steam generator are performed by using quality standards (supplemented and modified, as necessary) commensurate with the importance of the functions to be performed. Ferritic steel welding of steam generator components is acceptable if it complies with 10 CFR Part 50, **10 CFR 50.55a(c)**, **10 CFR 50.55a(d)**, and **10 CFR 0.55a(e)** and meets the following: i. Controls the amount of specified preheat in accordance with the requirements of **paragraph D-1210 of Appendix D to Section III of the ASME Code**, as supplemented by **Regulatory Guide 1.50**. ii. Follows **Regulatory Guide 1.34**. iii. Follows **Regulatory Guide 1.71**. With respect to the qualification of the welder or welding operators when limited accessibility is an issue, these qualifications may be waived provided that 100-percent radiographic and/or ultrasonic examination of the	Y	5.4.2

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	completed welded joint is performed. In these cases, the examination procedures and acceptance standards should meet the requirements of **Section III of the ASME Code**. Records of the examination reports and radiographs should be retained as part of the quality assurance documentation for the completed weld. iv. Follows **Regulatory Guide 1.43**.		
5.4.2.1-SAC-04	Fabrication and Processing of Austenitic Stainless Steel (if austenitic stainless steel is used for pressure boundary applications)		
	A. Limiting Susceptibility to Cracking Various factors can make austenitic stainless steel susceptible to stress-corrosion cracking. These factors include the yield strength of the material, exposure of the material to contaminants during cleaning and operation, and presence or absence of material sensitization. Consistent with GDC 14, 15, and 31, limiting the potential for stress-corrosion cracking provides assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low, (2) the RCPB design conditions are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure. The fabrication and processing of austenitic stainless steel steam generator components is acceptable if it complies with 10 CFR Part 50, 10 CFR 50.55a(c), 10 CFR 50.55a(d), and 10 CFR 50.55a(e) and meets the following: i. Limits the yield strength to 620 megapascal (MPa) (90,000 pounds per square inch (psi)). Laboratory stress-corrosion cracking tests and service experience provide the basis for the	Y	5.4.2

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	criterion that the cold-worked austenitic stainless steels used in the RCPB should have an upper limit on yield strength. ii. Follows **Regulatory Guide 1.37**. With respect to the source of water for final cleaning or flushing of finished surfaces, vented tanks with deionized or demineralized water are an acceptable source. The oxygen content of this water need not be controlled; however, the concentrations of other chemical species (e.g., chloride, fluoride) should be limited to the values listed in **Regulatory Guide 1.44**. iii. Controls abrasive work on austenitic stainless steel surfaces in accordance with **Position C.5 of Regulatory Guide 1.37**, at a minimum. iv. Follows **Regulatory Guide 1.44**. In addition to the methods discussed in Regulatory Guide 1.44 for verifying that austenitic stainless steel is not sensitized, alternative tests that have been previously accepted, based upon the adequacy of justifications presented and circumstances of proposed use, include the use of **ASTM A-708**. v. Follows **Regulatory Guide 1.36**. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation that meets the criteria of Regulatory Guide 1.36 is used.		
	B. The joining of the materials used to fabricate a steam generator is critical to ensuring that it can properly function. Consistent with the requirements of **GDC 1** and **GDC 30** (for RCPB materials), the welding qualification, weld fabrication processes, and inspection	Y	5.4.2

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	during fabrication and assembly of the steam generator are performed using quality standards (supplemented and modified, as necessary) commensurate with the importance of the functions to be performed. Austenitic stainless steel welding of steam generator components is acceptable if it complies with 10 CFR Part 50, **10 CFR 50.55a(c)**, **10 CFR 50.55a(d)**, and **10 CFR 50.55a(e)** and meets the following: i. **Regulatory Guide 1.31** ii. **Regulatory Guide 1.34** iii. **Regulatory Guide 1.71** iv. **NUREG-0313**, which may be appropriate for any austenitic stainless steel steam generator materials		
5.4.2.1-SAC-05	Compatibility of Materials with the Primary (Reactor) and Secondary Coolant and Cleanliness Control The materials used in the steam generator (including the tubes) can degrade. The degree of susceptibility to degradation and the rate of degradation depend, in part, on the materials, water chemistry, and operating environment (e.g., temperature). To ensure that the materials are compatible with the environment, consistent with the requirements of **GDC 4**, the primary and secondary coolant water chemistry should be controlled. In addition, material damage or deterioration can occur during construction and operation as a result of improper cleaning or cleanliness control. This damage/ deterioration can result from chemical impurities or from particulate matter. As a result, it is important to establish measures to control the cleaning of material and equipment, consistent with the requirements of **Criterion XIII of Appendix B to 10 CFR 50**.	Y	5.4.2

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	The overall purpose of determining the compatibility of the material with the environment is to ensure that the inservice inspection program is sufficient to manage any degradation. The intention of this approach is ultimately to ensure that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low, (2) the RCPB design conditions are not exceeded during operation, and (3) sufficient margin is available to prevent rapidly propagating failure, consistent with the requirements of **GDC 14, 15, and 31**. The primary water chemistry program is reviewed under **SRP Sections 5.2.3 and 9.3.4**. In addition, **Regulatory Guide 1.44** discusses appropriate chemistry limits for the reactor coolant. The secondary water chemistry program is acceptable if (1) the coolant chemistry is maintained and monitored as described in the **Branch Technical Position, BTP 5-1**, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generator,” (2) the secondary water chemistry requirements in the latest revisions of the Standard Technical Specifications, NUREG-1430, NUREG-1431, and NUREG-1432 are incorporated into the facility’s Technical Specifications (the secondary water chemistry program in the Standard Technical Specifications meets the requirements of **10 CFR 50.36**), and (3) the chemical additives that limit the steam generator’s susceptibility to corrosion are such that any degradation to which the steam generator remains susceptible can be managed through the inservice inspection program. The operating environment (temperature, pressure, and flow) includes important variables that must be considered in evaluating the effectiveness of the chemical additives in limiting the steam generator’s susceptibility to corrosion.		

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	The onsite cleaning and cleanliness controls of the steam generator are acceptable if they meet the regulatory provisions of **Regulatory Guide 1.37**, consistent with the requirements of **Criterion XIII of Appendix B to 10 CFR Part 50**.		
5.4.2.1-SAC-06	<u>Provisions for Accessing the Secondary Side of the Steam Generator</u> Corrosion products (including deposits and sludge) and other contaminants can accumulate in the secondary side of the steam generator. For example, corrosion products and contaminants have been observed along the length of the steam generator tubes, in the crevice between the tube and the tube supports, and at the top of the tubesheet. Depending on the nature of these corrosion products and contaminants, degradation of the tubes (or other components) can occur. Because this degradation could lead to degradation of the pressure boundary, the design of the steam generator should provide access for the removal of these corrosion products and contaminants. These provisions will supplement the removal of corrosion products and contaminants by blowdown, which is reviewed under **SRP Section 10.4.8**. In addition to corrosion products and other contaminants, foreign objects (including loose parts) can be introduced into the steam generator. These objects can also lead to degradation of the pressure boundary; therefore, the design of the steam generator should provide access for removing these objects. The steam generator design is considered acceptable from a secondary-side access standpoint if it provides adequate access to the internals so that tools may be inserted to inspect and remove (1) corrosion products and contaminants (such as those found on the tubesheet and at the tube-to-tube support crevice) that may lead to corrosion and (2) foreign objects	Y	5.4.2

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	(including loose parts) that may affect tube integrity. These provisions, in conjunction with appropriately performed inservice inspections, as discussed above, provide assurance that (1) the probability of abnormal leakage, rapidly propagating failure, and gross rupture is extremely low and (2) the RCPB design conditions are not exceeded during operation, consistent with the requirements of **GDC 14 and 15**.		
SRP 5.4.2.2	**Steam Generator Program (R2, 03/2007)**		
5.4.2.2-AC-01	**General Design Criterion (GDC) 32** of Appendix A to 10 CFR Part 50. GDC 32 requires, in part, that the designs of all components that are part of the RCPB permit periodic inspection and testing of critical areas and features to assess their structural and leaktight integrity.	Y	5.4.2
5.4.2.2-AC-02	**10 CFR 50.55a(g)** requires that ISI programs meet the applicable inspection requirements in Section XI of the ASME Code. The steam generator program is a portion of the ISI program. In addition, **10 CFR 50.55a(b)(2)(iii)** specifically addresses steam generator tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in **Article IWB-2000 of Section XI of the ASME Code**, the Technical Specifications govern.	Y	5.4.2
5.4.2.2-AC-03	**10 CFR 50.36** applies to the steam generator program in the Technical Specifications.	Y	5.4.2
5.4.2.2-AC-04	**Appendix B to 10 CFR Part 50** applies to the implementation of the steam generator program. Of particular note are **Criteria IX, XI, and XVI**. Criterion IX requires, in part, that measures shall be established to ensure that special processes, including nondestructive testing, are	Y	5.4.2

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	controlled and accomplished by qualified personnel using qualified procedures. Criterion XI requires, in part, the establishment of a test program to ensure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in applicable design documents. Criterion XVI requires, in part, that measures shall be established to ensure the prompt identification and correction of conditions that are adverse to quality.		
5.4.2.2-AC-05	**10 CFR 50.65** requires that licensees monitor the performance or condition of SSCs against goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.	Y	5.4.2
5.4.2.2-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.4.2.2-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
5.4.2.2-SAC-01	Steam generator tubes are susceptible to degradation. This degradation can occur anywhere along the length of the tube. As a result, each tube is required to be accessible for inspection along its entire length and removable from service if unacceptable flaws are observed. The entire length of each tube must be inspectable using currently available nondestructive examination methods and techniques capable of finding the forms of degradation that may occur during the service life of the steam generators. The design of the steam generators should permit tubes with unacceptable flaws to be removed from service to ensure that tube integrity will be maintained. Tubes with unacceptable flaws should also be capable of being stabilized if it is determined that a plugged tube potentially may sever (as a result of continued degradation) and subsequently affect the integrity of an active tube. Access to both the primary and secondary sides of the steam generator tubes is required because conditions may exist on either side of the steam generator tubes that could affect their structural and leakage integrity. This should include, for example, access to the secondary face of the tubesheet, open tube lanes, feedwater inlet area (e.g., J-tubes or preheater inlet), and other locations that may impair tube integrity. Degradation of secondary side internals can result in the generation of loose parts, inadequate tube support, and mechanical damage to the tubes. In addition, the introduction of foreign objects (including loose parts) into the steam generator during fabrication, maintenance, or operation of the steam generators could impact tube integrity. Sludge buildup and deposits on the tubes can increase the susceptibility of the tubing to corrosion and make it more difficult to inspect the tubing (e.g., because of noise in eddy current data or obstructions in a visual	Y	5.4.2

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	inspection). As a result, the design of the steam generators is considered acceptable for the purposes of implementing the steam generator program if it (1) ensures that all steam generator tubes are accessible for periodic inspection, testing, and repair (including plugging and stabilizing), (2) permits an inspection of the full length of every tube, using currently available nondestructive examination methods and techniques, (3) allows access to the tubes from the primary side, (4) permits access to the secondary side of the steam generator for assessing the condition of SSCs that may affect tube integrity and for taking appropriate corrective action if adverse/anomalous conditions are identified, (4) permits inspection for, and removal of, foreign objects (including loose parts), and (5) allows the removal of each tube from service.		
5.4.2.2-SAC-02	A steam generator program is needed to ensure the effective monitoring and management of tube degradation and degradation precursors (so as to ensure steam generator tube integrity). This permits prompt preventive and corrective actions to ensure that the structural and leakage integrity of the steam generator tubes is maintained. The steam generator program should include elements such as an assessment of degradation, inspection requirements for the tubes and any repairs to the tubes (including plugging), integrity assessment procedures, tube plugging and repairs, primary-to-secondary leak monitoring, foreign material exclusion (including management of loose parts), maintenance of steam generator secondary side integrity, contractor oversight, self assessment, and reporting. For light water reactors (LWRs), **Nuclear Energy Institute (NEI) 97-06** discusses many of the elements of a steam generator program. The water chemistry portion of the steam generator program is	Y	5.4.2

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Reactor Coolant System and Connected Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	reviewed under **BTP 5-3**, “Monitoring of Secondary Side Water Chemistry in PWR Steam Generators.”		
5.4.2.2-SAC-03	The latest revisions of **NUREG-1430**, **NUREG-1431**, and **NUREG-1432** provide for the establishment and implementation of a steam generator program to ensure that tube integrity is maintained for the operating interval between tube inspections, consistent with the requirements of **GDC 32**. The Technical Specifications provide the objectives of the steam generator program, maximum limits on the quantity of primary-to secondary leakage permitted during operation, maximum time interval between inspections, objectives of the techniques used to inspect the tubes, tube repair criteria, and tube repair methods. “Notice of Availability of Model Application Concerning Technical Specification Improvement To Modify Requirements Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process,” and “Notice of Opportunity To Comment on Model Safety Evaluation on Technical Specification Improvement To Modify Requirements Regarding the Addition of LCO 3.4.[17] on Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process” include the staff’s evaluation of these Technical Specifications. The intention of implementing this program is to ensure tube integrity consistent with the original design criteria for the tubes. Certain aspects of the steam generator portion of the Standard Technical Specifications specify a plant-specific evaluation. For example, the tube repair criteria and tube repair methods are evaluated on a plant-specific basis.	Y	5.4.2 16.5
5.4.2.2-SAC-04	With respect to the steam generator tube repair criteria, **Regulatory Guide (RG) 1.121** describes a methodology acceptable to the NRC staff	Y	5.4.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	for determining the repair criteria specified in the Technical Specifications. Specifically, RG 1.121 describes a methodology for determining the minimum acceptable steam generator tube wall thickness. This methodology accounts for flaw growth and the uncertainty in measuring the size of the flaw (i.e., nondestructive examination uncertainty). The general principles of RG 1.121 can also be used to evaluate the acceptability of alternate tube repair criteria, that is, to assess tube repair criteria based on inspection parameters (e.g., flaw length) other than the depth of the flaw (i.e., other than a minimum wall thickness repair criterion). Tubes with flaws that exceed the repair criteria, as determined by the steam generator program, are removed from service consistent with the objective of the steam generator program to maintain tube integrity.		
5.4.2.2-SAC-05	With respect to tube repair methods, the review of these methods ensures that the repair is accessible for inspection and that techniques are available to find the forms of degradation to which the repair may be susceptible. The acceptability of any materials used in the repair is evaluated under **SRP Section 5.4.2.1**. The review of the acceptability of the mechanical design of the repair is consistent with the design requirements of the ASME Code and the steam generator performance criteria in the Standard Technical Specifications. The repair criteria for the repair method are reviewed under the guidance in **RG 1.121**.	Y	5.4.2
5.4.2.2-SAC-06	The latest revisions of **NUREG-1430**, **NUREG-1431**, and **NUREG-1432** address ISI; however, preservice inspections are essential in assessing the nature and significance of indications detected during ISI. As a result, it is important to inspect all tubes before placing the steam generators in service, using techniques that should be used during subsequent	Y	5.4.2

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	inspections (i.e., ISI). Although preservice inspections should use techniques that are expected to be employed during ISI, this expectation should not be construed to inhibit the use of new technology or to imply that the techniques used during the preservice inspection will always remain acceptable (i.e., different techniques may be appropriate based on operating experience).		
5.4.2.2-SAC-07	**10 CFR 50.55a(b)(2)(iii)** specifically addresses the inspection of steam generator tubes and states that if the plant Technical Specifications include inspection requirements that differ from those in **Article IWB-2000 of Section XI of the ASME Code**, the Technical Specifications govern. This requirement is intended to resolve any conflict between the requirements in the ASME Code and the Technical Specifications. If a conflict (i.e., difference) does not exist pertaining to a specific requirement, both the requirements of the ASME Code and the Technical Specifications must be met. In general, the requirements in the ASME Code and the Technical Specifications are complementary.	Y	5.4.2
5.4.2.2-SAC-08	For applicants referencing a certified design, the Standard Technical Specifications associated with the referenced design will specify the guidelines for periodic inspection and testing of the steam generator tube portion of the RCPB.	Y	5.4.2
5.4.2.2-SAC-09	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Steam Generator Program are reviewed in accordance with 10 CFR.55a(g) as it relates to periodic inspection and testing of the steam generator tubes as detailed in Section XI of the ASME Code. The implementation milestone is the establishment	Y N/A - COL	5.4.2

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and completion of an acceptable steam generator program per Article IWA-2430(b) of Section XI of the ASME Code before placing the plant into commercial service. The steam generator program is acceptable if it: A. Complies with 10 CFR 50.55a as it relates to periodic inspection and testing of the steam generator tubes as detailed in Section XI of the ASME Code. B. Complies with 10 CFR 50.65 as it relates to monitoring SSCs and establishing goals to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions. C. Incorporates the steam generator program requirements in the latest revisions of the Standard Technical Specifications, NUREG-1430, NUREG-1431, and NUREG-1432, into the facility's Technical Specifications (the steam generator program in the Standard Technical Specifications meets the requirements of 10 CFR 50.36). D. Verifies that all potential conflicts between the Technical Specifications and the ASME Code are identified. E. Verifies that the steam generator program includes the elements discussed above. F. Ensures that all tubes are inspected before being placed in service, using techniques that are expected to be used during subsequent inspections.		
SRP 5.4.6	**Reactor Core Isolation Cooling System (BWR) (R4, 03/2007)**	N/A-BWR	N/A
SRP 5.4.7	**Residual Heat Removal (RHR) System (R4, 03/2007)**		
5.4.7-AC-01	**GDC 2**, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the RHR system,	Y	5.4.7 6.8

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Reactor Coolant System and Connected Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	specifically based on meeting **Regulatory Position C-2 of Regulatory Guide 1.29** or its equivalent.		
5.4.7-AC-02	**GDC 4**, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)	Y	5.4.7 6.8
5.4.7-AC-03	**GDC 5**, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.	Y	5.4.7 6.8
5.4.7-AC-04	**GDC 19**, as it relates to control room requirements for normal operations and shutdown.	Y	5.4.7 6.8
5.4.7-AC-05	**GDC 34**, as it relates to requirements for an RHR system.	Y	5.4.7
5.4.7-AC-06	NUREG-0737 Task Action Plan item III.D.1.1, equivalent to **10 CFR 50.34(f)(2)(xxvi)** for applicants subject to 10 CFR 50.34(f), as it relates to the provisions for a leakage detection and control program to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.	Y	5.4.7
5.4.7-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
5.4.7-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to	N/A-COL	N/A

CHAPTER 5 **Reactor Coolant System and Connected Systems**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
5.4.7-SAC-01	The system or systems must satisfy the functional, isolation, pressure relief, pump protection, and test requirements specified in **Branch Technical Position BTP 5-4**.	Y	5.4.7 6.8
5.4.7-SAC-02	To meet the requirements of **GDC 4**, design features and operating procedures should be provided to prevent damaging water hammer caused by such mechanisms as voided lines.	Y	5.4.7 6.8
5.4.7-SAC-03	Interfaces between the RHR system and the RCIC and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to **GDC 5**.	Y	5.4.7
5.4.7-SAC-04	When the RHR system is used to control or mitigate the consequences of an accident, it must meet the design requirements of an engineered safety feature system. This includes meeting the guidelines of **Regulatory Guide 1.82** regarding water sources for long term recirculation cooling following a loss-of-coolant accident.	Y	5.4.7 6.8
SRP 5.4.8	**Reactor Water Cleanup System (BWR) (R3, 03/2007)**	N/A-BWR	N/A
SRP 5.4.11	**Pressurizer Relief Tank (R3, 03/2007)**		

CHAPTER 5
Reactor Coolant System and Connected Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
5.4.11-AC-01	**General Design Criterion (GDC) 2**, as it relates to the protection of essential systems from the effects of earthquakes. Acceptance is based on meeting the guidelines in **Position C.2 of Regulatory Guide 1.29** regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) **and** in **Position C.3** regarding the extension of seismic Category I boundaries.	Y	5.4.11
5.4.11-AC-02	**GDC 4**, as it relates to a failure of the system that results in missiles or adverse environmental conditions that could produce unacceptable damage to safety-related systems or components.	Y	5.4.11
5.4.11-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.4.11-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
5.4.11-SAC-01	Acceptance as it relates to the protection of essential systems from the effects of earthquakes is based on meeting the guidelines in **Position C.2 of Regulatory Guide 1.29** regarding the location of the tank in relation to other plant systems (the design of the tank system should be such that the plant safety-related systems would continue to perform their safety functions in the event of a tank failure) **and** in **Position C.3** regarding the extension of seismic Category I boundaries.	Y	5.4.11
5.4.11-SAC-02	The staff uses the following specific criteria to determine whether the requirements of **GDC 4** are met: A. The rupture disks have a relief capacity that at least equals the combined capacity of the pressurizer relief and safety valves, with sufficient allowance for rupture disk tolerance. B. The pressurizer relief tank volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or AOOs. It should be assumed that the initial temperature of water inside the tank will be no lower than 49 °C (120 °F). Systems performing similar functions should also be shown to have no release to containment during normal operations and AOOs. C. The design of the pressurizer relief tank and rupture disk should accommodate full vacuum so that the tank will not collapse if the contents are cooled after a discharge of steam without the addition of nitrogen. D. Alarms for high temperature, high pressure, and high and low liquid levels for the pressurizer relief tank have been provided. Systems performing similar functions should also have appropriate	Y	5.4.11

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	instrumentation to inform the operator about the condition of the systems. E. The location of the tank should be such that the rupture discs do not pose a missile threat to safety-related equipment.		
SRP 5.4.12	**Reactor Coolant System High Point Vents (R1, 03/2007)**		
5.4.12-AC-01	**10 CFR 50.46a**, as it relates to the provision of, and requirements related to, high-point vents for the RCS, the reactor vessel head, and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems.	Y	5.4.12
5.4.12-AC-02	**10 CFR 50.46(b)**, as it relates to the long-term cooling of the core following any calculated successful initial operation of the emergency core cooling system (ECCS) to remove decay heat for an extended period of time.	Y	5.4.12
5.4.12-AC-03	**10 CFR 50.49**, as it relates to environmental qualification of electrical equipment necessary to operate the reactor coolant vent system.	Y	5.4.12
5.4.12-AC-04	**10 CFR 50.55a** and **General Design Criteria (GDC) 1 and 30** found in Appendix A to 10 CFR Part 50, as they relate to the vent system components that are part of the reactor coolant pressure boundary (RCPB) being designed, fabricated, erected, and tested and maintained to high quality standards.	Y	5.4.12
5.4.12-AC-05	**GDC 14**, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y	5.4.12
5.4.12-AC-06	**GDC 17 and 34**, as they relate to the provision of normal and emergency	Y	5.4.12

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	power for the vent system components.		
5.4.12-AC-07	**GDC 19**, as it relates to the vent system controls being operable from the control room.	Y	5.4.12
5.4.12-AC-08	**GDC 36**, as it relates to the vent system being designed to permit periodic inspection.	Y	5.4.12
5.4.12-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
5.4.12-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
5.4.12-SAC-01	The reactor coolant vent design must ensure that use of these vents during and following an accident does not aggravate the challenge to containment or the course of the accident.	Y	5.4.12
5.4.12-SAC-02	Vent capability should be provided on high points of the RCS (including the pressurizer on PWRs and the hot legs on Babcock and Wilcox	Y	5.4.12

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	designs) to vent gases which may inhibit core cooling. For reactors with U-tube steam generators, procedures should be developed to remove sufficient gas from the U-tubes to ensure continued core cooling, since it is impractical to individually vent the thousands of U-tubes. In general, vent paths are not required for local high points at locations where gas accumulation would not be expected to jeopardize core cooling such as a reactor coolant pump valve body.		
5.4.12-SAC-03	A single failure of a vent valve, power supply, or control system should not prevent isolation of the vent path. On boiling water reactors, block valves are not required in lines with safety valves used for venting.	Y	5.4.12
5.4.12-SAC-04	The design should incorporate sufficient redundancy to minimize the probability of inadvertent actuation. Other methods to reduce the chances of inadvertent actuation, such as removing power or administrative controls, may be considered.	Y	5.4.12
5.4.12-SAC-05	Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met.	Y	5.4.12
5.4.12-SAC-06	The size of the vent should be smaller than the size corresponding to the definition of a LOCA (**Appendix A to 10 CFR Part 50**, **10 CFR 52.47(a)(1)(ii)**, and **10 CFR 52.79(b)**) to avoid unnecessary challenges to the ECCS, unless the applicant provides justification for a larger size.	Y	5.4.12
5.4.12-SAC-07	Vent paths to the containment should discharge into areas that provide good mixing with containment air and are able to withstand steam, water, noncondensibles, and mixtures of the above.	Y	5.4.12
5.4.12-SAC-08	The vent system should be operable from the control room and provide positive valve position indication. Power should be supplied from	Y	5.4.12

CHAPTER 5 **Reactor Coolant System and Connected Systems**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	emergency buses.		
5.4.12-SAC-09	It is important that the control room displays and controls for the RCS vents do not increase the potential for operator error. A human-factor analysis should be performed that considers the following: A. The use of this information by an operator during both normal and abnormal plant conditions B. Integration into emergency procedures C. Integration into operator training D. Other alarms during an emergency and need for prioritization of alarms	Y	5.4.12
5.4.12-SAC-10	The design should have provisions for testing the operability of the reactor coolant vent system. Testing should be performed in accordance with **Subsection IWV of Section XI of the ASME Code** for Category B valves.	Y	5.4.12.4 3.9.6 5.2.4 6.6
5.4.12-SAC-11	The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) should be seismically and environmentally qualified in accordance with **IEEE 344**, as supplemented by **Regulatory Guide 1.100** and **Regulatory Guide 1.92**. Environmental qualifications must be in accordance with **10 CFR 50.49**.	Y	5.4.12
5.4.12-SAC-12	The reactor coolant vent system should be designed to withstand the dynamic loads that will be encountered during operation from high RCS pressure to the approximate atmospheric pressure at the vent system exhaust.	Y	5.4.12
5.4.12-SAC-13	Procedures to effectively operate the vent system must consider when venting is needed and when it is not needed. A variety of initial conditions	Y	5.4.12

CHAPTER 5 Reactor Coolant System and Connected Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	for which venting may be required should be considered. Operator actions and the necessary instrumentation should be identified.		
5.4.12-SAC-14	The reactor coolant vent system should meet the quality assurance acceptance criteria provided in **SRP Chapter 17**.	Y	5.4.12
SRP 5.4.13	**Isolation Condenser System (BWR) (03/2007)**	N/A-BWR	N/A
BTP 5-1	**Monitoring Secondary Side Water Chemistry In PWR Steam Generators (R3, 03/2007)**	Refer to SRP 5.4.2.1, 5.4.2.1-SAC-05	
BTP 5-2	**Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures (R3, 03/2007)**	Refer to SRP 5.2.2, 5.2.2-SAC-04	
BTP 5-3	**Fracture Toughness Requirements (R2, 03/2007)**	Refer to SRP 5.3.2, 5.3.2-SAC-01 & SRP 5.4.2.2, 5.4.2.2-SAC-02	
BTP 5-4	**Design Requirements of the Residual Heat Removal System (R4, 03/2007)**	Refer to SRP 5.4.7, 5.4.7-SAC-01; SRP 10.3, 10.3-SAC-04; & SRP 10.4.9, 10.4.9-SAC-04	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 6.1.1	**Engineered Safety Features Materials (R2, 03/2007)**		
6.1.1-AC-01	**GDC 1**, and **10 CFR 50.55a** as they relate to quality standards for design, fabrication, erection, and testing of ESF components and the identification of applicable codes and standards.	Y	6.1.1
6.1.1-AC-02	**GDC 4** as it relates to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.	Y	6.1.1
6.1.1-AC-03	**GDC 14** as it relates to design, fabrication, erection, and testing of the RCPB so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y	6.1.1
6.1.1-AC-04	**GDC 31** as it relates to designing the RCPB such that the boundary behaves in a nonbrittle manner and there is an extremely low probability of rapidly propagating fracture and of gross rupture of the RCPB.	Y	6.1.1
6.1.1-AC-05	**GDC 35** as it relates to providing adequate core cooling following a LOCA at such a rate that fuel and clad damage that could inhibit core cooling is prevented and that the clad metal-water reaction is limited to negligible amounts.	Y	6.1.1
6.1.1-AC-06	**GDC 41** as it relates to control of the concentration of hydrogen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.	Y	6.1.1
6.1.1-AC-07	**Appendix B to 10 CFR Part 50, Criteria IX and XIII**, as they relate to establishing and controlling work and inspection instructions that prescribe the special cleaning processes and measures necessary to prevent material and equipment damage or deterioration in accordance with applicable codes, standards, specifications, criteria, and other special requirements.	Y	6.1.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.1.1-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.1.1-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.1.1-SAC-01	Materials and Fabrication. To meet the requirements of **GDC 1** and **10 CFR 50.55a** to assure that SSCs important to safety shall are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, codes and standards should be identified and records maintained. The materials specified for use in these systems must be as given in **Parts A, B and C of Section II of the ASME Code** and **Appendix I to Section III, Division 1 of the Code**. **Regulatory Guide (RG) 1.84** describes acceptable Code Cases that may be used in conjunction with the above specifications. Fracture toughness of the materials should be as stated in **SRP Section 10.3.6**, “Steam and Feedwater System Materials,” subsection II.1.	Y	6.1.1
	A. Austenitic Stainless Steels. To meet the requirements of **GDC 4** relative to compatibility of components with environmental conditions; **GDC 14** with respect to fabrication and testing of the RCBP such that there is an extremely low probability of	Y	6.1.1

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	abnormal leakage, rapidly propagating failure, and gross rupture; and the quality assurance requirements of **Appendix B of 10 CFR Part 50**, the following guidelines should be used: i. **RG 1.44** describes acceptable criteria for preventing intergranular corrosion of stainless steel components of the ESF. Furnace-sensitized material should not be allowed in the ESF, and methods described in this guide should be followed for testing the materials prior to fabrication, and for ensuring that no deleterious sensitization occurs during welding. ii. **RG 1.31** describes acceptable criteria for assuring the integrity of welds in austenitic stainless steel ESF components. The control of delta ferrite content of weld filler metal is specified in this guide, which sets forth an acceptable basis for delta ferrite content of weld filler metal. iii. The controls for abrasive work on austenitic stainless steel surfaces should, at a minimum, be equivalent to the controls described in **RG 1.37, position C.5** to prevent contamination, which promotes stress corrosion cracking. Tools that contain materials that could contribute to intergranular or stress-corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces. iv. Criteria to assure adequate resistance to intergranular stress corrosion cracking (IGSCC) for susceptible boiling water reactors (BWR) austenitic stainless steel ESF piping are described in NUREG-0313 and in Attachment A to Generic Letter (GL) 88-01. The technical bases for the positions provided in GL 88-01 are detailed in NUREG-0313. These criteria are applied to piping specified in GL 88-01. GL 88-01 and NUREG-0313 criteria used for the evaluation of initial material selection and fabrication include welding controls (e.g., delta ferrite content limits) and material specifications (e.g., carbon content specifications) that are more stringent than specified in RGs 1.31 and 1.44 and should supplant the regulatory		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	guides to assure adequate resistance of susceptible piping to IGSCC.		
	B. Ferritic Steel Welding. To meet the requirements of **GDC 1** related to general quality assurance and codes and standards; **Appendix B to 10 CFR Part 50**, related to control of special processes; and **10 CFR 50.55a**, the following acceptance criteria for ferritic steel welding should be used: i. The amount of minimum specified preheat must be in accordance with the recommendations of the **Code, Section III, Appendix D, Article D-1000**, and **RG 1.50**, unless an alternate procedure is justified. ii. Moisture control on low hydrogen welding materials shall conform to the requirements of the **Code, Section III, Articles NB, NC, ND-2000 and 4000**, and **AWS D1.1**, unless alternate procedures are justified. iii. For areas of limited accessibility, the criteria of **Regulatory Guide 1.71** apply a discussed in **SRP Section 10.3.6**.	Y	6.1.1
6.1.1-SAC-02	Composition and Compatibility of ESF Fluids. In meeting the requirements of **GDC 4 and 41** that SSCs important to safety are designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including loss-of-coolant accidents, and to assure that the concentration of hydrogen in the containment atmosphere following postulated accidents is controlled to maintain containment integrity, hydrogen generation resulting from the corrosion of metals by containment sprays during a design-basis accident should be controlled as described in **RG 1.7, position C.6**.	Y	6.1.1
	A. Pressurized Water Reactors (PWRs). To meet the requirements of **GDC 4, 14, and 41**, the composition of containment	Y	6.1.1

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	spray and core cooling water should be controlled to ensure a minimum pH of 7.0, as addressed in **Branch Technical Position (BTP) 6-1**, "pH for Emergency Coolant Water for PWRs." Experience has shown that maintaining the pH of borated solutions at this level will help to inhibit initiation of stress corrosion cracking of austenitic stainless steel components. Hydrogen generation from the corrosion of materials within containment, such as aluminum and zinc, depends upon the corrosion rate, which in turn depends upon such factors as the coolant chemistry, the coolant pH, the metal and coolant temperature, and the surface area exposed to attack by the coolant. The assumed corrosion rates of materials in containment should be consistent with standard corrosion rate data.		
	B. Boiling Water Reactors (BWRs). To meet the requirements of GDC 4, 14, and 41, the water used in the ESF systems should be controlled to provide assurance against stress corrosion cracking of unstabilized austenitic stainless steel components. Water used for emergency core cooling systems and spray systems should be controlled to ensure the following limits: Conductivity ≤0.5 mS/m (≤5 μmhos/cm) @ 25 °C Chloride (Cl-) < 0.20 ppm pH = 5.3 to 8.6 @ 25 °C Hydrogen generation in BWR containments is assumed to follow the same characteristics as in pressurized water reactors (PWRs) in that the rates of hydrogen generation will rise with increasing zinc corrosion as the temperature rises, and will change with any change in pH.	N/A-BWR	N/A

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6.1.1-SAC-03	<u>Component and Systems Cleaning</u>. To meet the requirements of **Appendix B to 10 CFR Part 50, Criteria IX and XIII**, measures should be established to control the cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.	Y	6.1.1
6.1.1-SAC-04	<u>Thermal Insulation</u>. To meet the requirements of **GDC 1, 14, and 31**, the RCPB should be designed, fabricated, erected, and tested in conformance with the following guidelines, such that there is an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture:	Y	6.1.1
	A. The composition of nonmetallic thermal insulation on ESF components should be controlled as described in **RG 1.36**.		
	B. The use of nonmetallic insulation on nonaustenitic stainless steel components should be controlled as described in **RG 1.36**. Moisture dripping from wet insulation can affect austenitic stainless steel components at lower elevations.		
	C. Concentrations of leachable contaminants and added inhibitors should be controlled as specified in **position C.2.b and Figure 1 of RG 1.36** to reduce the probability of stress corrosion cracking of austenitic stainless steel components.		
SRP 6.1.2	**Protective Coating Systems (Paints) - Organic Materials (R3, 03/2007)**		
6.1.2-AC-01	**Appendix B to 10 CFR Part 50** as it relates to the quality assurance requirements for the design, fabrication and construction of safety-related structures, systems and components (SSCs).	Y	6.1.2.4
6.1.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are	ITAAC	Tier 1

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	performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.1.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
	As discussed in **Regulatory Guide 1.54** Rev. 1, to the extent that failure of protective coatings could prevent safety related SSCs from fulfilling their safety related function, the maintenance rule, **10 CFR 50.65**, requires that licensees monitor the effectiveness of maintenance for protective coatings, or demonstrate that their performance or condition is being effectively controlled through the performance of appropriate preventative maintenance. Acceptance criteria include verification that coating monitoring and maintenance procedures are capable of ensuring that the coatings will not fail (delaminate from the substrate) and therefore become a debris source that could prevent the Emergency Core Cooling System (ECCS) from performing its safety related function.	N/A-COL	6.1.2.4
6.1.2-SAC-01	A coating system to be applied inside a containment is acceptable if it meets the regulatory positions of **Regulatory Guide 1.54** and the standards of **ASTM D5144-00** and **ASTM D3911-03**.	Y	6.1.2.2
SRP 6.2.1	**Containment Functional Design (R3, 03/2007)**	N/A-INFO (See SRP 6.2.1.1.A through SRP 6.2.1.5)	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 6.2.1.1.A	**PWR Dry Containments, Including Subatmospheric Containments (R3, 03/2007)**		
6.2.1.1.A-AC-01	**General Design Criterion (GDC) 16**, as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Since the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environs, preserving containment integrity under the dynamic conditions imposed by postulated loss of coolant accidents is essential.	Y	6.2.1.1
6.2.1.1.A-AC-02	**GDC 50**, as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate and with sufficient margin.	Y	6.2.1.1
6.2.1.1.A-AC-03	**GDC 38**, as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.	Y	6.2.1.1
6.2.1.1.A-AC-04	**GDC 13**, as it relates to instrumentation and control, requires instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety.	Y	6.2.1.1
6.2.1.1.A-AC-05	**GDC 64**, as it relates to monitoring radioactivity releases, requires means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.	Y	6.2.1.1
6.2.1.1.A-AC-06	For those applicants subject to 10 CFR 50.34(f): **10 CFR 50.34(f)(3)(v)(A)(1)**, as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel clad metal-water reaction accompanied by hydrogen	Y	3.8.1.4.11 6.2.1.1 19.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	burning.		
6.2.1.1.A-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.1.1.A-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.1.1.A-SAC-01	To satisfy the requirements of **GDC 16 and 50** regarding sufficient design margin, for plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit or design certification stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin.	Y	6.2.1.1
6.2.1.1.A-SAC-02	To satisfy the requirements of **GDC 38** to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated	Y	6.2.1.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50% of the peak calculated pressure within 24 hours, the organization responsible for SRP Section 15.0.3 should be notified.		
6.2.1.1.A-SAC-03	To satisfy the requirement of **GDC 38** to rapidly reduce the containment pressure, the containment pressure for subatmospheric containments should be reduced to below atmospheric pressure within one hour after the postulated accident, and the subatmospheric condition maintained for at least 30 days.	N/A-OTHER (Not a subatmospheric containment)	N/A
6.2.1.1.A-SAC-04	To satisfy the requirements of **GDC 38 and 50** with respect to the containment heat removal capability and design margin, the loss-of-coolant accident analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.	Y	6.2.1.1
6.2.1.1.A-SAC-05	To satisfy the requirements of **GDC 38 and 50** with respect to the containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single active failure in the containment heat removal systems (e.g., a fan, pump, or valve failure) or the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in **NUREG-0588**.	Y	6.2.1.1
6.2.1.1.A-SAC-06	To satisfy the requirements of **GDC 38 and 50** with respect to the functional capability of	N/A-OTHER	N/A

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	the containment heat removal systems and containment structure under loss-of-coolant accident conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.	(No containment heat removal systems with potential to inadvertently result in an external pressure)	
6.2.1.1.A-SAC-07	In accordance with the requirements of **GDC 13 and 64**, and **10 CFR 50.34(f)(2)(xvii)** (for those applicants subject to 10 CFR 50.34(f)), instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water level and temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. See Item II.F.1 of **NUREG-0737** and **NUREG-0718**, and **Branch Technical Position 7-10**, “Guidance on Application of Regulatory Guide 1.97.”	Y	6.2.1.1
6.2.1.1.A-SAC-08	In accordance with **10 CFR 50.46 Appendix K, I.D.2**, the minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See **SRP Section 6.2.1.5**, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").	Y	6.2.1.1
6.2.1.1.A-SAC-09	In accordance with **GDC 4**, containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to	Y	6.2.1.1

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	withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See **SRP Section 6.2.1.2**, "Subcompartment Analysis").		
6.2.1.1.A-SAC-10	In meeting the requirements of **10 CFR 50.34(f)(3)(v)(A)(1)**, applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for service level C limits (considering pressure and dead load only), for either concrete or steel containments, from **ASME Boiler Pressure Vessel Code, Section III**, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.	Y	3.8.1.4.11 6.2.1.1 19.2
SRP 6.2.1.1.B	**Ice Condenser Containments (Draft R3, 04/1996)**	N/A-ICE	N/A
SRP 6.2.1.1.C	**Pressure-Suppression Type BWR Containments (R7, 03/2007)**	N/A-BWR	N/A
SRP 6.2.1.2	**Subcompartment Analysis (R3, 03/2007)**		
6.2.1.2-AC-01	**General Design Criterion (GDC) 4**, as it relates to the design of containment internal compartments to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. The containment internal compartments shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.	Y	6.2.1.2
6.2.1.2-AC-02	**GDC 50**, as it relates to the design of the containment internal compartments to ensure that the reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from	Y	6.2.1.2

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	any loss-of-coolant accident.		
6.2.1.2-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.1.2-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.1.2-SAC-01	<u>Nodalization Schemes</u>. Subcompartment nodalization schemes should be chosen so that there is no substantial pressure gradient within a node. A sensitivity study which includes increasing the number of nodes until the peak calculated pressures converge to small resultant changes should be used to verify the nodalization scheme. The guidelines of Section 3.2 of **NUREG-0609** (Ref. 1) should be followed and a nodalization sensitivity study should be performed, which should include the consideration of spatial pressure variations (e.g., pressure variations circumferentially, axially, and radially within the subcompartment). These variations are use to calculate the transient forces and moments acting on components.	Y	6.2.1.2
6.2.1.2-SAC-02	<u>Initial Thermodynamic Conditions</u>. The initial atmospheric conditions within a subcompartment should maximize the resultant differential pressure. An acceptable model would assume air at the maximum allowable	Y	6.2.1.2

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	temperature, minimum absolute pressure, and zero percent relative humidity. If the assumed initial atmospheric conditions differ from this model, the selected values should be justified by the applicant. Another acceptable model that may be used for a restricted class of subcompartments involves simplifying the air model outlined above. In this case, the initial atmosphere within the subcompartment is modeled as a homogeneous water-steam mixture with an average density equivalent to the dry air model. This approach should be limited to subcompartments that have choked flow within the vents because the adequacy of this simplified model for subcompartments having primarily subsonic flow through the vents has not been established.		
6.2.1.2-SAC-03	Vent Flow Path and Distribution of Mass and Energy Released. Assumptions with regard to the distribution of mass and energy release should be biased towards maximizing the subcompartment pressure. The vent flow behavior through all flowpaths within the nodalized compartment model should be based on a homogeneous mixture in thermal equilibrium, with the assumption of 100-percent water entrainment. In addition, the selected vent critical flow correlation should be conservative with respect to available experimental data. Currently acceptable vent critical flow correlations are the “frictionless Moody” (Ref. 2), with a multiplier of 0.6 for water-steam mixtures, and the thermal homogeneous equilibrium model for air-steam-water mixtures.	Y	6.2.1.2
	If vent flowpaths are used that are not immediately available at the time of pipe rupture, the following criteria apply:		
	A. The vent area and resistance as a function of time after the break should be based on a dynamic analysis of the subcompartment pressure response to pipe ruptures.		
	B. The validity of the analysis should be supported by experimental data, or a testing program should be proposed at the construction permit or DC stage that will support		

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	this analysis. C. To meet the requirements of **GDC 4**, the safety analysis should consider the effects of missiles that may be generated during the transient.		
6.2.1.2-SAC-04	Design Pressure. For the review of a construction permit (CP) preliminary safety analysis report (PSAR) or a factor of 1.4 should be applied to the peak differential pressure which is calculated in a manner acceptable to the reviewer for the subcompartment structure, and the enclosed components for use in the design of the structure and the component supports. For the review of the operating license (OL) final safety analysis report (FSAR), DC or COL FSAR, the peak calculated differential pressure should not exceed the design pressure. It is expected that the peak calculated differential pressure will not be substantially different from that of the construction permit. However, improvements in the analytical models or changes in the as-built subcompartment may affect the available margin.	Y	6.2.1.2
SRP 6.2.1.3	**Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs) (R3, 03/2007)**		
6.2.1.3-AC-01	**General Design Criterion 50**, as it relates to the containment and subcompartments being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any LOCA.	Y	6.2.1.3
6.2.1.3-AC-02	**10 CFR Part 50, Appendix K**, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.	Y	6.2.1.3
6.2.1.3-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are	ITAAC	Tier 1

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	performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.2.1.3-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.1.3-SAC-01	**General Design Criterion 50** and **Appendix K to 10 CFR Part 50**		
	A. Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system (PWR plants only), including the steam generator tubing and secondary water. Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of **10 CFR Part 50, Appendix K, paragraph I.A**. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the reactor coolant system (RCS) and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion. The requirements of **paragraph I.B in Appendix K to 10 CFR Part 50**, concerning	Y	6.2.1.3

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	the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.		
	B. Break Size and Location	Y	6.2.1.3
	i. The staff's review of the applicant's choice of break locations and types is discussed in **SRP Section 3.6.2**		
	ii. Of several breaks postulated on the basis of a., above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.		
	iii. Containment design basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.		
	C. Calculations. In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist.	Y	6.2.1.3
	i. Subcompartment Analysis The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume nodding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include SATAN-V CRAFT-2, CE FLASH-4, and RELAP4, when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is		

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	also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.		
	ii. <u>Initial Blowdown Phase Containment Design Basis</u> The initial mass of water in the reactor coolant system should be based on the reactor coolant system volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data. Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection Calculations of heat transfer from the secondary coolant to the steam generator tubes for PWRs should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.		
	iii. <u>PWR Core Reflood Phase (Cold Leg Breaks Only)</u> Following initial blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the reactor coolant system broken loop pressure equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when the emergency core cooling		

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	system (ECCS) refills the vessel lower plenum. An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time. Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure. Calculations of liquid entrainment, i.e., the carryout rate fraction, which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the PWR full length emergency cooling heat transfer experiments. Liquid entrainment should be assumed to continue until the water level in the core is 61 cm (2 feet) from the top of the core. An acceptable approach is to assume a carryout rate fraction (CRF) of 0.05 to the 46 cm (18-inch) core level, a linearly increasing CRF to 0.80 at the 61 cm (24-inch) level, and a constant CRF of 0.80 until the water level is 61 cm (2 feet) from the top of the core. Above this level, a CRF of 0.05 may be used. The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water. Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.		
	iv. PWR Post-Reflood Phase All remaining stored energy in the primary and secondary systems should be		

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	removed during the post-reflood phase. Steam quenching should be justified by comparison with applicable experimental data. The results of post-reflood analytical models should be compared to applicable experimental data.		
	v. <u>PWR Decay Heat Phase</u> The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in **SRP Section 9.2.5**. Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water.		
	The following methods and computer models are acceptable for calculating the mass and energy releases for containment design basis calculations: Babcock and Wilcox / Framatome ANP: CRAFT, CRAFT-2, RELAP5/MOD2-B&W, Revision 1 and RELAP5/MOD2-B&W, Revision 4. Combustion Engineering: CEFLASH-4A and CESSAR System 80. General Electric: M3CPT, NEDO-20533, and SHEX. Westinghouse: WCAP-8312, SATAN-V, WCAP-10325, SATAN-VI, and WREFLOOD. Theses codes and methods have been referenced in licensee submittals and on a case by case basis have been found to be acceptable for these purposes.		

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	Other methods will be acceptable if they are found to be conservative for these calculations.		
6.2.1.3-SAC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.1.3-SAC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations. **10 CFR 52.47(a)(1)(vi)** provides the requirement for ITAAC for design certification reviews.	N/A-COL	N/A
SRP 6.2.1.4	**Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures (R2, 03/2007)**		
6.2.1.4-AC-01	**General Design Criteria (GDC) 50**, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated pressurized-water reactor (PWR) secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.	Y	6.2.1.4

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6.2.1.4-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.1.4-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.1.4-SAC-01	<u>Sources of Energy</u>. The sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the affected steam generator's metal, including the vessel tubing, feedwater line, and steamline; stored energy in the water contained within the affected steam generator; stored energy in the feedwater transferred to the affected steam generator before closure of the isolation valves in the feedwater line; stored energy in the steam from the unaffected steam generator(s) before the closure of the isolation valves in the steam generator crossover lines; and energy transferred from the primary coolant to the water in the affected steam generator during blowdown. The steamline break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. The applicant need only analyze the 102-percent power condition if it can demonstrate that the feedwater flows and fluid inventory are greatest at full power.	Y	6.2.1.4

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6.2.1.4-SAC-02	<u>Mass and Energy Release Rate</u>. In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint (i.e., the postaccident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired:	Y	6.2.1.4
	A. Mass release rates should be calculated using the Moody model (Ref. 1) for saturated conditions or a model that is demonstrated to be equally conservative.		
	B. Calculations of heat transfer to the water in the affected steam generator should be based on nucleate boiling heat transfer.		
	C. Calculations of mass release should consider the water in the affected steam generator and feedwater line, feedwater transferred to the affected steam generator before the closure of the isolation valves in the feedwater lines, steam in the affected steam generator, and steam coming from the unaffected steam generator(s) as the secondary system is being depressurized before the closure of the isolation valves in the steam generator crossover lines.		
	D. If liquid entrainment is assumed in the steamline breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steamline breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case.		
	If no liquid entrainment is assumed, a spectrum of the steamline breaks should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered.		
	E. Feedwater flow to the affected steam generator should be calculated considering the		

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	diversion of flow from the other steam generators, feedwater flashing, and increased feedwater pump flow caused by the reduction in steam generator pressure. An acceptable method for computing feedwater flow is to assume all feedwater travels to the affected steam generator at the pump runout rate before isolation. After isolation, the unisolated feedwater mass should be added to the affected steam generator.		
	Operator action to terminate auxiliary feedwater flow will be reviewed under **SRP Section 10.4.9**. Any general-purpose thermal-hydraulics computer codes that the responsible reviewing organization for the subject application finds acceptable may be used to compute mass and energy releases from steam and feedwater line break accidents.		
6.2.1.4-SAC-03	Single-Failure Analyses. Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions or feedwater pumps to maximize the containment peak pressure and temperature. For the assumed failure of a safety-grade steam or feedwater line isolation valve, operation of non-safety-grade equipment may be relied upon as a backup to the safety-grade equipment. In this event, the reviewer will confer with the responsible organizations for **SRP Sections 3.2.1, 3.2.2, 3.6.2, and 10.4.9** to ensure a consistent staff position regarding the acceptability of the design criteria for the nonsafety-grade equipment.	Y	6.2.1.4 15.0.0.3.8
SRP 6.2.1.5	**Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies (R3, 03/2007)**	Y (Per AREVA Topical Report ANP-10278)	6.2.1.5
SRP 6.2.2	**Containment Heat Removal Systems (R5, 03/2007)**		
6.2.2-AC-01	**GDC 38** as it relates to the following:	Y	5.4.7

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	A. The ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain these indicators at acceptably low levels. B. The ability of the containment heat removal system to perform in a manner consistent with the function of other systems. C. The safety-grade design of the containment heat removal system (i.e., suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished in the event of a single failure).		6.2.2
6.2.2-AC-02	**GDC 39**, as it relates to the design of the containment heat removal system to permit periodic inspection of components.	Y	5.4.7 6.2.2
6.2.2-AC-03	**GDC 40**, as it relates to the design of the containment heat removal system to allow periodic testing to ensure system integrity and the operability of the system and active components	Y	5.4.7 6.2.2
6.2.2-AC-04	**10 CFR 50.46(b)(5)**, as it relates to requirements for long-term cooling, including adequate NPSH margin in the presence of LOCA-generated and latent debris.	Y	5.4.7 6.2.2
6.2.2-SAC-01	The containment heat removal systems should meet the redundancy and power source requirements for an engineered safety feature (i.e., the results of failure modes and effects analyses of each system should ensure that the system is capable of withstanding a single failure without loss of function). This conforms to the requirements of **GDC 38**.	Y	5.4.7 6.2.2
6.2.2-SAC-02	With regard to **GDC 38** as it relates to the capability of the containment system to accomplish its safety function, the spray system should be designed to accomplish this without pump damage caused by cavitation.	Y	5.4.7 6.2.2

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6.2.2-SAC-03	In evaluating the performance capability of the CSS to satisfy **GDC 38**, the analyses of its heat removal capability should be based on the following considerations: A. The locations of the spray headers relative to the internal structures. B. The arrangement of the spray nozzles on the spray headers and the expected spray pattern. The spray systems should be designed to ensure that the spray header and nozzle arrangements produce spray patterns which maximize the containment volume covered and minimize the overlapping of the sprays. C. The spray drop size spectrum and mean drop size emitted from each type of nozzle as a function of differential pressure across the nozzle. D. The effect of drop residence time and drop size on the heat removal effectiveness of the spray droplets.	Y	5.4.7 6.2.2
6.2.2-SAC-04	In evaluating the performance capability of the fan cooler system to satisfy **GDC 38**, the design heat removal capability (i.e., heat removal rate versus containment temperature) of the fan coolers should be established on the basis of qualification tests on production units or acceptable analyses that take into account the expected post accident environmental conditions and variations in major operating parameters, such as the containment atmosphere steam-air ratio, condensation on finned surfaces, and cooling water temperature and flow rate. The equipment housing and ducting associated with the fan cooler system should be analyzed to determine that the design is adequate to withstand the effects of containment pressure following a LOCA. Fan cooler system designs that contain components that do not have a post accident safety function should be designed so that failure of nonsafety-related equipment will not prevent the fan cooler system from accomplishing its safety function.	Y	5.4.7 6.2.2
6.2.2-SAC-05	In evaluating the heat removal capability of the containment heat removal system to satisfy **GDC 38**, the potential for surface fouling of the secondary sides of fan cooler, recirculation, and RHR heat exchangers by the cooling water over the life of the plant and	Y	5.4.7 6.2.2

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	the effect of surface fouling on the heat removal capacity of the heat exchangers. The application should discuss the results of the analysis. The results will be acceptable if they demonstrate that provisions such as closed cooling water systems are provided to prevent surface fouling or that surface fouling has been taken into account in the establishment of the heat removal capability of the heat exchangers.		
6.2.2-SAC-06	To satisfy the requirements of **GDC 38** and **10 CFR 50.46(b)(5)** regarding the long-term spray system(s) and ECCS(s), the containment emergency sump(s) in PWRs and suppression pools in BWRs should be designed to provide a reliable, long-term water source for ECCS and CSS pumps. The containment design should allow for the drainage of spray and emergency core cooling water to the emergency sump(s) or suppression pool and for recirculation of this water through the containment sprays and ECCSs. The design of the sumps or suppression pools and the protective strainer assemblies is a critical element in ensuring long-term recirculation cooling capability. Therefore, adequate design consideration of (1) sump and suppression pool hydraulic performance, (2) evaluation of potential debris generation and associated effects including debris screen blockage, (3) RHR and CSS pump performance under postulated post-LOCA conditions, and (4) impacts of debris penetrating strainers on long-term coolability of the core is necessary. Regulatory Guide 1.82, Revision 3, as modified and supplemented for PWRs by the Nuclear Energy Institute (NEI) Guidance Report (GR) (Ref. 3) and the NRC safety evaluation (SE) (Ref. 4), provide guidance for PWR debris evaluations. Regulatory Guide 1.82, Revision 3, as supplemented by the NRC-approved Boiling Water Reactor Owners' Group (BWROG) Utility Resolution Guidance (URG) (Ref. 5), provide guidance for BWR debris evaluations.	Y (for PWR) N/A-BWR	5.4.7 6.2.2 N/A
6.2.2-SAC-07	In meeting the requirements of **GDC 39 and 40** regarding inspection and testing, the design of the containment heat removal systems should provide for periodic inspection and operability testing of the systems and system components such as pumps, valves, duct pressure-relieving devices, and spray nozzles.	Y	5.4.7 6.2.2

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6.2.2-SAC-08	To satisfy the system design requirements of **GDC 38**, instrumentation should be provided to monitor the performance of the containment heat removal system and its components under normal and accident conditions. The instrumentation should determine whether a system is performing its intended function or whether a system train or component is malfunctioning and should be isolated.	Y	5.4.7 6.2.2
SRP 6.2.3	**Secondary Containment Functional Design (R3, 03/2007)**		
6.2.3-AC-01	**General Design Criterion (GDC) 4** of Appendix A to Part 50 of Title 10, Code of Federal Regulations (10 CFR Part 50 Appendix A) as to SSCs important to safety designed to accommodate the effects of environmental conditions of normal operation, maintenance, testing, and postulated accidents with protection against dynamic effects (e.g., effects of missiles, pipe whipping, and discharging fluids) that may result from equipment failures.	Y	6.2.3
6.2.3-AC-02	**GDC 16** as to reactor containment and associated systems establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.	Y	6.2.3
6.2.3-AC-03	**GDC 43** as to reactor containment and associated systems designed to permit appropriate periodic pressure and functional testing to assure structural integrity and operability.	Y	6.2.3
6.2.3-AC-04	**10 CFR Part 50, Appendix J** as it relates to secondary containment leakage rate testing in accordance with the procedures specified in the technical specifications, or associated bases, so that bypass leakage paths are identified and associated bypass leakage rates are determined.	Y	6.2.3
6.2.3-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design	ITAAC	Tier 1

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	certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
6.2.3-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.3-SAC-01	In meeting **GDC 16** requirements for functional capability of the secondary containment, the analysis of pressure and temperature response of the secondary containment to a LOCA in the primary containment should follow these guidelines:		
	A. Heat transfer from the primary to the secondary containment should be considered. i. Heat transfer from the primary containment atmosphere to the primary containment structure should be calculated by conservative heat transfer coefficients like those in Branch **Technical Position (BTP) 6-2**. ii. Conductive heat transfer through the primary containment structure and convective heat transfer to the secondary containment atmosphere should be considered. iii. Radiant heat transfer to the secondary containment should be considered.	Y	6.2.3
	B. Adiabatic boundary conditions should be assumed for the surface of the secondary containment structure exposed to the outside environment.	Y	6.2.3
	C. The compressive effect of primary containment expansion on the secondary containment atmosphere should be considered.	Y	6.2.3
	D. Secondary containment in-leakage should be considered.	Y	6.2.3

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	E. No credit should be taken for secondary containment out-leakage.	Y	6.2.3
	F. For secondary containment response analyses loss of offsite power and the most severe single active failure in the emergency power system (e.g., a diesel generator failure), in the primary containment heat removal systems, in the core cooling systems, or in the secondary containment depressurization and filtration system should be assumed. Any delay due to system design in secondary containment depressurization and filtration system actuation should be considered.	Y	6.2.3
	G. Heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered.	Y	6.2.3
	H. Fan performance characteristics should be considered in evaluating secondary containment depressurization.	Y	6.2.3
6.2.3-SAC-02	To meet the **GDC 4** requirement to protect SSCs important to safety against dynamic effects, high-energy lines passing through the secondary containment should have guard pipes. Design criteria for guard pipes are in **SRP Section 3.6.2**. If there are no guard pipes, analyses should demonstrate that both primary containment and secondary containment structures are capable of withstanding the effects of a high-energy pipe rupture inside the secondary containment without loss of integrity.	Y	3.6.2 6.2.3
6.2.3-SAC-03	In meeting **GDC 16** requirements for the functional capability of the secondary containment, the following criteria apply:		
	A. The secondary containment depressurization and filtration systems should meet the guidelines of **Regulatory Guide (RG) 1.52** and be capable of maintaining a uniform negative pressure throughout the secondary containment as well as other areas served by the systems.	Y	6.2.3
	B. The negative pressure differential to be maintained in the secondary containment and other contiguous plant areas should be no less than 0.063 kPa (0.25 inches water	Y	6.2.3

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	gauge) compared to adjacent regions under all wind conditions up to the wind speed at which diffusion becomes sufficient to assure site boundary exposures less than those calculated for the design basis accident even if exfiltration occurs. If the leakage rate exceeds 100 percent of the volume per day, there should be a special exfiltration analysis.		
	C. All openings like personnel doors and equipment hatches should be under administrative control with readout position indicators and alarms in the main control room. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and confirmatory preoperational tests conducted.	N/A-COL	6.2.3
	D. Some plants may have only portions of the primary containment enclosed rather than a secondary containment structure or shield building completely enclosing the primary containment. These enclosures are areas into which the primary containment most likely would leak, and they may be equipped with air filtration.	N/A-OTHER (All portions of primary containment enclosed)	N/A
	E. The external design pressure of the secondary containment structure should provide an adequate margin above the maximum expected external pressure.	Y	6.2.3
6.2.3-SAC-04	In meeting **GDC 43** and **10 CFR Part 50, Appendix J**, requirements for secondary containment system testing the following criteria apply:		
	A. The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment should be specified. **BTP 6-3** provides guidance for detecting leakage paths to the environment which may bypass the secondary containment. The periodic leakage rate testing program for measuring the fraction of primary containment leakage that may directly bypass the secondary containment and other contiguous areas served by ventilation and filtration systems	Y	6.2.3

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	should be described. Individual tests should be according to procedures from technical specifications or their bases.		
	B. There should be provisions in the design of the secondary containment system for inspections and monitoring of the functional capability. Preoperational and periodic test programs determine the depressurization time, the secondary containment in-leakage rate, the uniformity of negative pressure throughout the secondary containment and other contiguous areas, and the potential for ex-filtration.	Y	6.2.3
SRP 6.2.4	**Containment Isolation System (R3, 03/2007)**		
6.2.4-AC-01	**General Design Criterion (GDC) 1**, as it relates to designing, fabricating, erecting, and testing safety-related SSCs to quality standards commensurate with the importance of the safety functions to be performed.	Y	6.2.4
6.2.4-AC-02	**GDC 2**, as it relates to designing safety-related SSCs to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform safety functions.	Y	6.2.4
6.2.4-AC-03	**GDC 4**, as it relates to designing safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and as it relates to the requirement that these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.	Y	6.2.4
6.2.4-AC-04	**GDC 16**, as it relates to reactor containment and associated systems, establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.	Y	6.2.4
6.2.4-AC-05	**GDC 54**, as it relates to the requirement that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to	Y	6.2.4

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	safety, and as it relates to designing such piping systems with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.		
6.2.4-AC-06	**GDCs 55 and 56**, as to isolation valves for lines penetrating (GDC 55) the primary containment boundary as parts of the reactor coolant pressure boundary or as direct connections to the containment atmosphere (GDC 56) as follows: A. One locked-closed isolation valve inside and one outside containment; or B. One automatic isolation valve inside and one locked-closed isolation valve outside containment; or C. One locked-closed isolation valve inside and one automatic isolation valve outside containment; or D. One automatic isolation valve inside and one outside containment.	Y	6.2.4
		EXCEPTION (IRWST suction lines utilize guard pipe in conjunction with single isolation valve)	6.2.4
6.2.4-AC-07	**GDC 57**, as it relates to the requirement that lines penetrating the primary containment boundary and neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment.	Y	6.2.4
6.2.4-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.4-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide	N/A-COL	N/A

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.2.4-AC-10	**10 CFR 52.47(a)(8)** and **10 CFR 52.79(a)(17)**, as they relate to demonstrating compliance with any technically relevant portions of the Three Mile Island (TMI)-related requirements set forth in **10 CFR 50.34(f)(2)(xiv)** and **10 CFR 50.34(f)(2)(xv)**, for DC and COL reviews, respectively	Y	6.2.4
6.2.4-AC-11	**10 CFR 50.63(a)(2)**, as it relates to ensuring that appropriate containment integrity is maintained in the event of a station blackout for a specified duration.	Y	6.2.4
6.2.4-SAC-01	**Regulatory Guide (RG) 1.11** describes acceptable containment isolation provisions for instrument lines. In addition, instrument lines closed both inside and outside containment are designed to withstand pressure and temperature conditions following a loss-of-coolant accident (LOCA) and dynamic effects are acceptable without isolation valves.	Y	6.2.4
6.2.4-SAC-02	Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems may include remote-manual valves, but should detect possible leakage from these lines outside containment.	Y	6.2.4
6.2.4-SAC-03	Containment isolation provisions for lines in systems needed for safe shutdown of the plant (e.g., liquid poison system, reactor core isolation cooling system, and isolation condenser system) may include remote-manual valves, but there should be provisions for detecting leakage from such lines outside containment	Y	6.2.4
6.2.4-SAC-04	Containment isolation provisions for lines in the systems of items 2 and 3 normally consist of one isolation valve inside and one outside containment. If it is not practical to locate a valve inside containment (for example, the valve may be under water as a result of an accident), both valves may be located outside containment. For this type of isolation valve	Y	6.2.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	arrangement, the valve nearer the containment and the piping between the containment and the valve should be enclosed in a leak-tight or controlled-leakage housing. If, in lieu of housing, the piping and valve are designed to preclude a breach of piping integrity, the design should comply with **SRP Section 3.6.2** requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.		
6.2.4-SAC-05	Containment isolation provisions for lines in engineered safety feature or engineered safety feature-related systems normally consist of two isolation valves in series. A single isolation valve is acceptable if system reliability can be shown to be greater, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category I and Group B quality standards, and have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak-tested unless system integrity can be shown to be maintained during normal plant operations. For this type of isolation valve arrangement the valve is located outside containment, and the piping between the containment and the valve should be enclosed in leak-tight or controlled-leakage housing. If, in lieu of housing, piping and valve are designed conservatively to preclude a breach of piping integrity, the design should comply with **SRP Section 3.6.2** requirements. Design of the valve or the piping compartment should provide the capability to detect and terminate leakage from the valve shaft or bonnet seals.	Y	6.2.4
6.2.4-SAC-06	Sealed-closed barriers may be used in place of automatic isolation valves. Sealed-closed barriers include blind flanges and sealed-closed isolation valves which may be closed manual valves, closed remote-manual valves, or closed automatic valves which remain closed after a LOCA. Sealed-closed isolation valves should be under administrative control so they cannot be opened inadvertently. Administrative control includes	Y	6.2.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	mechanical devices to seal or lock the valve closed or to prevent power supply to the valve operator.		
6.2.4-SAC-07	Relief valves may be used as isolation valves provided the relief setpoint is greater than 1.5 times the containment design pressure.	N/A-OTHER	6.2.4
6.2.4-SAC-08	**10 CFR 50.34(f)(2)(xiv)** requires that systems penetrating the containment be classified as either essential or nonessential. Reference 26 presents guidance on the classification of systems as essential and nonessential. Essential systems, like those described in items 2 and 3, may include remote-manual containment isolation valves, but there should be provisions for detecting leakage from the lines outside containment. 10 CFR 50.34(f)(2)(xiv) also requires that nonessential systems be isolated automatically by the containment isolation signal.	Y	6.2.4
6.2.4-SAC-09	Isolation valves outside containment should be located as close to it as practical, as required by **GDCs 55, 56, and 57**.	Y	6.2.4
6.2.4-SAC-10	To meet the requirements of **GDCs 55 and 56**, upon loss of actuating power, automatic isolation valves should take the position of greatest safety. The position of an isolation valve for normal and shutdown plant operating and post-accident conditions depends on the fluid system function. If a fluid system has no post-accident function, the isolation valves in the lines should be closed automatically. For engineered safety feature or engineered safety feature-related systems, isolation valves in the lines may remain open or be opened. In a power failure to the valve operator isolation valves should be in the "safe" position, normally the post-accident valve position. For lines equipped with motor-operated valves, a loss of actuating power leaves the affected valve in the "as-is" position, which may be the open position; however, redundant isolation barriers ensure that the isolation function for the line is satisfied. All power-operated isolation valves should have position indications in the main control room.	Y	6.2.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.2.4-SAC-11	To improve the reliability of the isolation function, addressed in **GDC 54**, **10 CFR 50.34(f)(2)(xiv)** requires reduction of the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum value compatible with normal operating conditions.	Y	6.2.4
6.2.4-SAC-12	There should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the **GDC 54** requirement for reliable isolation capability.	Y	6.2.4
6.2.4-SAC-13	To improve the reliability of the isolation function, addressed in **GDC 56**, system lines which provide open paths from the containment to the environs (e.g., purge and vent lines addressed in **10 CFR 50.34(f)(2)(xiv)**) should be equipped with radiation monitors capable of isolating these lines upon a high-radiation signal, which should not be considered a diverse containment isolation parameter.	Y	6.2.4
6.2.4-SAC-14	In meeting **GDC 54** requirements, the performance capability of the isolation function should reflect the safety importance of isolating system lines. Consequently, containment isolation valve closure times should be selected for rapid isolation of the containment following postulated accidents. Valve closure time for a power-operated valve to be in the fully-closed position after the actuator power has reached the operator assembly does not include the time to reach actuation signal setpoints or instrument delay times, which, with system design capabilities, should be considered for establishing valve closure times. For lines providing open paths from the containment to the environs (e.g., the containment purge and vent lines), isolation valve closure times of five seconds or less may be necessary. The closure times of these valves should be established to minimize the release of containment atmosphere to the environs, to mitigate the offsite radiological consequences, and to prevent degradation of emergency core cooling system effectiveness by reduced containment back-pressure. Analyses of the radiological consequences and the effect on the containment back-pressure of the release of containment atmosphere should justify the selected valve closure time. **Branch Technical**	Y	6.2.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Position (BTP) 6-4 presents additional guidance on the design and use of containment purge systems which may be used during the normal plant operating modes (i.e., startup, power operation, hot standby, and hot shutdown). Containment purge valves that do not satisfy the operability criteria of **Branch Technical Position 6-4** must be sealed closed as defined in **subsection II.6 of this SRP section** during operational conditions 1, 2, 3, and 4. Furthermore, closure of these valves must be verified at least every 31 days. These requirements should be incorporated into the technical specifications for plant operation.		
6.2.4-SAC-15	The use of a closed system inside containment as one of the isolation barriers is acceptable if the closed system design satisfies the following requirements: A. The system does not connect with either the reactor coolant system or the containment atmosphere. B. The system is protected against missiles and pipe whip. C. The system is designated seismic Category I. D. The system is classified Quality Group B. E. The system is designed to withstand temperatures equal to at least that of the containment design. F. The system is designed to withstand the external pressure from the containment structure acceptance test. G. The system is designed to withstand the LOCA transient and environment. As to the structural design of containment internal structures and piping systems, the protection against loss of function from missiles, pipe whip, and earthquakes is acceptable if 1) isolation barriers are located behind missile barriers; 2) pipe whip was considered in the design of pipe restraints and the location of piping penetrating the containment; and 3) the isolation barriers, including the piping between isolation valves, are designated	Y	6.2.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	seismic Category I, i.e., designed to withstand the effects of the safe-shutdown earthquake, as recommended by **Regulatory Guide 1.29**.		
6.2.4-SAC-16	To meet the requirements of **GDCs 1, 2, 4, and 54**, appropriate reliability and performance considerations should be included in the design of isolation barriers to reflect the safety importance of their integrity (i.e., containment capability) under accident conditions. The design criteria for components performing a containment isolation function, including the isolation barriers and the piping between them or the piping between the containment and the outermost isolation barrier, are acceptable if: A. Group B quality standards, as defined in **RG 1.26**, apply to the components, unless the service function dictates that Group A quality standards apply. B. The components are designated seismic Category I in accordance with **RG 1.29**.	Y	6.2.4
6.2.4-SAC-17	**GDC 54** requires reliable isolation capability; therefore, for remote-manual isolation valves, the design of the containment isolation system is acceptable if there are provisions to allow the operator in the main control room to know when to isolate fluid systems equipped with remote-manual isolation valves. Such provisions may include instruments to measure flow rate, sump water level, temperature, pressure, and radiation level.	Y	6.2.4
6.2.4-SAC-18	**GDC 54** specifies requirements for the containment isolation system; therefore, to satisfy GDC 54, the design of the containment isolation system should provide for operability testing of the containment isolation valves and leakage rate testing of the isolation barriers. The isolation valve testing program should be consistent with that proposed for other engineered safety features. **SRP Section 6.2.6** presents acceptance criteria for the leakage rate testing program for containment isolation barriers.	Y	6.2.4
6.2.4-SAC-19	**GDC 54** requires reliable isolation capability. To satisfy this requirement, the design of the containment isolation system should reduce the possibility of unintended isolation valve reopening following isolation. **10 CFR 50.34(f)(2)(xiv)** requires control systems for automatic containment isolation valves be designed for resetting the isolation signal	Y	6.2.4

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	without automatically reopening the valves. Reopening of containment isolation valves should require deliberate operator action and combined reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be valve by valve or line by line, provided that electrical independence and other single-failure criteria remain satisfied. Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method for meeting this design requirement.		
6.2.4-SAC-20	In meeting **10 CFR 50.34(f)(2)(xv)** purging requirements, the regulatory guidance of **BTP 6-4**, "Containment Purging During Normal Plant Operations," should be used to establish compliance with this regulation.	Y	6.2.4
6.2.4-SAC-21	**RG 1.155, "Station Blackout," Regulatory Position C.3.2.7**, provides guidance for meeting the requirements of the SBO rule, **10 CFR 50.63(a)(2)**, for containment isolation valves and valve position indication.	Y	6.2.4
6.2.4-SAC-22	**10 CFR Part 50, Appendix K**, provides guidance for the determination of the extent of fuel failure (source term) in the radiological calculations	Y	15.0.3
SRP 6.2.5	**Combustible Gas Control in Containment (R3, 03/2007)**		
6.2.5-AC-01	**10 CFR Part 50, § 50.44**, as it relates to BWR and PWR plants being designed to: - Accommodate hydrogen generation equivalent to a 100 percent fuel clad-coolant reaction; limit containment hydrogen concentration to no greater than 10 percent; - Have a capability for ensuring a mixed atmosphere during design bases and significant beyond-design-bases accidents (a significant beyond-design-basis accident is an accident comparable to a degraded core accident at an operating (as of October 16, 2003) light-water reactor in which a metal-water reaction occurs involving 100 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume)); and - Provide containment-wide hydrogen control (such as igniters or inerting), if necessary,	Y	6.2.5

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	for certain severe accidents. Post-accident conditions should be such that an uncontrolled hydrogen/oxygen recombination would not take place in the containment, or the plant should withstand the consequences of uncontrolled hydrogen/oxygen recombination without loss of safety function or containment structural integrity.		
6.2.5-AC-02	**GDC 5** as it relates to providing assurance that sharing of structures, systems, and components important to safety among nuclear power units will not significantly impair their ability to perform their safety functions.	Y	6.2.5
6.2.5-AC-03	**GDC 41** as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; systems being designed to suitable requirements, i.e., that there be suitable redundancy in components and features, and suitable interconnections to ensure that for either a loss of onsite or a loss of offsite power the system safety function can be accomplished, assuming a single failure; and systems being provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished.	Y	6.2.5
6.2.5-AC-04	**GDC 42** as it relates to the design of the systems to permit appropriate periodic inspection of components to ensure the integrity and capability of the systems.	Y	6.2.5
6.2.5-AC-05	**GDC 43** as it relates to the systems being designed to permit periodic testing to ensure system integrity, and the operability of the systems and active components.	Y	6.2.5
6.2.5-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the	ITAAC	Tier 1

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	provisions of the Atomic Energy Act, and the NRC's regulations.		
6.2.5-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.5-SAC-01	In meeting the requirements of **10 CFR Part 50, § 50.44**, and **GDC 41** to provide systems to control the concentration of hydrogen in the containment atmosphere, materials within the containment that would yield hydrogen gas due to corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.	Y	6.2.5
6.2.5-SAC-02	In meeting the requirements of **10 CFR Part 50, § 50.44**, and **GDC 41** to provide systems to control the concentration of hydrogen or oxygen in the containment atmosphere, the applicant should demonstrate by analysis, for non-inerted containments, that the design can safely accommodate hydrogen generated by an equivalent of a 100 percent fuel clad-coolant reaction, while limiting containment hydrogen concentration, with the hydrogen uniformly distributed, to less than 10 percent (by volume), and while maintaining containment structural integrity.	Y	6.2.5
6.2.5-SAC-03	In meeting the requirements of **10 CFR Part 50, § 50.44(c)(3)**, regarding equipment survivability, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment structural integrity should 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 percent fuel clad-coolant reaction including the environmental conditions created by activation of the combustible gas	Y	6.2.5

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CHAPTER 6 **Engineered Safety Features**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	control system.		
6.2.5-SAC-04	In meeting the requirements of **10 CFR Part 50, § 50.44**, to provide the capability for ensuring a mixed atmosphere in the containment during design bases and significant beyond-design-bases accidents, and of **GDC 41** to provide systems as necessary to ensure that containment integrity is maintained, this capability may be provided by an active, passive, or combination system. Active systems may consist of a fan, a fan cooler, or containment spray. For passive or combination systems that use convective mixing to mix the combustible gases, the containment internal structures should have design features which promote the free circulation of the atmosphere. For all containment types, an analysis of the effectiveness of the method used for providing a mixed atmosphere should be provided. This analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity. Atmosphere mixing systems prevent local accumulation of combustible or detonable gases which could threaten containment integrity or equipment operating in a local compartment. Active systems installed to mitigate this threat should be reliable, redundant, single-failure proof, able to be tested and inspected, and remain operable with a loss of onsite or offsite power.	Y	6.2.5
6.2.5-SAC-05	In meeting the requirements of 1**0 CFR Part 50, § 50.44**, and **GDC 41** regarding the functional capability of the combustible gas control systems to ensure that containment integrity is maintained, the design should meet the provisions of **RG 1.7, Revision 3, Section C.1**.	Y	6.2.5
6.2.5-SAC-06	To satisfy the design requirements of **GDC 41**:		
	A. Performance tests should be performed on system components, such as hydrogen igniters and combustible gas monitors. The tests should support the analyses of the functional capability of the equipment.	Y	6.2.5

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	B. Combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal and accident conditions. The instrumentation should be capable of determining that a system is performing its intended function, or that a system train or component is malfunctioning and should be isolated. The instrumentation should have readout and alarm capability in the control room. The containment hydrogen and oxygen monitors should meet the provisions of **RG 1.7, Revision 3, Section C.2**.	Y	6.2.5
6.2.5-SAC-07	To satisfy the inspection and test requirements of **GDC 41, 42, and 43**, combustible gas control systems should be designed with provisions for periodic inservice inspection, operability testing, and leak rate testing of the systems or components.	Y	6.2.5
6.2.5-SAC-08	In meeting the requirements of **10 CFR Part 50, § 50.44(c)(5)**, regarding containment structural integrity, an analysis must demonstrate containment structural integrity, using an analytical technique that is accepted by the NRC staff and including sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by combustible gas burning. Systems necessary to ensure containment integrity must also demonstrate the capability to perform their functions under these conditions. One acceptable analytical technique is a demonstration that specific criteria of the ASME Boiler and Pressure Vessel Code, described in **RG 1.7, Revision 3, Section C.5**, are met.	Y	6.2.5
6.2.5-SAC-09	In meeting the requirements of **10 CFR Part 50, § 50.44(c)**, and **GDC 41** for the design and functional capability of the combustible gas control systems, preliminary system designs and statements of intent in the SAR are acceptable at the CP stage of review if the guidelines of **RG 1.7**, Revision 3, are endorsed.	Y	6.2.5

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 6.2.6	**Containment Leakage Testing (R3, 03/2007)**		
6.2.6-AC-01	**General Design Criterion 52 (GDC 52),** "Capability for Containment Leakage Rate Testing," as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the CILRT (up to the containment design pressure).	Y	6.2.6
6.2.6-AC-02	**General Design Criterion 53 (GDC 53)**, "Provisions for Containment Testing and Inspection," as it relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate testing at the containment design pressure of penetrations having resilient seals and expansion bellows.	Y	6.2.6
6.2.6-AC-03	**General Design Criterion 54 (GDC 54)**, "Piping System Penetrating Containment," as it relates to piping systems penetrating primary reactor containment being designed with a capability to determine if valve leakage rate is within acceptable limits.	Y	6.2.6
6.2.6-AC-04	**10 CFR 100.11** requires that, as an aid in evaluating a proposed nuclear power plant site, an applicant should assume the expected demonstrable leakage rate from the containment. Nuclear power plant leakage rate testing experience shows that a design leakage rate of 0.1% per day provides adequate margin above typically measured containment leakage rates and is compatible with current leakage rate test methods and test acceptance criteria. Therefore, the minimum acceptable design containment leakage rate should not be less than 0.1% per day.	Y	6.2.6
6.2.6-AC-05	**10 CFR 100.10** addresses factors to be considered when evaluating nuclear power plant sites and includes the safety features that are engineered into the facility. The secondary containment of dual-type containments, which provide for a controlled, filtered release to the environs of leakage from the primary reactor containment, is such an engineered safety feature, whose effectiveness should be periodically tested as stated in 10 CFR 50,	Y	6.2.6

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Appendix J, Option A, in Section IV.B.; Option B plants should also be tested in the same way. In so doing, the leakage limit of the secondary containment is acceptable if it is based on the limit used in the analysis of the secondary containment depressurization time. The test should be conducted at each refueling outage or at a comparable frequency. The test limit should be consistent with the limit used for direct leakage in the analysis of the radiological consequences by the organization responsible for analysis of radiological consequences. Potential bypass leak paths (identified in accordance with **Branch Technical Position 6-3**, "Determination of Bypass Leakage Paths in Dual Containment Plants") should be locally leakage rate tested in accordance with the requirements of Appendix J.		
6.2.6-AC-06	The reactor containment leakage rate testing program, as described in the safety analysis report (SAR) or design certification document (DCD), will be acceptable if: A. Under Option A, it meets the requirements stated in **Option A of Appendix J to 10 CFR Part 50. Appendix J**, Option A, provides the test requirements and acceptance criteria for preoperational and periodic leakage rate testing of the reactor containment and of systems and components which penetrate the containment. Exemption from Appendix J requirements will be reviewed on a case-by-case basis. B. Under Option B, it meets the requirements stated in **Option B of Appendix J to 10 CFR Part 50** and, **under section V.B.2 and V.B.3** of Option B, either complies with methods approved by the Commission and endorsed in a regulatory guide (**RG 1.163**) and includes a requirement to do so in the Technical Specifications, or complies with the provisions of some other implementation document which has been adequately justified to the staff, with supporting analyses, and is cited as a requirement in the Technical Specifications. As of the publication date of this SRP revision, virtually all applicants and licensees using Option B have chosen compliance with RG 1.163, so this Standard Review Plan (SRP) is written assuming that future applicants will do the same.	Y	6.2.6

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.2.6-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.6-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.2.6-SAC	**10 CFR 50, Appendix J, Option A, Section III.A.1(a)**, requires that no repairs or adjustments be made to the containment prior to the performance of the CILRT so that the containment can be tested in as close to the "as is" condition as practical. Under **10 CFR 50, Appendix J, Option B**, **RG 1.163** endorses NEI 94-01, Rev. 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 9.0. Instrumentation lines that penetrate containment, however, are sometimes isolated for the CILRT. To ensure that they are included in the test, the following should be done. Leakage rate testing of instrumentation lines that penetrate containment may be done in conjunction with either the LLRTs or the CILRT. Instrumentation lines that are not locally leakage rate tested should not be isolated from the containment atmosphere during the performance of the CILRT. The measured leakage rates from instrumentation lines that are locally leakage rate tested, and also isolated during CILRTs, should be added to the CILRT result. Provisions should be made to ensure that instrumentation lines isolated during the	Y N/A-BWR (where noted)	6.2.6 N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	CILRT are restored to their operable status following the test. All leakage rate tests, performed by either pneumatic or hydrostatic means, should have the capability to quantify the leakage rates either explicitly or by a conservative bounding method to satisfy test acceptance criteria in Appendix J and the Technical Specifications. Appendix J, Option A, Section III.C.1, prescribes methods for conducting the containment isolation valve leakage rate tests. Under Option B, RG 1.163 endorses NEI 94-01, Rev. 0 (with certain exceptions), which provides similar guidance in Sections 8.0 and 10.0. At the construction permit (CP) or standard design certification stage, the applicant should identify all containment isolation valves that will be locally (Type C) leakage rate tested with the test pressure applied in a direction opposite to that which would occur under accident conditions and should commit to justify, at the OL or COL stage, that such testing will result in equivalent or more conservative results. With regard to the application of Appendix J, Option A, Section III.C.1, and Option B, Section III.B., for leakage rate testing of main steam isolation valves (MSIVs) in boiling water reactor (BWR) plants, if a test pressure of less than Pa (calculated peak containment accident pressure) is necessary, the test pressure and the test acceptance criteria should be justified and included in the plant Technical Specifications. Further, this will require an exemption from the applicable Appendix J requirement and the applicant or licensee must request one, with appropriate justification. In addition, it is typical for BWR applicants or licensees to request that MSIV local leakage rates be excluded from the Type A test leakage rate and the sum of Type B and Type C leakage rates, which would require exemption from Option A, Sections III.A, III.B.3, and III.C.3, or Option B, Sections III.A and III.B. Such exemptions must also be requested and justified, and may readily be combined with an exemption request regarding test pressure and acceptance criteria, mentioned above. NEI 94-01, Rev. 0 (Section 6.0), and ANSI/ANS-56.8-1994 (Section 3.3.1) state that Type		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B or Type C tests are not required for the following cases: 1. Containment boundaries that do not constitute potential containment atmospheric leakage pathways during and following a design-basis loss-of-coolant accident (DB LOCA); 2. Containment boundaries sealed with a qualified seal system; 3. Test connections, vents, and drains between containment isolation valves which: A. are one inch or less in size, and B. administratively secured closed, and C. consist of a double barrier (e.g., two valves in series, one valve with a nipple and cap, one valve and a blind flange). This guidance may be applied to either Option A or Option B of Appendix J. Examples of Case No. 1 are lines that terminate below the minimum post-accident water level of the suppression pool in a BWR or the recirculation sump in a PWR. For Case No. 2, a qualified seal system is defined in ANSI/ANS-56.8-1994 as a system that is capable of sealing the leakage with a liquid at a pressure no less than 1.1 Pa, for at least 30 days following the DB LOCA. The staff's position is that the analysis of the sealing capability includes the assumption of the most limiting single failure of any active component. Also, unless there is a virtually unlimited supply of sealing liquid (such as from a suppression pool or recirculation sump), limits for liquid leakage rate should be assigned to these valves based on analysis and included in the plant technical specifications. Periodic leakage rate testing, using the sealing liquid as the test medium, is then needed to ensure that the technical specification limits are maintained. For Case No. 3, to ensure that containment integrity is restored following testing, the test, vent, and drain connections that are used to facilitate local leakage rate testing and the performance of the CILRT should be under administrative control and should be subject to periodic surveillance, to ensure their integrity and to verify the effectiveness of		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	administrative controls. The testing requirements for BWR drywell steam bypass are discussed in SRP Section 6.2.1.1.C.		
SRP 6.2.7	**Fracture Prevention of Containment Pressure Boundary (R1, 03/2007)**		
6.2.7-AC-01	**GDC 1**, found in Appendix A to Part 50, as it relates to the quality standards for design and fabrication.	Y	6.2.7
6.2.7-AC-02	**GDC 16**, as it relates to the prevention of the release of radioactivity to the environment.	Y	6.2.7
6.2.7-AC-03	**GDC 51**, as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.	Y	6.2.7
6.2.7-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.2.7-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.2.7-SAC-01	To meet the requirements of **GDC 1, 16 and 51**, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in **Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code** or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the **Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code**.	Y	6.2.7
6.2.7-SAC-02	Mandatory fracture toughness testing of ASME Code Section III Class 2 materials was first identified in the **Summer 1977 Addenda Code Class 2 rules**. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in **NUREG-0577**, “Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports,” and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials which were not fracture toughness tested.	Y	6.2.7
SRP 6.3	**Emergency Core Cooling System (R3, 03/2007)**		
6.3-AC-01	**General Design Criterion (GDC) 2** as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction	Y	5.4.7 6.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	in the capability of the ECCS to perform its safety function.		6.8
6.3-AC-02	**GDC 4** as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer)	Y	5.4.7 6.3
6.3-AC-03	**GDC 5** as it relates to SSCs important to safety shall not be shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function.	Y	5.4.7 6.3 6.8
6.3-AC-04	**GDC 17** as it relates to the design of the ECCS having sufficient capacity and capability to assure that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences and that the core is cooled during accident conditions.	Y	5.4.7 6.3 6.8
6.3-AC-05	**GDC 27** as it relates to the system design having the capability to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.	Y	5.4.7 6.3 6.8
6.3-AC-06	**GDCs 35, 36, and 37** as they relate to the ECCS being designed to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing.	Y	5.4.7 6.3 6.8
6.3-AC-07	**10 CFR 50.46**, in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models, which identifies and accounts for uncertainties in the analysis method and inputs; alternatively, an ECCS evaluation model may be developed in conformance with **Appendix K to 10 CFR Part 50**.	Y	5.4.7 6.3 6.8
6.3-AC-08	TMI Action Plan item II.K.3.18 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(vii)** for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual	N/A-BWR	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	actuation of the BWR ADS to assure adequate core cooling.		
6.3-AC-09	TMI Action Plan item II.K.3.21 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(viii)** for applicants subject to 10 CFR 50.34(f), with respect to studying the design of BWR 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.	N/A-BWR	N/A
6.3-AC-10	TMI Action Plan item II.K.3.28 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(x)** for applicants subject to 10 CFR 50.34(f), with respect to BWR ADS-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves.	N/A-BWR	N/A
6.3-AC-11	TMI Action Plan item II.K.3.45 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(xi)** for applicants subject to 10 CFR 50.34(f), with regard to providing an evaluation of depressurization methods, other than full actuation of the ADS, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown for BWRs.	N/A-BWR	N/A
6.3-AC-12	TMI Action Plan item III.D.1.1 of NUREG-0737, equivalent to **10 CFR 50.34(f)(2)(xxvi)** for applicants subject to 10 CFR 50.34(f), with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the ECCS outside of the containment that contain or may contain radioactive material following an accident.	Y	5.2.5 6.3
6.3-AC-13	TMI Action Plan item II.K.3.16 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(vi)** for applicants subject to 10 CFR 50.34(f), with regard to providing an evaluation of methods to reduce challenges and failures of reactor coolant system relief valves for BWRs.	N/A-BWR	N/A
6.3-AC-14	TMI Action Plan item II.K.3.24 of NUREG-0737, equivalent to **10 CFR 50.34(f)(1)(ix)** for	N/A-BWR	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	applicants subject to 10 CFR 50.34(f), with respect to the adequacy of space cooling for long-term operation of HPCI and RCIC systems for BWRs to maintain the operating environment within allowable limits.		
6.3-AC-15	TMI Action Plan item II.D.3 of NUREG-0737, equivalent to **10 CFR 50.34(f)(2)(xi)** for applicants subject to 10 CFR 50.34(f), with respect to the requirements that reactor coolant system relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe.	Y	5.4.7 5.4.13 6.3
6.3-AC-16	TMI Action Plan item II.F.2 of NUREG-0737, equivalent to **10 CFR 50.34(f)(2)(xviii)** for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling.	Y	5.4.7 6.3 7.5
6.3-AC-17	**10 CFR 52.47(b)(1)**, which requires that a Design Certification (DC) application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.3-AC-18	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.3-SAC-01	In regard to the ECCS acceptance criteria of **10 CFR 50.46**, the five major performance	Y	6.8

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria deal with: A. Peak cladding temperature. B. Maximum calculated cladding oxidation. C. Maximum hydrogen generation. D. Coolable core geometry E. Long-term cooling. 10 CFR 50.46, requires that the ECCS be designed so that the calculated cooling performance is in accordance with an acceptable evaluation model or alternately a model in conformance with the features of Appendix K.		15.0
6.3-SAC-02	The ECCS must meet the requirements of **GDC 35**. The system must have alternate sources of electric power, as required by **GDC 17**, and must be able to withstand a single failure. The ECCS should retain its capability to cool the core in the event of a failure of any single active component during the short term immediately following an accident, or a single active or passive failure during the long-term recirculation cooling phase following an accident.	Y	5.4.7 6.3 6.8
6.3-SAC-03	The ECCS must be designed to permit periodic inservice inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, piping, pumps, and valves in accordance with the requirements of **GDC 36**. The ECCS must be designed to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation, as required by **GDC 37**.	Y	5.4.7 6.3 6.8
6.3-SAC-04	The combined reactivity control system capability associated with ECCS must meet the requirements of **GDC 27** and should conform to the recommendation of **Regulatory Guide 1.47**. The primary mode of actuation for the ECCS must be automatic, and actuation must be initiated by signals of suitable diversity and redundance. Provisions	Y	5.4.7 6.3 6.8

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	should also be made for manual actuation, monitoring, and control of the ECCS from the reactor control room.		
6.3-SAC-05	The design of the ECCS should conform to the recommendations of **Regulatory Guide 1.1**.	Y	5.4.7 6.3
6.3-SAC-06	Design features and operating procedures, designed to prevent damaging water hammer due to such mechanisms as voided discharge lines and water entrainment in steam lines shall be provided, in order to meet the requirements of **GDC 4**.	Y	6.3
6.3-SAC-07	The design of those portions of the system which are not safety related, whose failures could have an adverse effect on the ECCS system, must be in accordance with **GDC 2**, and acceptance is based on meeting **Position C2 of Regulatory Guide 1.29**.	Y	5.4.7 6.3 6.8
6.3-SAC-08	Interfaces between the ECCS and component or service water systems must be such that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems, e.g., residual heat removal (RHR) and containment heat removal systems, the ECCS must conform to **GDC 5**.	Y	5.4.7 6.3
6.3-SAC-09	The requirements of **Task Action Plan Item II.K.3(15) of NUREG-0737 and NUREG-0718**, which involves isolation of HPCI and RCIC for BWR plants, should also be satisfied.	N/A-BWR	N/A
6.3-SAC-10	The requirements and guidance regarding ECCS outage times and reports on ECCS unavailability, contained in **Task Action Plan Item II.K.3.17**, and **Generic issue B-61**, must also be satisfied.	Y	5.4.7 6.3 6.8
SRP 6.4	**Control Room Habitability System (R3, 03/2007)**		
6.4-AC-01	**General Design Criterion 4 (GDC 4)**, "Environmental and Dynamic Effects Design Bases," as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents.	Y	6.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.4-AC-02	**General Design Criterion 5 (GDC 5)**, "Sharing of Structures, Systems and Components," as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s).	Y	6.4
6.4-AC-03	**General Design Criterion 19 (GDC 19)**, "Control Room," as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection.	Y	6.4
6.4-AC-04	**10 CFR 50.34(f)(2)(xxviii)**, as it relates to evaluations and design provisions to preclude certain control room habitability problems. For Part 50 applicants not listed in 10 CFR 57.34(f), the provisions of 50.34(f) will be made a requirement during the licensing review.	Y	6.4
6.4-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.4-AC-06	**10 CFR 52.80(a),** which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
6.4-SAC-01	<u>Control Room Emergency Zone</u> The control room emergency zone should include the following: A. Instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file; B. Computer room, if it is used as an integral part of the emergency response plan; C. Shift supervisor's office; and D. Operator washroom and the kitchen. E. The control room emergency zone should conform to the guidelines of **Regulatory Guide 1.196**, May 2003, "Control Room Habitability at Light Water Nuclear Power Reactors," and **Regulatory Guide (RG) 1.197**, May 2003, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."	Y	6.4
6.4-SAC-02	<u>Ventilation System Criteria</u>. The ventilation system should include the following design features: A. Isolation dampers used to isolate the control zone from adjacent zones or the outside should be low leakage dampers or valves. The degree of leaktightness should be documented in the SAR. B. Single failure of an active component should not result in loss of the system's functional performance. All the components of the control room emergency filter train should be considered active components. See Appendix A to this SRP for criteria regarding valve or damper repair.	Y	6.4
6.4-SAC-03	<u>Pressurization Systems</u>. Ventilation systems that will pressurize the control room during a radiation emergency should meet the following criteria: A. Systems having pressurization rates of greater than or equal to 0.5 volume changes per hour should be subject to periodic verification (every 18 months) that the makeup	Y	6.4

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	is + 10% of design value. During plant construction or after any modification to the control room that might significantly affect its capability to maintain a positive pressure, measurements should be taken to verify that the control room emergency zone is pressurized to at least to the value used in the accident analysis relative to all surrounding air spaces while applying makeup air at the design rate. B. Systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour should have identical testing requirements as indicated in acceptance criteria 1 above. In addition, at the construction permit (CP), combined license, or standard design certification stage, an analysis should be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining the tested differential pressure with the design makeup airflow rate. C. Systems having pressurization rates of less than 0.25 volume changes per hour should meet all the criteria for acceptance criteria 2 above, except that periodic verification of control room pressurization (every 18 months) should be specified.		
6.4-SAC-04	Emergency Standby Atmosphere Filtration System. Iodine removal for this system should be in accordance with the guidelines of **Regulatory Guide 1.52**. For new applications, the system should also conform with **ASME Code AG-1**, "Code on Nuclear Air and Gas Treatment" including the AG-1a-92 Addenda (Reference 14). Protection of control room personnel from releases of chlorine or other toxic gases is addressed in **Regulatory Guide 1.78** as discussed in the criteria below.	Y	6.4
6.4-SAC-05	Relative Location of Source and Control Room. The control room inlets should be located with consideration of the potential release points of radioactive material and toxic gases. Specific criteria as to radiation and toxic gas sources are as follows: A. Radiation sources. As a general rule the control room ventilation inlets should be separated from the	Y	6.4

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	major potential release points by at least 31 meters (100 feet) laterally and by 16 meters (50 feet) vertically. However, the actual minimum distances should be based on the dose analyses (Ref. 9). B. Toxic gases. The minimum distance between the toxic gas source and the control room is dependent upon the amount and type of the gas in question, the container size, and the available control room protection provisions. The acceptance criteria for the control room habitability system are provided in the regulatory positions of **Regulatory Guide 1.78** with respect to postulated hazardous chemical releases in general.		
6.4-SAC-06	Radiation Hazards A. For current operating reactors that do not implement an alternative source term under **10 CFR 50.67**, 10 CFR Part 50, Appendix A, **General Design Criterion 19 (GDC 19)** “Control room,” requires that “Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.” In accordance with GDC 19, these doses to an individual in the control room should not be exceeded for any postulated design basis accident. The whole body gamma dose consists of contributions from airborne radioactivity inside and outside the control room, as well as direct shine from all radiation sources. i. For current operating reactors the dose guidelines for evaluating the emergency zone radiation protection provisions are as follows: whole body gamma: 50 mSv (5 rem) thyroid: 300 mSv (30 rem)	Y	6.4

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	beta skin dose: 300 mSv (30 rem) ii. For current operating reactors conforming to and implementing the guidance of **RG 1.195** in conjunction with **RG 1.196**, the dose guidelines for evaluating the emergency zone radiation protection provisions are relaxed as follows: whole body gamma: 50 mSv (5 rem) thyroid: 500 mSv (50 rem) beta skin dose: 500 mSv (50 rem) B. Applicants for and holders of construction permits and operating licenses under 10 CFR Part 50 who apply on or after January 10, 1997, applicants for design certifications under **10 CFR Part 52** who apply on or after January 10, 1997, applicants for and holders of combined licenses under 10 CFR Part 52 who do not reference a standard design certification, or holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of **GDC 19**, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in **10 CFR 50.2** for the duration of the accident.		
6.4-SAC-07	<u>Toxic Gas Hazards</u>. Three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance; there should be no chronic effects from exposure; acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of any measures other than the use of self-	Y	6.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Engineered Safety Features

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	contained breathing apparatus. The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows: A. Protective action limit (2 minutes or less): Use a limit that will ensure that the operators will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes. B. Short-term limit (2 minutes to 1 hour): Use a limit that will ensure that the operators will not suffer incapacitating effects after a 1-hour exposure. C. Long-term limit (1 hour or greater): Use a limit assigned for occupational exposure (40-hour week). The protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. They are also used in those cases where the toxic levels are such that emergency zone isolation without use of protective gear is sufficient. Self-contained breathing apparatus for the control room personnel (at least 5 individuals) should be on hand. A 6-hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). As an example of appropriate limits, the following are the three levels for chlorine gas:		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	protective action: 15 ppm by volume short-term: 4 ppm by volume long-term: 1 ppm by volume **Regulatory Guide 1.78** provides a partial list of protective action levels for other toxic gases.		
SRP 6.5.1	**ESF Atmosphere Cleanup Systems (R3, 03/2007)**		
6.5.1-AC-01	**General Design Criterion (GDC) 19**, as it relates to maintaining the control room in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs).	Y	6.5.1
6.5.1-AC-02	**GDC 41**, as it relates to providing systems to control the release of fission products to the environment and to control the concentration of hydrogen, oxygen, and other substances in containment following postulated accidents.	Y	6.5.1
6.5.1-AC-03	**GDC 42**, as it relates to designing containment ESF atmosphere cleanup systems to permit inspection.	Y	6.5.1
6.5.1-AC-04	**GDC 43**, as it relates to designing containment ESF atmosphere cleanup systems to permit pressure and functional testing.	Y	6.5.1
6.5.1-AC-05	**GDC 61** as it relates to the design of systems for radioactivity control under normal and postulated accident conditions.	Y	6.5.1
6.5.1-AC-06	**GDC 64** as it relates to monitoring releases of radioactivity from normal operations , including anticipated operational occurrences, and from postulated accidents.	Y	6.5.1
6.5.1-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.5.1-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.5.1-SAC-01	Compliance with **GDC 19** requires radiation protection of the control room to ensure that access and occupancy under accident conditions, including LOCAs, will not result in radiation exposures in excess of the specified limit. GDC 19 applies to this section because control room radiation protection under accident conditions may require an ESF atmosphere cleanup system. Meeting this criterion provides assurance that personnel needed to monitor and control an accident will be able to perform those functions effectively.	Y	6.5.1
6.5.1-SAC-02	Compliance with **GDC 41** requires systems to control fission products that may be released into the reactor containment, thereby reducing the concentration of fission products released to the environment after an accident. GDC 41 also includes redundancy and reliability requirements for such systems. GDC 41 applies to this section because control of fission products released from the containment after an accident may require an ESF atmosphere cleanup system. Meeting this criterion provides assurance that offsite radiation doses resulting from an accident will be within regulatory limits.	Y	6.5.1
6.5.1-SAC-03	Compliance with **GDC 42** requires that containment atmosphere cleanup systems be	Y	6.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	designed to accommodate periodic inspection of important components such as filter frames, ducts, and piping. GDC 42 applies to this section because the containment atmosphere cleanup system may be an ESF. Meeting this criterion provides assurance that the equipment necessary to mitigate the consequences of an accident will maintain its functional capability.		
6.5.1-SAC-04	Compliance with **GDC 43** requires that containment atmosphere cleanup systems be designed to accommodate periodic pressure and functional testing. GDC 43 applies to this section because the containment atmosphere cleanup system may be an ESF. Meeting this criterion provides assurance that the equipment necessary to mitigate an accident will maintain its functional capability.	Y	6.5.1
6.5.1-SAC-05	Compliance with **GDC 61** requires that fuel storage and handling, radioactive waste, and other systems that may contain radioactive material be designed to ensure adequate safety under normal and postulated accident conditions. These systems shall be designed with appropriate containment, confinement, and filtering systems. GDC 61 applies to this section because attainment of the objectives for postulated accident conditions may require an ESF atmosphere cleanup system. Meeting this criterion provides assurance that offsite doses of radiation resulting from accident conditions will not exceed regulatory limits.	Y	6.5.1
6.5.1-SAC-06	Compliance with **GDC 64** requires monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs to detect radioactivity that may be released from normal operations (including anticipated operational occurrences) and postulated accidents.	Y	6.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	GDC 64 applies to this section because the ESF atmosphere cleanup systems are used to control releases of radioactivity from postulated accidents. The review ensures that means are provided to monitor releases. Meeting this criterion provides assurance that offsite doses of radiation resulting from accident conditions will not exceed regulatory limits and that releases will be adequately documented.		
6.5.2	**Containment Spray as a Fission Product Cleanup System (R4, 03/2007)**	N/A-OTHER (Containment sprays not credited in DBA.)	6.5.2
6.5.3	**Fission Product Control Systems and Structures (R3, 03/2007)**		
6.5.3-AC-01	**General Design Criterion (GDC) 41** as it relates to the containment atmosphere cleanup system being designed to control fission product releases to the environment following postulated accidents.	N/A-OTHER (Containment sprays not credited in DBA.)	6.5.3
6.5.3-AC-02	**General Design Criterion (GDC) 42** as it relates to the containment atmosphere cleanup system being designed to permit periodic inspections.	N/A-OTHER (Containment sprays not credited in DBA.)	6.5.3
6.5.3-AC-03	**General Design Criterion (GDC) 43** as it relates to the containment atmosphere cleanup system being designed to permit appropriate functional testing.	N/A-OTHER (Containment sprays not credited in DBA.)	6.5.3
6.5.3-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.5.3-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
6.5.3-SAC-01	Primary Containment. Primary containment design leakage rates for which credit is given should not be less than 0.1% per day due to difficulties in measuring lower leakage rates. Containment isolation methods and times must be such that the calculated radiological doses resulting from the escape of radioactive material prior to and following isolation after a LOCA do not exceed the applicable dose requirements of **10 CFR Part 100** and **GDC 19**. The primary reactor containment and associated systems should be designed so that periodic inspections and functional testing can be performed.	Y	6.5.3
6.5.3-SAC-02	Secondary Containment. To be classified as a secondary containment for the purpose of fission product control, a structure or structures should completely surround the primary containment, and at least should be held at a pressure of 0.6 cm (0.25 in) (water), below adjacent regions, under all wind conditions up to the wind speed at which diffusion becomes great enough to ensure site boundary exposures less than those calculated for the design basis accidents even if	Y	6.5.3

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	exfiltration occurs. Acceptance of other fission product control structures for collection and control of postaccident releases will be determined following consultation with the organization responsible for the review of reactor accident consequence assessment, (specifically design basis containment and ventilation performance) and the organization responsible for structural design of containment and ventilation systems, on a case-by-case basis. The leakage and filtration rates of such structures are acceptable provided that the offsite doses calculated by the organization responsible for radiation protection under **SRP Section 15.6.5** will meet the dose guidelines of **10 CFR Part 100** and provided that the preoperational testing and appropriate technical specifications are acceptable. Other criteria include specifications for intake and return headers on recirculation systems. These should be placed as far away from each other as practical. The return header should provide a wide distribution over the secondary containment. The purpose of this placement is to ensure some degree of mixing of the return flow in the secondary containment volume before it is again drawn into the system intake. With judicious placement, up to 50% mixing may be assumed. A claim for greater than 50% mixing must be supported by the applicant to the satisfaction of the staff. Spacing between intake and return headers is reviewed on a case-by-case basis. Adjustments in the mixing fraction to less than 50% may be indicated by some designs. Past practice has been to allow mixing in 50% of the volume between — and within 3 or 6 meters (10 or 20 feet) of — the inlet and outlet headers if both have distributed openings or if one has distributed openings and the other is at the top of the containment. Partial dual containments should meet the same basic criteria as secondary containments in order to be given credit for fission product holdup and removal. The fraction of leakage source considered to be controlled by such partial fission products control structures is determined after consultation with the SCSB reviewer on a case-by-case basis.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
6.5.3-SAC-03	Specific SRP acceptance criterion 3 **not used**.	N/A	N/A
6.5.3-SAC-04	Other Fission Product Control Systems. Fission product retention credit may be taken by the applicant for other systems- e.g., containment spray systems as evaluated in **SRP Section 6.5.2**, pressure suppression pools as evaluated in **SRP Section 6.5.5**, and filtration and adsorption units as described in **Regulatory Guide 1.52**. Justification for fission product retention systems should include analytical bases addressing the important physical and chemical variables of the fission product removal and retention processes.	Y	6.5.3
SRP 6.5.4	**Ice Condenser as a Fission Product Cleanup System (Draft R4, 04/1996)**	N/A-ICE	N/A
SRP 6.5.5	**Pressure Suppression Pool as a Fission Product Cleanup System (R1, 03/2007)**	N/A-BWR	N/A
SRP 6.6	**Inservice Inspection and Testing of Class 2 and 3 Components (R2, 03/2007)**		
6.6-AC-01	**10 CFR 50.55a** as it pertains to specification of the preservice and periodic inspection and testing requirements of the ASME Code for Class 2 and 3 systems and components.	Y	6.6
6.6-AC-02	**General Design Criterion (GDC) 36** found in Appendix A to 10 CFR Part 50, as it pertains to designing the emergency core cooling system to permit appropriate periodic inspection of important safety components, such as spray rings in the reactor pressure vessel.	Y	6.6
6.6-AC-03	**GDC 37** found in Appendix A to 10 CFR Part 50, as it pertains to designing the emergency core cooling system to permit appropriate testing to assure structural integrity, leak tightness, and the operability of the system.	Y	6.6
6.6-AC-04	**GDC 39** found in Appendix A to 10 CFR Part 50, as it pertains to designing the containment heat removal system to permit inspection of important components, such as the torus and spray nozzles to asure the integrity and capability of the system.	Y	6.6
6.6-AC-05	**GDC 40** found in Appendix A to 10 CFR Part 50, as it pertains to designing the	Y	6.6

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	containment heat removal system to permit appropriate pressure and functional testing.		
6.6-AC-06	**GDC 42** found in Appendix A to 10 CFR Part 50, as it pertains to designing the containment atmospheric clean up system tp permit appropriate inspection of components such as filter frames and ducts.	Y	6.6
6.6-AC-07	**GDC 43** found in Appendix A to 10 CFR Part 50, as it pertains to designing the containment atmospheric clean up system to permit appropriate periodic pressure and functional testing to assure structural integrity of components and the operability and performance of active components of the system, such as fans, filters, and dampers.	Y	6.6
6.6-AC-08	**GDC 45** found in Appendix A to 10 CFR Part 50, as it pertains to designing the cooling water system tp permit appropriate periodic inspection of important components, such as heat exchangers.	Y	6.6
6.6-AC-09	**GDC 46** found in Appendix A to 10 CFR Part 50, as it pertains to designing the cooling water system to permit appropriate pressure and functional testing to assure structural and leaktight integrity of it components.	Y	6.6
6.6-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
6.6-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
6.6-SAC-01	Components Subject to Inspection. The applicant's definition of ASME Code Class 2 and 3 components and systems subject to an ISI program is acceptable if it is in agreement with the NRC quality group classification system or the definitions in **Article NCA-2000 of Section III of the ASME Code**. The classification of components by the applicant is subject to review under **SRP Section 3.2.2** for compliance with safety criteria pertaining to component classification. Where a specific item will be subject to inspection requirements different in any way from the **ASME Code Section XI** requirements corresponding to the item's Code Class, the exceptions for the item, including the inservice inspection requirements to be applied, should be clearly identified and described. Exceptions involving less stringent inspection requirements for Code Class 2 or 3 items other than those required by Section XI must be adequately justified. (Refer to **SRP Section 3.2.2** or **Article NCA-2000 of Section III of the ASME Code**.)	Y	6.6
6.6-SAC-02	Accessibility. The design and arrangement of Class 2 and 3 systems should include allowances for adequate clearances to conduct the examinations specified in **Articles IWC-2000 and IWD-2000** at the frequency specified. The design and arrangement of system components are acceptable if adequate clearance is provided in accordance with **Subarticle IWA-1500**. Special design considerations are given to those systems that are intended to be examined during normal reactor operation.	Y	6.6
6.6-SAC-03	Examination Categories and Methods. The examination categories and requirements specified in the SAR are acceptable if they are in agreement with the rules of **Articles IWA-2000, IWC-2000, and IWD-2000**. Every area subject to examination should fall within one or more of the examination categories	Y	6.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and must be examined at least to the extent specified. The applicant's examination techniques and procedures used for preservice inspection and inservice inspection are acceptable if they are in agreement with the following criteria: A. The methods, techniques, and procedures for visual, surface, or volumetric examination are in accordance with **Article IWA-2000**. B. Alternative examination methods, combination of methods, or newly developed techniques to those given in A. above are acceptable provided that the results are equivalent or superior. The acceptance standards for these alternate methods are given in **Articles IWC-3000 and IWD-3000**. C. The methods, procedures, and requirements regarding qualification of personnel performing ultrasonic examination reflect the guidance provided in **Appendix VII**, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," **to Division 1 of Section XI of the ASME Code**. D. Performance demonstration for ultrasonic examination procedures, equipment, and personnel used to detect and size flaws are in accordance with the requirements of **Appendix VIII**, "Performance Demonstration for Ultrasonic Examination Systems," **to Division 1 of Section XI of the ASME Code**.		
6.6-SAC-04	Inspection Intervals. The ISI program schedule provided in the SAR is acceptable if the required examinations and pressure tests are specified for completion during each ten-year interval, hereinafter designated as the "inspection interval," and as required by **ASME Section XI, Articles IWA-2000, IWC-2000, and IWD-2000**.	Y	6.6
6.6-SAC-05	Evaluation of Examination Results. The methods for evaluation of examination results are reviewed for compliance with **Articles IWC-3000 and IWD-3000 in the Code**. If the applicable edition of the Code	Y	6.6

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CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	states that these articles are in the course of preparation, the rules of **Article IWB-3000** shall apply. The repair procedures are acceptable if they are in compliance with **ASME Section XI, Article IWA-4000**.		
6.6-SAC-06	System Pressure Tests. The program provided in the SAR for Class 2 and 3 system pressure testing is acceptable if it meets the criteria of **ASME Section XI, Articles IWC-5000 and IWD-5000**.	Y	6.6
6.6-SAC-07	Augmented ISI to Protect Against Postulated Piping Failures. The augmented ISI program for high-energy fluid system piping between containment isolation valves is acceptable if it specifies the following requirements: A. Protective measures, structures, and guard pipes should not prevent the access required to conduct the inservice examinations specified in the **Division 1 of Section XI of the ASME Code**. B. For those portions of high energy fluid system piping between containment isolation valves, the extent of inservice examination completed during each inspection interval should provide 100% volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping. C. For those portions of high-energy fluid system piping enclosed in guard pipes, inspection ports should be provided in the guard pipes to permit the required examination of circumferential pipe welds. Inspection ports should not be located in that portion of the guard pipe passing through the annulus of dual barrier containment structures. D. The areas subject to examination should be defined in accordance with **Article IWC-2000, Examination Category C-F** for Class 2 piping welds.	Y	6.6
6.6-SAC-08	Code Exemptions. The exemptions from Code examination requirements identified by the applicant are	Y	6.6

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	acceptable if they have been permitted by **Subsubarticles IWC-1220 or IWD-1220 of Section XI of the ASME Code**.		
6.6-SAC-09	Relief Requests. Request for relief from the **ASME Code Section XI** examination requirements that are found to be impractical due to the limitations of design, geometry, or materials of construction of components are evaluated in accordance with **10 CFR 50.55a**.	Y	6.6
6.6-SAC-10	Code Cases. The exemptions from Code examination requirements identified by the applicant or licensee are acceptable if they have been permitted by appropriate ASME code cases.	Y	6.6
6.6-SAC-11	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Preservice Inspection and Inservice Inspection and testing programs for Class 2 and 3 components are reviewed in accordance with the requirements of **10 CFR 50.55a**, “Codes and Standards.” The implementation milestone for the inservice inspection program is when the plant enters into commercial operation.	N/A-COL	N/A
SRP 6.7	**Main Steam Isolation Valve Leakage Control System (BWR), (Draft R3, 04/1996)**	N/A-BWR	N/A
BTP 6-1	**pH for Emergency Coolant Water for Pressurized Water Reactors (03/2007)**	See SRP 6.1.1, 6.1.1-SAC-02.A	
BTP 6-2	**Minimum Containment Pressure Model for PWR ECCS Performance Evaluation (R3, 03/2007)**	See SRP 6.2.3, 6.2.3-SAC-01.A	
BTP 6-3	**Determination of Bypass Leakage Paths in Dual Containment Plants (R3, 03/2007)**	See SRP 6.2.3, 6.2.3-SAC-04.A & SRP 6.2.6, 6.2.6-AC-05	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 6 Engineered Safety Features			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
BTP 6-4	**Containment Purging During Normal Plant Operations (R3, 03/2007)**	See SRP 6.2.4, 6.2.4-SAC-14 & 6.2.4-SAC-20	
BTP 6-5	**Currently the Responsibility of Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps (R3, 03/2007)**	N/A-OTHER (No ECCS Sources Outside of Containment)	N/A

CHAPTER 7
Instrumentation and Controls

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 7.0	**Instrumentation and Controls – Overview of Review Process (R5, 03/2007)**	N/A-INFO	N/A
SRP 7.0-A	**Review Process for Digital Instrumentation and Control Systems (R5, 03/2007)**	N/A-INFO	N/A
SRP 7.1	**Instrumentation and Controls – Introduction (R5, 03/2007)**		
7.1-AC-01	**SRP Table 7-1, Section 1** (10 CFR 50 and 52) **and Section 2** (10 CFR 50, Appendix A, General Design Criteria [GDC]), list the requirements applicable to I&C systems important to safety.	Refer to SRP 7.1-T	
7.1-AC-02	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with **IEEE Std. 603-1991**, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.1-AC-03	I&C safety systems are the systems that are relied on to remain functional during and following design basis events to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to **10 CFR**	Y	7.1

CHAPTER 7 **Instrumentation and Controls**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	100, "Reactor Site Criteria," and/or **10 CFR 50.67**, “Accident Source Term,” guidelines. Protection systems are a subset of I&C safety systems, and I&C safety systems are a subset of I&C systems important to safety.		
7.1-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.1-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.1-SAC-01	**SRP Table 7-1, Section 3** (Staff Requirements Memoranda)**, Section 4** (Regulatory Guides)**, and Section 5** (Branch Technical Positions), list the SRP acceptance criteria applicable to I&C systems important to safety. Sources of the acceptance criteria are as follows: • Commission Papers (SECY) are issue papers submitted by the staff to the NRC commissioners to inform them about policy matters. Staff Requirements Memoranda (SRM) provide the NRC's decisions and	Y (Refer to SRP 7.1-T below)	Table 7.1-3

CHAPTER 7 Instrumentation and Controls			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	directions on the issues discussed in the SECY. • Regulatory guides describe acceptable methods for meeting regulatory requirements and provide guidance to applicant/licensees. Industry codes and standards set forth industry consensus requirements and recommended practices applicable to I&C systems for nuclear power plants. These standards are endorsed by regulatory guides, with or without modification, and provide acceptable methods for meeting the requirements of the NRC's regulations. • Branch technical positions (BTP) document the resolution of significant technical issues or questions of interpretation that have arisen in past reviews. BTPs outline acceptable approaches to a particular issue. The approaches taken in BTPs, like the recommendations of regulatory guides, are not mandatory. SECY and associated SRM, regulatory guides and their endorsed industry codes and standards, and BTPs are the guidelines used as SRP acceptance criteria for the evaluation of conformance to the requirements of the NRC's regulations.		
7.1-SAC-02	Use of **IEEE Std. 603-1991** and **IEEE Std. 279-1971** for Non-Safety Systems. IEEE Std. 603-1991 is an NRC requirement for safety systems and IEEE Std. 279-1971 is an NRC requirement only for protection systems. However, these standards require that protection and safety systems be appropriately isolated from non-safety systems. Consequently, the requirements of IEEE Std. 603-1991 and IEEE Std. 279-1971 apply to the interface between safety and non-safety systems.	Y	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2

<table>
<tr><th colspan="4">CHAPTER 7
Instrumentation and Controls</th></tr>
<tr><th>SRP Criterion</th><th>Description
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SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>The quality and reliability of systems important to safety that are not classified as safety systems should still be sufficient to minimize challenges to safety systems and to fulfill their overall role in plant non-safety strategy. Although IEEE Std. 603-1991 and IEEE Std. 279-1971 are not requirements for non-safety I&C systems, these standards describe concepts that are useful in any situation in which functional reliability is a goal. Consequently, although these standards are not SRP acceptance criteria for non-safety I&C systems, they are a source of design concepts that may be useful for the reviewer to consider. The scope of IEEE Std. 603-1991 is broader than that of IEEE Std. 279-1971, and the guidance of IEEE Std. 603-1991 is consequently readily adaptable for use in the review of non-safety I&C systems.</td><td></td><td></td></tr>
<tr><td>7.1-SAC-03</td><td><u>Location of Detailed Acceptance Criteria and Review Methods</u>
• SRP Appendix 7.1-A provides guidance on the applicability and review methods to be used in evaluating conformance to the regulatory requirements and SRP acceptance criteria for I&C systems important to safety.
In three cases the discussion of review methods are extensive and is located in separate appendices that are referenced by SRP Appendix 7.1-A. These appendices are:
• SRP Appendix 7.1-B provides guidance for evaluating conformance to the requirements of IEEE Std. 279-1971.
• SRP Appendix 7.1-C provides guidance for evaluating conformance to IEEE Std. 603-1991.
• SRP Appendix 7.1-D provides guidance for evaluating conformance to SRP acceptance criteria contained in IEEE Std. 7-4.3.2-2003,</td><td>Y
(Per AREVA Topical Reports EMF-2110-A & ANP-10272)</td><td>7.1.2.4
7.2.2.3
7.6.2.1</td></tr>
</table>

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	"IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by **Regulatory Guide 1.152**, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."		
SRP 7.1-T	**Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety (Second R5, 03/2007)**		
7.1-T-AC-1.a	**10 CFR 50.55a(a)(1) –** Quality Standards for Systems Important to Safety	Y	7.1.2.1 7.8.2.1
7.1-T-AC-1.b	**10 CFR 50.55a(h)(2) –** Protection Systems (IEEE Std 603-1991 or IEEE Std 279-1971)	Y	7.1.2.1
7.1-T-AC-1.c	**10 CFR 50.55a(h)(3) –** Safety Systems (IEEE Std 603-1991)	Y	7.1.2.1 7.8.2.1
7.1-T-AC-1.d	**10 CFR 50.34(f)(2)(v)** [I.D.3] **–** Bypass and Inoperable Status Indication	Y	7.1.2.1
7.1-T-AC-1.e	**10 CFR 50.34(f)(2)(xi)** [II.D.3] **–** Direct Indication of Relief and Safety Valve Position	Y	7.1.2.1
7.1-T-AC-1.f	**10 CFR 50.34(f)(2)(xii)** [II.E.1.2] **–** Auxiliary Feedwater System Automatic Initiation and Flow Indication	Y	7.1.2.1
7.1-T-AC-1.g	**10 CFR 50.34(f)(2)(xvii)** [II.F.1] **–** Accident Monitoring Instrumentation	Y	7.1.2.1
7.1-T-AC-1.h	**10 CFR 50.34(f)(2)(xviii)** [II.F.2] **–** Instrumentation for the Detection of Inadequate Core Cooling	Y	7.1.2.1
7.1-T-AC-1.i	**10 CFR 50.34(f)(2)(xiv)** [II.E.4.2] **–** Containment Isolation Systems	Y	7.1.2.1
7.1-T-AC-1.j	**10 CFR 50.34(f)(2)(xix)** [II.F.3] **–** Instruments for Monitoring Plant	Y	7.1.2.1

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	Conditions Following Core Damage		
7.1-T-AC-1.k	**10 CFR 50.34(f)(2)(xx)** [II.G.1] **–** Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves	Y	7.1.2.1
7.1-T-AC-1.l	**10 CFR 50.34(f)(2)(xxii)** [II.K.2.9] **–** Failure Mode and Effect Analysis of Integrated Control System	N/A-BWR	Table 7.1-3
7.1-T-AC-1.m	**10 CFR 50.34(f)(2) (xxiii)** [II.K.2.10] **–** Anticipatory Trip on Loss of Main Feedwater or Turbine Trip	N/A- BWR	Table 7.1-3
7.1-T-AC-1.n	**10 CFR 50.34(f)(2)(xxiv)** [II.K.3.23] **–** Central Reactor Vessel Water Level Recording	N/A- BWR	Table 7.1-3
7.1-T-AC-1.o	**10 CFR 50.62 –** Requirements for Reduction of Risk from Anticipated Transients without Scram	Y	7.1.2.1 7.8.2.1
7.1-T-AC-1.p	**10 CFR 52.47(b)(1) –** ITAAC for Standard Design Certification	ITAAC	Tier 1
7.1-T-AC-1.q	**10 CFR 52.80(a) –** ITAAC for Combined Licensee Applications	N/A-COL	N/A
7.1-T-AC-2.a	**GDC 1 –** Quality Standards and Records	Y	7.1.2.2 7.8.2.1
7.1-T-AC-2.b	**GDC 2 –** Design Bases for Protection Against Natural Phenomena	Y	7.1.2.2
7.1-T-AC-2.c	**GDC 4 –** Environmental and Dynamic Effects Design Bases	Y	7.1.2.2
7.1-T-AC-2.d	**GDC 10 –** Reactor Design	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.1-T-AC-2.e	**GDC 13 –** Instrumentation and Control	Y	7.1.2.2 7.5.2.1

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			7.8.2.1
7.1-T-AC-2.f	**GDC 15 –** Reactor Coolant System Design	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.1-T-AC-2.g	**GDC 16 –** Containment Design	Y	7.1.2.2
7.1-T-AC-2.h	**GDC 19 –** Control Room	Y	7.1.2.2 7.8.2.1
7.1-T-AC-2.i	**GDC 20 –** Protection System Functions	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.1-T-AC-2.j	**GDC 21 –** Protection Systems Reliability and Testability	Y	7.1.2.2
7.1-T-AC-2.k	**GDC 22 –** Protection System Independence	Y	7.1.2.2
7.1-T-AC-2.l	**GDC 23 –** Protection System Failure Modes	Y	7.1.2.2
7.1-T-AC-2.m	**GDC 24 –** Separation of Protection and Control Systems	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.1-T-AC-2.n	**GDC 25 –** Protection System Requirements for Reactivity Control Malfunctions	Y	7.1.2.2
7.1-T-AC-2.o	**GDC 28 –** Reactivity Limits	Y	7.1.2.2
7.1-T-AC-2.p	**GDC 29 –** Protection Against Anticipated Operational Occurrences	Y (Per AREVA Topical Report ANP-	7.1.2.2

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		10275)	
7.1-T-AC-2.q	**GDC 33 –** Reactor Coolant Makeup	Y	7.1.2.2
7.1-T-AC-2.r	**GDC 34 –** Residual Heat Removal	Y	7.1.2.2
7.1-T-AC-2.s	**GDC 35 –** Emergency Core Cooling	Y	7.1.2.2
7.1-T-AC-2.t	**GDC 38 –** Containment Heat Removal	Y	7.1.2.2
7.1-T-AC-2.u	**GDC 41 –** Containment Atmosphere Cleanup	Y	7.1.2.2
7.1-T-AC-2.v	**GDC 44 –** Cooling Water	Y	7.1.2.2
7.1-T-SAC-3.a	**SRM to SECY 93-087, II.Q –** Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	Y (Per AREVA Topical Report ANP-10284)	7.1.2.3
7.1-T-SAC-3.b	**SRM to SECY 93-087, II.T –** Control Room Annunciator (Alarm) Reliability	Y	7.1.2.3
7.1-T-SAC-4.a	**Regulatory Guide 1.22 –** Periodic Testing of Protection System Actuation Functions	Y	7.1.2.4 7.1.2.5
7.1-T-SAC-4.b	**Regulatory Guide 1.47 –** Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety System	Y	7.1.2.4 7.5.2.2
7.1-T-SAC-4.c	**Regulatory Guide 1.53 –** Application of the Single-Failure Criterion to Safety Systems	Y	7.1.2.4
7.1-T-SAC-4.d	**Regulatory Guide 1.62 –** Manual Initiation of Protection Actions	Y	7.1.2.4
7.1-T-SAC-4.e	**Regulatory Guide 1.75 –** Independence of Electrical Safety Systems	Y	7.1.2.4
7.1-T-SAC-4.f	**Regulatory Guide 1.97 –** Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following	Y	7.1.2.4 7.1.2.5

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	an Accident and Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants		7.5.2.2
7.1-T-SAC-4.g	**Regulatory Guide 1.105 –** Setpoints for Safety-Related Instrumentation	Y (Per AREVA Topical Report ANP-10275)	7.1.2.4
7.1-T-SAC-4.h	**Regulatory Guide 1.118 –** Periodic Testing of Electric Power and Protection Systems	Y	7.1.2.4
7.1-T-SAC-4.i	**Regulatory Guide 1.151 –** Instrument Sensing Lines	Y	7.1.2.4
7.1-T-SAC-4.j	**Regulatory Guide 1.152 –** Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4
7.1-T-SAC-4.k	**Regulatory Guide 1.168 –** Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4
7.1-T-SAC-4.l	**Regulatory Guide 1.169 –** Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4
7.1-T-SAC-4.m	**Regulatory Guide 1.170 –** Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4
7.1-T-SAC-4.n	**Regulatory Guide 1.171 –** Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4

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7.1-T-SAC-4.o	**Regulatory Guide 1.172 –** Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4
7.1-T-SAC-4.p	**Regulatory Guide 1.173 –** Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	Y (Per AREVA Topical Report ANP-10272)	7.1.2.4
7.1-T-SAC-4.q	**Regulatory Guide 1.174 –** An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	N/A-OTHER (Risk-based concepts not used for design)	Table 7.1-3
7.1-T-SAC-4.r	**Regulatory Guide 1.177 –** An Approach for Plant-Specific Risk-Informed Decision Making: Technical Specifications	N/A-OTHER (Risk-based concepts not used for design)	Table 7.1-3
7.1-T-SAC-4.s	**Regulatory Guide 1.180 –** Guidelines for Evaluating Electromagnetic and Radio- Frequency Interference in Safety-Related Instrumentation and Control Systems	Y	7.1.2.4
7.1-T-SAC-4.t	**Regulatory Guide 1.189 –** Fire Protection for Operating Nuclear Power Plants	Y	7.1.2.4
7.1-T-SAC-4.u	**Regulatory Guide 1.200 –** An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	N/A-OTHER (Risk-based concepts not used for design)	Table 7.1-3
7.1-T-SAC-4.v	**Regulatory Guide 1.204 –** Guidelines for Lightning Protection of Nuclear Power Plants	Y	7.1.2.4
7.1-T-SAC-4.w	**Regulatory Guide 1.209 –** Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems	Y	7.1.2.4

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	in Nuclear Power Plants		
7.1-T-SAC-5.a	**BTP 7-1 –** Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System	Y	7.1.2.5
		EXCEPTION (RHR valves not automatically closed upon RCS re-pressurization)	7.1.2.5
7.1-T-SAC-5.b	**BTP 7-2 –** Guidance on Requirements on Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	Y	7.1.2.5
7.1-T-SAC-5.c	**BTP 7-3 –** Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	Y	7.1.2.5
7.1-T-SAC-5.d	**BTP 7-4 –** Guidance on Design Criteria for Auxiliary Feedwater Systems	Y	7.1.2.5
7.1-T-SAC-5.e	**BTP 7-5 –** Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	Y	7.1.2.5
7.1-T-SAC-5.f	**BTP 7-6 –** Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	N/A-OTHER (Recirculation switchover not provided in IRWST design)	Table 7.1-3
7.1-T-SAC-5.g	**BTP 7-7 –** Not used	N/A	N/A
7.1-T-SAC-5.h	**BTP 7-8 –** Guidance on Application of Regulatory Guide 1.22	Y	7.1.2.4 7.1.2.5
7.1-T-SAC-5.i	**BTP 7-9 –** Guidance on Requirements for Reactor Protection System Anticipatory Trips	Y	7.1.2.5
7.1-T-SAC-5.j	**BTP 7-10 –** Guidance on Application of Regulatory Guide 1.97	Y	7.1.2.4 7.1.2.5 7.5.2.2

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7.1-T-SAC-5.k	**BTP 7-11 –** Guidance on Application and Qualification of Isolation Devices	Y	7.1.2.5
7.1-T-SAC-5.l	**BTP 7-12 –** Guidance on Application and Qualification of Isolation Devices	Y (Per AREVA Topical Report ANP-10275)	7.1.2.5
7.1-T-SAC-5.m	**BTP 7-13 –** Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Y	7.1.2.5
7.1-T-SAC-5.n	**BTP 7-14 –** Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems	Y (Per AREVA Topical Report ANP-10272)	7.1.2.5
7.1-T-SAC-5.o	**BTP 7-15 –** Not used	N/A	N/A
7.1-T-SAC-5.p	**BTP 7-16 –** Not used	N/A	N/A
7.1-T-SAC-5.q	**BTP 7-17 –** Guidance on Self-Test and Surveillance Test Provisions	Y	7.1.2.5
7.1-T-SAC-5.r	**BTP 7-18 –** Guidance on Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	Y (Per AREVA Topical Report EMF-2110-A & ANP-10272)	7.1.2.5
7.1-T-SAC-5.s	**BTP 7-19 –** Guidance on Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems	Y (Per AREVA Topical Report ANP-10284)	7.1.2.5
7.1-T-SAC-5.t	**BTP 7-20 –** Not used	N/A	N/A
7.1-T-SAC-5.u	**BTP 7-21 –** Guidance on Digital Computer Real-Time Performance	Y (Per AREVA Topical Report EMF-2110-A)	7.1.2.5

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SRP 7.1-A	**Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety (Second R5, 03/2007)**	N/A-INFO	N/A
SRP 7.1-B	**Guidance for Evaluation of Conformance to IEEE Std. 279 (R5, 03/2007)**	N/A-INFO	N/A
SRP 7.1-C	**Guidance for Evaluation of Conformance to IEEE Std. 603 (R5, 03/2007)**	N/A-INFO	N/A
SRP 7.1-D	**Guidance for Evaluation of Conformance to IEEE Std. 7-4.3.2 (Second Issuance, 03/2007)**	N/A-INFO	N/A
SRP 7.2	**Reactor Trip System (R5, 03/2007)**		
7.2-AC-01	**10 CFR 50.55a(a)(1)**, “Quality Standards.”	Y	7.1.2.1 7.8.2.1
7.2-AC-02	**10 CFR 50.55a(h)**, “Protection and Safety Systems,” requires compliance with **IEEE Std 603-1991**, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.”	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.2-AC-03	**10 CFR 50.34(f)**, “Additional TMI-Related Requirements,” or equivalent TMI action requirements imposed by Generic Letters.		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(2)(v), “Bypass and Inoperable Status Indication.”	Y	Table 7.1-3 7.1.2.1
	(2)(xxiii), “Anticipatory Trip on Loss of Main Feedwater or Turbine Trip.”	N/A-VEN	N/A
7.2-AC-04	**10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, “Quality Standards and Records.”	Y	7.1.2.2 7.8.2.1
7.2-AC-05	**GDC 2**, “Design Basis for Protection Against Natural Phenomena.”	Y	7.1.2.2
7.2-AC-06	**GDC 4**, “Environmental and Missile Design Basis.”	Y	7.1.2.2
7.2-AC-07	**GDC 10**, “Reactor Design.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.2-AC-08	**GDC 13**, “Instrumentation and Control.”	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.2-AC-09	**GDC 15**, “Reactor Coolant System Design.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.2-AC-10	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.2-AC-11	**GDC 20**, “Protection Systems Functions.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.2-AC-12	**GDC 21**, “Protection System Reliability and Testability.”	Y	7.1.2.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.2-AC-13	**GDC 22**, "Protective System Independence."	Y	7.1.2.2
7.2-AC-14	**GDC 23**, "Protection System Failure Modes."	Y	7.1.2.2
7.2-AC-15	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.2-AC-16	**GDC 25**, "Protection System Requirements for Reactivity Control Malfunctions."	Y	7.1.2.2
7.2-AC-17	**GDC 29**, "Protection Against Anticipated Operational Occurrences."	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.2-AC-18	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.2-AC-19	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been	N/A-COL	N/A

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
7.2-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1
7.2-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)(2)**.	N/A-OTHER (Using IEEE-603)	N/A
7.2-SAC-03	**IEEE Std 7-4.3.2-2003**, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by **Regulatory Guide 1.152**, Revision 2, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” provides guidance on applying the safety system criteria to computer-based safety systems. **SRP Appendix 7.1-D** provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
7.2-SAC-04	**Item II.Q,** “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” **of the Staff Requirements Memorandum on SECY-93-087**, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs,” provides guidance on Diversity and Defense-in-Depth. SRP **BTP 7-19** provides additional guidance.	Y (Per AREVA Topical Report ANP-10284)	7.1.2.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 7.3	**Engineered Safety Features Systems (R5, 03/2007)**		
7.3-AC-01	**10 CFR 50.55a(a)(1)**, "Quality Standards."	Y	7.1.2.1 7.8.2.1
7.3-AC-02	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant specific licensing basis.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.3-AC-03	**10 CFR 50. Appendix A, General Design Criterion (GDC) 1**, "Quality Standards and Records."	Y	7.1.2.2 7.8.2.1
7.3-AC-04	**GDC 2**, "Design Basis for Protection Against Natural Phenomena."	Y	7.1.2.2
7.3-AC-05	**GDC 4**, "Environmental and Missile Design Basis."	Y	7.1.2.2
7.3-AC-06	**GDC 10**, "Reactor Design."	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.3-AC-07	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.3-AC-08	**GDC 15**, "Reactor Coolant System Design."	Y (Per AREVA Topical Report ANP-	7.1.2.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
		10275)	
7.3-AC-09	**GDC 16**, “Containment Design.”	Y	7.1.2.2
7.3-AC-10	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.3-AC-11	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.3-AC-12	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.3-AC-ESFAS-01	**10 CFR 50.34(f)** “Additional TMI-Related Requirements,” or equivalent TMI action requirements imposed by Generic Letters.		
	(2)(v), “Bypass and Inoperable Status Indication.”	Y	Table 7.1-3 7.1.2.1
	(2)(xii), “Auxiliary Feedwater System Automatic Initiation and Flow	Y	Table 7.1-3

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Indication.”		7.1.2.1
	(2)(xiv), “Containment Isolation Systems.”	Y	Table 7.1-3 7.1.2.1
7.3-AC-ESFAS-02	**GDC 20**, “Protection Systems Function.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.3-AC-ESFAS-03	**GDC 21**, “Protection System Reliability and Testability.”	Y	7.1.2.2
7.3-AC-ESFAS-04	**GDC 22**, “Protective System Independence.”	Y	7.1.2.2
7.3-AC-ESFAS-05	**GDC 23**, “Protection System Failure Modes.”	Y	7.1.2.2
7.3-AC-ESFAS-06	**GDC 24**, “Separation of Protection and Control Systems.”	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.3-AC-ESFAS-07	**GDC 29**, “Protection against Anticipated Operational Occurrences.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.3-AC-ESF-01	**GDC 33**, “Reactor Coolant Makeup.”	Y	7.1.2.2
7.3-AC-ESF-02	**GDC 34**, “Residual Heat Removal.”	Y	7.1.2.2
7.3-AC-ESF-03	**GDC 35**, “Emergency Core Cooling.”	Y	7.1.2.2

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
7.3-AC-ESF-04	**GDC 38**, “Containment Heat Removal.”	Y	7.1.2.2
7.3-AC-ESF-05	**GDC 41**, “Containment Atmosphere Cleanup.”	Y	7.1.2.2
7.3-AC-ESF-06	**GDC 44**, “Cooling Water.”	Y	7.1.2.2
7.3-SAC- 01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1
7.3-SAC- 02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.3-SAC- 03	**IEEE Std 7-4.3.2-2003**, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by **Regulatory Guide 1.152**, Revision 2, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants,” provides guidance on applying the safety system criteria to computer-based safety systems. **SRP Appendix 7.1-D** provides SRP acceptance criteria for safety and protection systems using digital computer-based technology.	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
7.3-SAC- 04	**Item II.Q**, “Defense Against Common-Mode Failures in Digital Instrument and Control Systems,” **of the Staff Requirements Memorandum on SECY-93-087**, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor	Y (Per AREVA Topical Report ANP-10284)	7.1.2.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP **BTP 7-19** provides additional guidance.		
SRP 7.4	**Safe Shutdown Systems (R5, 03/2007)**		
7.4-AC-01	**10 CFR 50.55a(a)(1)**, "Quality Standards."	Y	7.1.2.1 7.8.2.1
7.4-AC-02	**10 CFR 50.55a(h)**, "Protection Systems and Safety Systems," requires compliance with **IEEE Std 603-1991**, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." For safe shutdown systems that are not safety systems as defined by IEEE Std 603-1991 and that are isolated from safety systems, the applicable requirements of 10 CFR 50.55a(h) are IEEE Std 279-1971 Clause 4.7, "Control and Protection System Interaction" IEEE Std 603-1991 Clause 5.6.3, "Independence Between Safety Systems and Other Systems;" and IEEE Std 603-1991, Clause 6.3, "Interaction Between the Sense and Command Features and Other Systems."	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.4-AC-03	**10 CFR 50.34(f)(2)(xx)**, "Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves," or equivalent TMI action plan requirements imposed by Generic Letters.	Y	7.1.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.4-AC-04	**10 CFR 50, Appendix A, GDC 1**, “Quality Standards and Records.”	Y	7.1.2.2 7.8.2.1
7.4-AC-05	**GDC 2**, “Design Bases for Protection against Natural Phenomena.”	Y	7.1.2.2
7.4-AC-06	**GDC 4**, “Environmental and Missile Design Bases.”	Y	7.1.2.2
7.4-AC-07	**GDC 13**, “Instrumentation and Control.”	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.4-AC-08	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.4-AC-09	**GDC 24**, “Separation of Protection and Control Systems.”	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.4-AC-10	**GDC 34**, “Residual Heat Removal.”	Y	7.1.2.2
7.4-AC-11	**GDC 35**, “Emergency Core Cooling.”	Y	7.1.2.2
7.4-AC-12	**GDC 38**, “Containment Heat Removal.”	Y	7.1.2.2
7.4-AC-13	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.4-AC-14	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.4-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1
7.4-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.4-SAC-03	**SRP Appendix 7.1-D** provides SRP acceptance criteria for the digital I&C compliance with **IEEE Std 7-4.3.2-2003**, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by **Regulatory Guide 1.152**, Revision 2, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants.”	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 7.5	**Information Systems Important to Safety (R5, 03/2007)**		
	Requirements applicable to accident monitoring instrumentation		
7.5-AC-01	**10 CFR 50.55a(a)(1)**.	Y	7.1.2.1 7.8.2.1
7.5-AC-02	**10 CFR 50.55a(h)**, “Protection Systems and Safety Systems,” requires compliance with **IEEE Std. 603-1991**, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.” For accident monitoring instrumentation isolated from the protection system, the applicable requirements of 10 CFR 50.55a(h) for IEEE Std. 279-1971 is Clause 4.7, “Control and Protection System Interaction,” and for IEEE Std. 603-1991 are Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” and Clause 6.3, “Interaction Between the Sense and Command Features and Other Systems.”	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.1 7.1.2.6 7.5.2.2
7.5-AC-03	**10 CFR 50.34(f)**, “Additional TMI-Related Requirements,” or equivalent TMI action plan requirements imposed by orders; For Part 50 applicants not listed in 10 CFR 50.54(b), the applicable provisions of 10 CFR 50.34(f) will be made a requirement during the licensing process. The following portions of 10 CFR 50.34(f) apply to accident monitoring		

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	U.S. EPR Assessment	FSAR Section(s)
	instrumentation.		
	(2)(v), regarding bypass and inoperable status indication.	Y	Table 7.1-3 7.1.2.1
	(2)(xi), regarding direct indication of relief and safety valve position.	Y	Table 7.1-3 7.1.2.1
	(2)(xii), regarding auxiliary feedwater system flow indication (applicable to PWRs only).	Y	Table 7.1-3 7.1.2.1
	(2)(xvii), regarding accident monitoring instrumentation.	Y	Table 7.1-3 7.1.2.1
	(2)(xviii), regarding inadequate core cooling instrumentation.	Y	Table 7.1-3 7.1.2.1
	(2)(xix), regarding instruments for monitoring plant conditions following core damage.	Y	Table 7.1-3 7.1.2.1
	(2)(xx), regarding power for pressurizer level indication (applicable to PWRs only)	Y	Table 7.1-3 7.1.2.1
	(2)(xxiv), regarding central reactor vessel water level recording (applicable to BWRs only).	N/A-BWR	N/A
7.5-AC-04	**10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, “Quality Standards and Records.”	Y	7.1.2.2 7.8.2.1
7.5-AC-05	**GDC 2**, “Design Basis for Protection Against Natural Phenomena” (applicable to channels classified as Category 1 or 2 in **Regulatory Guide 1.97**, Revisions 2 and 3, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During	Y	7.1.2.2 7.1.2.4 7.1.2.5

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and Following an Accident," or to channels classified as Types A, B, C, or D in Regulatory Guide 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants").		7.5.2.2
7.5-AC-06	**GDC 4**, "Environmental and Missile Design Basis" (applicable to channels classified as Category 1 or 2 in **Regulatory. Guide 1.97**, Revisions 2 or 3, or as Type A, B, C, or D in Regulatory Guide 1.97, Revision 4).	Y	7.1.2.2 7.1.2.4 7.1.2.5 7.5.2.2
7.5-AC-07	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.5-AC-08	**GDC 19**, "Control Room."	Y	7.1.2.2 7.8.2.1
7.5-AC-09	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
	Requirements applicable to bypassed and inoperable status indication		
7.5-AC-10	**10 CFR 50.55a(a)(1)**, "Quality Standards."	Y	7.1.2.1 7.8.2.1
7.5-AC-11	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with **IEEE Std. 603-1991** and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272	7.1 7.1.2 7.1.2.1 7.1.2.6

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	comply instead with their plant licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971. For BISI, the applicable requirements for IEEE Std. 279-1974 is Clause 4.13, “Indication of Bypasses,” and for IEEE Std. 603-1991 is Clause 5.8.3, “Indication of Bypasses.” For BISI that are isolated from safety systems the requirements for IEEE Std. 279-1971 is Clause 4.7, “Control and Protection System Interaction,” and for IEEE Std. 603-1991 are Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” and Clause 6.3, “Interaction Between the Sense and Command Features and Other Systems.”	ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.5.2.2
7.5-AC-12	**10 CFR 50.34(f)(2)(v)**, “Additional TMI-Related Requirements” - bypass and inoperable status indication, or equivalent TMI action plan requirements imposed by Orders; For Part 50 applicants not listed in 10 CFR 50.54(b), the applicable provisions of 10 CFR 50.34(b) will be made a requirement during the licensing process.	Y	7.1.2.1
7.5-AC-13	**GDC 1**, “Quality Standards and Records.”	Y	7.1.2.2 7.8.2.1
7.5-AC-14	**GDC 24**, “Separation of Protection and Control Systems.”	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
	Requirements applicable to annunciator systems		
7.5-AC-15	**10 CFR 50.55a(a)(1)**, “Quality Standards.”	Y	7.1.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
			7.8.2.1
7.5-AC-16	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with **IEEE Std. 603-1991** and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant specific-licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements in IEEE Std. 279-1971. For annunciators that are isolated from the protection system, the applicable requirement(s) of 10 CFR 50.55a(h) for IEEE Std. 279-1971 is Clause 4.7, "Control and Protection System Interaction," and for IEEE Std. 603-1991 are Clause 5.6.3, "Independence Between Safety Systems and Other Systems," and Clause 6.3, "Interaction Between the Sense and Command Features and Other Systems."	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.5-AC-17	**GDC 1**, "Quality Standards and Records."	Y	7.1.2.2 7.8.2.1
7.5-AC-18	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.5-AC-19	**GDC 19**, "Control Room."	Y	7.1.2.2 7.8.2.1
7.5-AC-20	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
			7.8.2.1
	Requirements applicable to the review of SPDS, ERF information systems, and ERDS information systems		
7.5-AC-21	**10 CFR 50.55a(a)(1)**, “Quality Standards.”	Y	7.1.2.1 7.8.2.1
7.5-AC-22	**10 CFR 50.55a(h)**, “Protection and Safety Systems,” requires compliance with **IEEE Std. 603-1991** and the correction sheet dated January 30, 1995. Nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971. For SPDS, ERF information systems, and ERDS information systems isolated from the protection system, the applicable requirements of 10 CFR 50.55a(h) for IEEE Std. 279-1971 is Clause 4.7, “Control and Protection System Interaction,” and for IEEE Std. 603-1991 are Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” and Clause 6.3, “Interaction Between the Sense and Command Features and Other Systems.”	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.5-AC-23	**GDC 1**, “Quality Standards and Records.”	Y	7.1.2.2 7.8.2.1
7.5-AC-24	**GDC 24**, “Separation of Protection and Control Systems.”	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Additional requirements applicable to any information system important to safety proposed for standard DC or COLs under 10 CFR 52.		
7.5-AC-25	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.5-AC-26	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.5-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
7.5-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.5-SAC-03	**SRP Appendix 7.1-D** provides SRP acceptance criteria for the application of the requirements of **IEEE Std. 603-1991** to digital I&C. **Appendix 7.1-D** discusses the application of the guidance in **IEEE Std. 7-4.3.2-2003**, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by **Regulatory Guide 1.152**, Revision 2.	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4. 7.2.2.3 7.3.2.3 7.6.2.1
7.5-SAC-04	**Item II.Q**, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," **of the Staff Requirements Memorandum on SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Diversity and Defense-in-Depth. SRP **BTP 7-19** provides additional guidance.	Y (Per AREVA Topical Report ANP-10284)	7.1.2.3
7.5-SAC-05	**Regulatory Guide 1.97**, Revision 2, 3, and 4, describe methods acceptable to the NRC staff for providing instrumentation to monitor variables for accident conditions. For plants with operating licenses issued before June 2006, Regulatory Guide 1.97, Revision 2 and 3, are still effective. Licensees of these plants may, however, convert to the criteria of Revision 4 or use the criteria of Revision 4 when performing modifications that do not involve a conversion. The guidance contained in Regulatory Position 1 of Regulatory Guide 1.97, Revision 4, should be followed in these cases. Plants that obtained an operating license after June 2006 should reference the guidance of Regulatory Guide 1.97, Revision 4. SRP **BTP 7-10** provides guidance on the application of Regulatory Guide 1.97.	Y	7.1.2.4 7.1.2.5 7.5.2.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 7.6	**Interlock Systems Important to Safety (R5, 03/2007)**		
7.6-AC-01	**10 CFR 50.55a(a)(1)**, “Quality Standards.”	Y	7.1.2.1 7.8.2.1
7.6-AC-02	**10 CFR 50.55a(h)**, “Protection and Safety Systems,” requires compliance with **IEEE Std 603-1991**, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with their plant-specific licensing basis. For nuclear power plants with construction permits issued after January 1, 1971, and before May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.” For interlock systems that are not safety systems as defined by IEEE Std 603-1991 and isolated from the safety systems, the applicable requirements of 10 CFR 50.55a(h) are IEEE Std 279-1971 Clause 4.7, “Control and Protection System Interaction,” IEEE Std 603-1991 Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” and IEEE Std 603-1991, Clause 6.3, “Interaction Between the Sense and Command Features and Other Systems.”	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.6-AC-03	**10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1**, “Quality Standards and Records.”	Y	7.1.2.2
7.6-AC-04	**GDC 2**, “Design Bases for Protection Against Natural Phenomena.”	Y	7.1.2.2
7.6-AC-05	**GDC 4**, “Environmental and Dynamic Effects Design Bases.”	Y	7.1.2.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.6-AC-06	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.6-AC-07	**GDC 19**, "Control Room."	Y	7.1.2.2 7.8.2.1
7.6-AC-08	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.6-AC-09	**10 CFR 50.34(f)(2)(v)**, "Additional TMI-Related Requirements, Bypass and Inoperable Status Indication," or equivalent TMI action requirements imposed by Generic Letters.	Y	7.1.2.1
7.6-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.6-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
7.6-AC-INTERLOCK-01	**GDC 10**, “Reactor Design.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.6-AC-INTERLOCK-02	**GDC 15**, “Reactor Coolant System Design.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.6-AC-INTERLOCK-03	**GDC 16**, “Containment Design.”	Y	7.1.2.2
7.6-AC-INTERLOCK-04	**GDC 28**, “Reactivity Limits.”	Y	7.1.2.2
7.6-AC-INTERLOCK-05	**GDC 33**, “Reactor Coolant Makeup.”	Y	7.1.2.2
7.6-AC-INTERLOCK-06	**GDC 34**, “Residual Heat Removal.”	Y	7.1.2.2
7.6-AC-INTERLOCK-07	**GDC 35**, “Emergency Core Cooling.”	Y	7.1.2.2
7.6-AC-INTERLOCK-08	**GDC 38**, “Containment Heat Removal.”	Y	7.1.2.2
7.6-AC-INTERLOCK-09	**GDC 41**, “Containment Atmosphere Cleanup.”	Y	7.1.2.2

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.6-AC-INTERLOCK-10	**GDC 44**, “Cooling Water.”	Y	7.1.2.2
7.6-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1
7.6-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.6-SAC-03	**SRP Appendix 7.1-D** provides SRP acceptance criteria for digital I&C compliance with **IEEE Std 7-4.3.2-2003**, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by **Regulatory Guide 1.152**, Revision 2, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants.”	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
SRP 7.7	**Control Systems (R5, 03/2007)**		
7.7-AC-01	**10 CFR 50.55a(a)(1)**.	Y	7.1.2.1 7.8.2.1
7.7-AC-02	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with **IEEE Std. 603-1991**, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with	Y (Per following Areva Topical Reports: EMF-2110-A	7.1 7.1.2 7.1.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with its plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Station." For control systems isolated from safety systems, the applicable requirements of 10 CFR 50.55a(h) are defined in IEEE Std. 279-1971 Clause 4.7, "Control and Protection System Interaction," IEEE Std. 603-1991 Clause 5.6.3, "Independence Between Safety Systems and Other Systems," and IEEE Std. 603-1991 Clause 6.3, "Interaction Between the Sense and Command Features and Other Systems."	ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.6 7.5.2.2
7.7-AC-03	**10 CFR 50.34(f)(2)(xxii)**, "Additional TMI-Related Requirements;" (applies only to B&W plants) or equivalent TMI action plan requirements imposed by Commission order.	Y	Table 7.1-3 7.1.2.1
7.7-AC-04	**10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, "Quality Standards and Records."	Y	7.1.2.2
7.7-AC-05	**GDC 10**, "Reactor Design."	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.7-AC-06	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.7-AC-07	**GDC 15**, "Reactor Coolant System Design."	Y	7.1.2.2

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
		(Per AREVA Topical Report ANP-10275)	
7.7-AC-08	**GDC 19**, "Control Room."	Y	7.1.2.2 7.8.2.1
7.7-AC-09	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.7-AC-10	**GDC 28**, “Reactivity Limits.”	Y	7.1.2.2
7.7-AC-11	**GDC 29**, “Protection Against Anticipated Operational Occurences.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.7-AC-12	**GDC 44**, “Cooling Water.”	Y	7.1.2.2
7.7-AC-13	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.7-AC-14	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide	N/A-COL	N/A

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
7.7-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**. Although compliance with IEEE Std. 603-1991 is required by 10 CFR 50.55a(h) only for safety systems, the criteria of IEEE Std. 603-1991 may be used as review guidance for any I&C system. Therefore, for control systems, the reviewer may use the concepts in IEEE Std. 603-1991 as a starting point.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1.2.1
7.7-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.7-SAC-03	**SRP Appendix 7.1-D** provides SRP acceptance criteria for digital I&C compliance with **IEEE Std. 7-4.3.2-2003**, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by Regulatory Guide 1.152, Revision 2, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants."	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
7.7-SAC-04	**Item II.Q**, "Defense Against Common-Mode Failures in Digital Instrument and Control Systems," **of the Staff Requirements Memorandum on SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," provides guidance on Defense-in-Depth and Diversity. SRP **BTP 7-19** provides additional guidance.	Y (Per AREVA Topical Report ANP-10284)	7.1.2.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 7.8	**Diverse Instrumentation and Control Systems (R5, 03/2007)**		
7.8-AC-01	**10 CFR 50.55a(a)(1)**, "Quality Standards."	Y	7.1.2.1 7.8.2.1
7.8-AC-02	**10 CFR 50.55a(h)**, "Protection and Safety Systems," requires compliance with **IEEE Std 603-1991**, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with the plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." For diverse actuation systems isolated from safety systems, the applicable requirements of 10 CFR 50.55a(h) are IEEE Std 279-1971, Clause 4.7, "Control and Protection System Interaction"; IEEE Std 603-1991, Clause 5.6.3, "Independence Between Safety Systems and Other Systems"; and IEEE Std 603-1991, Clause 6.3, "Interaction Between the Sense and Command Features and Other Systems."	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1 7.1.2.6 7.5.2.2
7.8-AC-03	**10 CFR 50, Appendix A, General Design Criterion (GDC) 1**, "Quality Standards and Records."	Y	7.1.2.2
7.8-AC-04	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
7.8-AC-05	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.8-AC-06	**GDC 24**, “Separation of Protection and Control Systems.”	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.8-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.8-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.8-AC-09	**10 CFR 50.62**, “Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants.”	Y	7.1.2.1 7.8.2.1
7.8-SAC-01	For plants with a digital RTS or ESFAS, the NRC position on D3 should be especially noted. This position is contained in **Item II.Q**, “Defense	Y (Per AREVA Topical Report ANP-	7.1.2.3

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	U.S. EPR Assessment	FSAR Section(s)
	Against Common-Mode Failures in Digital Instrument and Control Systems," **of the Staff Requirements Memorandum on SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." SRM requirements applicable to diverse I&C functions are as follows: "If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure [as the safety system], shall be required to perform either the same function [as the safety system function that is vulnerable to common mode failure] or a different function [that provides adequate protection]. The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary functions under the associated event conditions." "A set of displays and controls located in the main control room shall be provided for manual system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system[s] . . ."	10284)	
7.8-SAC-02	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284	7.1.2.1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
		ANP-10275)	
7.8-SAC-03	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.8-SAC-04	**SRP Appendix 7.1-D** provides SRP acceptance criteria for digital I&C compliance with **IEEE Std 7-4.3.2-2003**, “IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations,” as endorsed by **Regulatory Guide 1.152**, Revision 2.	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
SRP 7.9	**Data Communications Systems (R5, 03/2007)**		
7.9-AC-01	**10 CFR 50.55a(a)(1)**, “Quality Standards for Systems Important to Safety.”	Y	7.1.2.1 7.8.2.1
7.9-AC-02	**10 CFR 50.55a(h)**, “Protection and Safety Systems,” requires compliance with **IEEE Std. 603-1991**, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, the applicant/licensee may elect to comply instead with the plant-specific licensing basis. For nuclear power plants with construction permits issued between January 1, 1971, and May 13, 1999, the applicant/licensee may elect to comply instead with the requirements stated in IEEE Std. 279-1971, “Criteria for Protection Systems for Nuclear Power Generating Stations.” The minimum requirements that are applicable to all DCS are IEEE Std. 603-1991, Clause 5.6.3, “Independence Between Safety Systems and Other Systems,” or IEEE Std. 279-1971, Clause 4.7.2, “Isolation Devices,” or the plant-specific licensing basis, as defined by 10 CFR 50.55a(h), as noted above.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273 ANP-10281 ANP-10284 ANP-10275)	7.1 7.1.2 7.1.2.1. 7.1.2.6 7.5.2.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
7.9-AC-03	**10 CFR Part 50, Appendix A, General Design Criteria (GDC) 1**, "Quality Standards and Records."	Y	7.1.2.2
7.9-AC-04	**GDC 24**, "Separation of Protection and Control Systems."	Y	7.1.2.2 7.2.2.3 7.3.2.3 7.8.2.1
7.9-AC-PROTECTION-01	**10 CFR 50.34(f)(2)(v)**, regarding automatic indication of bypassed and inoperable status of safety system equipment. For Part 50 applicants not listed in 10 CFR 50.34(f), the provisions of 50.34(f) will be made a requirement during its licensing review.	Y	7.1.2.1
7.9-AC-PROTECTION-02	**10 CFR 50.55a(h)(2)**, "Protection Systems."	Y	7.1.2.1
7.9-AC-PROTECTION-03	**GDC 2**, "Design Basis for Protection Against Natural Phenomena."	Y	7.1.2.2
7.9-AC-PROTECTION-04	**GDC 4**, "Environmental and Dynamic Effects Design Basis."	Y	7.1.2.2
7.9-AC-PROTECTION-05	**GDC 13**, "Instrumentation and Control."	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.9-AC-PROTECTION-06	**GDC 21**, "Protection System Reliability and Testability."	Y	7.1.2.2
7.9-AC-PROTECTION-07	**GDC 22**, "Protection System Independence."	Y	7.1.2.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
7.9-AC-PROTECTION-08	**GDC 23**, “Protection System Failure Modes.”	Y	7.1.2.2
7.9-AC-PROTECTION-09	**GDC 29**, “Protection Against Anticipated Operational Occurrences.”	Y (Per AREVA Topical Report ANP-10275)	7.1.2.2
7.9-AC-SHUTDOWN-01	**10 CFR 50.55a(h)(3)**, “Safety Systems,” to the extent that the DCS supports the safety functions of these systems.	Y	7.1.2.1 7.8.2.1
7.9-AC-SHUTDOWN-02	**GDC 4**, “Environmental and Missile Design Basis.”	Y	7.1.2.2
7.9-AC-SHUTDOWN-03	**GDC 13**, “Instrumentation and Control.”	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.9-AC-SHUTDOWN-04	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.9-AC-CONTROL-01	**GDC 13**, “Instrumentation and Control.”	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.9-AC-CONTROL-02	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.9-AC-DIVERSE-01	**10 CFR 50.62**, “Requirements for the Reduction of Risk from Anticipated Transients without Scram.”	Y	7.1.2.1 7.8.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
7.9-AC-DIVERSE-02	**GDC 13**, “Instrumentation and Control.”	Y	7.1.2.2 7.5.2.1 7.8.2.1
7.9-AC-DIVERSE-03	**GDC 19**, “Control Room.”	Y	7.1.2.2 7.8.2.1
7.9-AC-PART 52-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
7.9-AC-PART 52-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
7.9-SAC-01	**SRP Appendix 7.1-C** provides SRP acceptance criteria for safety system compliance with **10 CFR 50.55a(h)**.	Y (Per following Areva Topical Reports: EMF-2110-A ANP-10272 ANP-10273	7.1.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
		ANP-10281 ANP-10284 ANP-10275)	
7.9-SAC-02	**SRP Appendix 7.1-B** provides SRP acceptance criteria for protection system compliance with **10 CFR 50.55a(h)**.	N/A-OTHER (Using IEEE-603)	N/A
7.9-SAC-03	**SRP Appendix 7.1-D** provides SRP acceptance criteria for digital I&C compliance with **IEEE Std. 7-4.3.2-2003**, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," as endorsed by **Regulatory Guide 1.152**, Revision 2.	Y (Per AREVA Topical Reports EMF-2110-A & ANP-10272)	7.1.2.4 7.2.2.3 7.3.2.3 7.6.2.1
SRP Appendix 7-A	**General Agenda, Station Site Visits (R5, 03/2007)**	N/A-COL	N/A
SRP Appendix 7-B	**Acronyms, Abbreviations, and Glossary**	N/A-INFO	N/A
BTP 7-1	**Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.a	
BTP 7-2	**Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.b	
BTP 7-3	**Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps out of Service (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.c	
BTP 7-4	**Guidance on Design Criteria for Auxiliary Feedwater Systems (Second R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.d	
BTP 7-5	**Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.e	

CHAPTER 7 Instrumentation and Controls			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
BTP 7-6	**Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode (R5, 03/2007)**	N/A-OTHER (Refer to SRP 7.1-T, 7.1-T-SAC-5.f)	N/A
BTP 7-8	**Guidance for Application of Regulatory Guide 1.22 (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.h	
BTP 7-9	**Guidance on Requirements for Reactor Protection System Anticipatory Trips (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.i	
BTP 7-10	**Guidance on Application of Regulatory Guide 1.97 (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.j Refer to SRP 7.5, 7.5-SAC-05 Refer to SRP 6.2.1.1.A 6.2.1.1.A-SAC-07 Refer to SRP 11.5, 11.5-SAC-01, 11.5-SAC-02, & 11.5-SAC-04	
BTP 7-11	**Guidance on Application and Qualifications of Isolation Devices (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.k	
BTP 7-12	**Guidance on Establishing and Maintaining Instrument Setpoints (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.l	
BTP 7-13	**Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.m	
BTP 7-14	**Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.n	
BTP 7-17	**Guidance on Self-Test and Surveillance Test Provisions (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.q	
BTP 7-18	**Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems (R5,**	Refer to SRP 7.1-T, 7.1-T-SAC-5.r	

CHAPTER 7 Instrumentation and Controls			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	03/2007)		
BTP 7-19	**Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.s Refer to SRP 7.2, 7.2-SAC-04 Refer to SRP 7.3, 7.3-SAC-04 Refer to SRP 7.5, 7.5-SAC-04 Refer to SRP 7.7, 7.7-SAC-04	
BTP 7-21	**Guidance on Digital Computer Real-Time Performance (R5, 03/2007)**	Refer to SRP 7.1-T, 7.1-T-SAC-5.u	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 8.1	**Electric Power – Introduction (R3, 03/2007)**		
8.1-AC-01	Table 8-1 of this SRP section lists the acceptance criteria the staff currently applies to electric power systems. Implementation of these criteria in accordance with applicable regulatory guides and branch technical positions will provide assurance that systems will perform their design safety functions when required.	Y (Refer to SRP 8.1-01-1.a through SRP 8.1-01-6.e below)	(As noted below)
8.1-AC-01.1.a	**GDC 2** - Design Bases for Protection Against Natural Phenomena	Y	8.1.4.3 8.2.2.1 8.3.1.2 8.3.2.2
8.1-AC-01.1.b	**GDC 4** - Environmental and Dynamic Effects Design Bases	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.1-AC-01.1.c	**GDC 5** - Sharing of Structures, Systems, and Components	Y	8.3.1.2 8.3.1.2 8.3.2.2
8.1-AC-01.1.d	**GDC 17** - Electric Power Systems	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.1-AC-01.1.e	**GDC 18** - Inspection and Testing of Electrical Power Systems	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.1-AC-01.1.f	**GDCs 33, 34, 35, 38, 41, and 44** - Inspection and Testing of Electrical Power	Y	8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Systems		
8.1-AC-01.1.g	**GDC 50** - Containment Design Bases	Y	8.3.1.2 8.3.2.2
8.1-AC-01.2.a.i	**10 CFR 50.34(f)(2)(v)** - (Related to TMI Item I.D.3)	Y	8.3.1.2
8.1-AC-01.2.a.ii	**10 CFR 50.34(f)(2)(xiii)** - (Related to TMI Item II.E.3.1)	Y	8.3.1.2
8.1-AC-01.2.a.iii	**10 CFR 50.34(f)(2)(xx)** - (Related to TMI Item II.G.1)	Y	8.3.1.2
8.1-AC-01.2.b	**10 CFR 50.55a** - Codes and Standards		8.3.2.2
8.1-AC-01.2.c	**10 CFR 50.63** - Loss of All Alternating Current Power	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2
8.1-AC-01.2.d	**10 CFR 50.65(a)(4)** - Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	N/A-COL	8.2.2.8 8.3.1.2 8.3.2.2
8.1-AC-01.2.e	**10 CFR 52.47(b)(1)** - Contents of Applications	ITAAC	Tier 1
8.1-AC-01.2.f	**10 CFR 52.80(a)** - Contents of Applications; Additional Technical Information	N/A-COL	N/A
8.1-AC-01.3.a	**RG 1.6** - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.1-AC-01.3.b	**RG 1.9** - Application, and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	N/A-COL	8.4.2.5
8.1-AC-01.3.c	**RG 1.32** - Criteria for Power Systems for Nuclear Power Plants	Y	8.1.4.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
			8.2.2.4 8.3.1.2 8.3.2.2
8.1-AC-01.3.d	**RG 1.47** - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Y	8.3.1.2 8.3.2.2
8.1-AC-01.3.e	**RG 1.53** - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Y	8.1.4.3 8.3.1.1 8.3.1.2 8.3.2.2
8.1-AC-01.3.f	**RG 1.63** - Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	Y	8.1.4.3 8.3.1.2 8.3.1.2 8.3.2.2
8.1-AC-01.3.g	**RG 1.75** - Physical Independence of Electric Systems	Y	8.1.4.3 8.3.1.1 8.3.1.2 8.3.2.2
8.1-AC-01.3.h	**RG 1.81** - Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	Y	8.3.1.2 8.3.2.2
8.1-AC-01.3.i	**RG 1.106** - Thermal Overload Protection for Electric Motors on Motor-Operated Valves	Y	8.3.1.1
8.1-AC-01.3.j	**RG 1.118** - Periodic Testing of Electric Power and Protection Systems	Y	8.1.4.3 8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
8.1-AC-01.3.k	**RG 1.128** - Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Y	8.1.4.3 8.3.2.1 8.3.2.2
8.1-AC-01.3.l	**RG 1.129** - Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	Y	8.1.4.3 8.3.2.2 8.3.2.4
8.1-AC-01.3.m	**RG 1.153** - Criteria for Safety Systems	Y	8.3.1.2 8.3.2.2
8.1-AC-01.3.n	**RG 1.155** - Station Blackout	Y	8.4.2.6
8.1-AC-01.3.o	**RG 1.160** - Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	N/A-COL	17.6
8.1-AC-01.3.p	**RG 1.182** - Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants	N/A-COL	17.6
8.1-AC-01.3.q	**RG 1.204** - Guidelines for Lightning Protection of Nuclear Power Plants	Y	8.1.4.3 8.2.2.5 8.3.1.2 8.3.1.3
8.1-AC-01.3.r	**RG 1.206** - Combined License Applications for Nuclear Power Plants (LWR Edition)	Y	1.1.6.1
8.1-AC-01.4.a	**BTP 8-1** - Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	Y	7.5.2.2 7.6.1.2 8.1.4.3 8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
8.1-AC-01.4.b	**BTP 8-2** - Use of Diesel-Generator Sets for Peaking	Y	8.1.4.3 8.3.1.2
8.1-AC-01.4.c	**BTP 8-3** - Stability of Offsite Power Systems	Y	8.1.4.3 8.2.2.9
8.1-AC-01.4.d	**BTP 8-4** - Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves	Y	8.1.4.3 8.3.1.2
8.1-AC-01.4.e	**BTP 8-5** - Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.1-AC-01.4.f	**BTP 8-6** - Adequacy of Station Electric Distribution System Voltages	Y	8.1.4.3 8.2.2.10 8.3.1.1 8.3.1.2
8.1-AC-01.4.g	**BTP 8-7** - Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	Y	8.1.4.3 8.3.1.2
8.1-AC-01.5.a	**NUREG-0718, Revision 1** - Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License	Y	8.3.1.2 8.3.2.2
8.1-AC-01.5.b	**NUREG-0737** - Clarification of TMI Action Plan Requirements	Y	8.3.1.2
8.1-AC-01.5.c	**NUREG/CR-0660** - Enhancement of Onsite Diesel Generator Reliability	Y	8.3.1.1 8.3.1.2
8.1-AC-01.5.d	**NUREG-1793** - Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design	N/A-VEN	Table 8.1-1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
8.1-AC-01.6.a	**SECY-90-016** - Evolutionary Light Water Reactor Certification Issues and Their Relationships to Current Regulatory Requirements,1990	Y	8.4.2.6
8.1-AC-01.6.b	**SECY-94-084** - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, 1994	N/A-PAS	Table 8.1-1
8.1-AC-01.6.c	**SECY-95-132** - Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs, 1995	N/A-PAS	Table 8.1-1
8.1-AC-01.6.d	**SECY-91-078** - EPRI's Requirements Document and Additional Evolutionary LWR Certification Issues, 1991	Y	8.3.1.2
8.1-AC-01.6.e	**SECY-05-0227** - Final Rule – AP1000 Design Certification, 2005	N/A-VEN	Table 8.1-1
8.1-AC-02	**SRP Sections 8.2, 8.3.1, 8.3.2, and 8.4** detail the specific acceptance criteria presented in Table 8-1. Each SRP section also describes the technical rationale for applying these criteria to reviews of electrical power systems.	Y	8.2 8.3.1 8.3.2 8.4
8.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
8.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC's regulations.		
SRP 8.2	**Offsite Power System (R4, 03/2007)**		
8.2-AC-01	**General Design Criterion (GDC) 2**, found in Appendix A to 10 CFR Part 50, as it relates to structures, systems, and components of the offsite power systems being capable of withstanding the effects of natural phenomena (excluding seismic, tornado, and flood) without the loss of the capability to perform their safety functions.	Y	8.2.2.1 8.3.1.2 8.3.2.2
8.2-AC-02	**GDC 4** as it relates to structures, systems, and components of the offsite power systems being protected against dynamic effects, including the effects of missile that may result from equipment failures during normal operation, maintenance, testing, and postulated accidents.	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.2-AC-03	**GDC 5** as it relates to sharing of structures, systems, and components of the preferred power systems.	Y	8.3.1.2 8.3.2.2
8.2-AC-04	**GDC 17** as it relates to the preferred power system's (i) capacity and capability to permit functioning of structures, systems, and components important to safety; (ii) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies; (iii) physical independence; (iv)	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.2-AC-05	**GDC 18** as it relates to inspection and testing of the offsite power systems.	Y	8.2.2.5 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
8.2-AC-06	**GDCs 33, 34, 35, 38, 41, and 44** as they relate to the operation of the offsite electric power system, encompassed in GDC 17, to ensure that the safety functions of the systems described in GDC's 33, 34, 35, 38, 41, and 44 are accomplished.	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.2-AC-07	**10 CFR 50.63** as it relates to an AAC power source (as defined in 10 CFR 50.2) provided for safe shutdown (non-DBA) in the event of a station blackout.	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2
8.2-AC-08	**10CFR 50.65(a)(4)** as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. These activities include, but are not limited to, surveillances, post-maintenance testing, and corrective and preventive maintenance. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17. Programs for incorporation of requirements into appropriate procedures are reviewed under SRP Chapter 13.	Y	8.2.2.8 8.3.1.2 8.3.2.2
8.2-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
8.2-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
8.2-SAC-01	**GDC 2** is satisfied as it relates to structures, systems, and components of the offsite power system being capable of withstanding the effects of natural phenomena such as high and low atmospheric temperatures, high wind, rain, lightning discharges, ice and snow conditions, and weather events causing regional effects as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering.	Y	8.1.4.3 8.2.2.1 8.3.1.2 8.3.2.2
8.2-SAC-02	**GDC 4** is satisfied as it relates to structures, systems, and components of the offsite power system being protected against dynamic effects, including the effects of missile that may result from equipment failures during normal operation, maintenance, testing, and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering.	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.2-SAC-03	**GDC 5** is satisfied as it relates to: sharing of structures, systems, and components of the preferred power systems; guidelines of **Regulatory Guide 1.32** as related to its endorsement of **Section 7 of IEEE Std 308**, relating to sharing of structures, systems, and components of the Class 1E power system at multi-unit stations; and guidance related to the sharing of structures, systems, and components of the offsite power system (preferred	Y	8.1.4.3 8.2.2.4 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	power supply) at multi-unit stations, previously addressed in the 1980 and earlier versions of IEEE Std 308, but now covered in the industry standard for preferred power supply.		
8.2-SAC-04	**GDC 17** is satisfied as it relates to the preferred power system's (i) capacity and capability to permit functioning of structures, systems, and components important to safety; (ii) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies; (iii) physical independence; (iv) availability and the guidelines of **Regulatory Guide 1.32** (see also **IEEE Std 308**) as related to the availability and number of immediate access circuits from the transmission network; and (v) capability to meet the guidelines of **Appendix A to SRP Section 8.2** as related to acceptability of generator circuit breakers and generator load break switches. For evolutionary light water reactor design applications, as documented in **SECY 94-084** for designs such as the CE-ABB System 80+ and the GE ABWR, the design should provide at least one offsite circuit to each redundant safety division that is supplied directly from an offsite power source with no intervening non-safety buses, thereby permitting the offsite source to supply power for safety buses in the event the non-safety bus(es) fails. The design should also include an alternate power source to non-safety loads, unless it can be demonstrated that existing design margins will ensure that transients for loss of non-safety power events are no more severe than those associated with the turbine-trip-only event specified in current plant designs (References 33 and 35). These issues are reviewed in detail in SRP	Y	8.2.2.4 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Section 8.3.1. 8.2-8 For passive reactor design applications, the passive safety-related systems only require electric power for valves and related instrumentation, which can be supplied from the onsite Class 1E batteries and associated dc and ac distribution systems. The acceptability of this design for the AP 1000 is documented in SECY-05-0227 and FSER NUREG-1793. If no offsite power is available, it is expected that the non-safety-related diesel generators would be available for important plant functions, but this non-safety-related ac power is not relied on to maintain core cooling or containment integrity. Therefore, this passive reactor design supports an exemption to the requirement of GDC 17 for two physically independent offsite circuits, by providing safety-related passive safety systems for core cooling and containment integrity (see also References 33, 34, 35). However, one offsite power source with sufficient capacity and capability from the transmission network must be provided to power the safety-related systems and all other auxiliary systems under normal, abnormal, and accident conditions. The offsite power source should be designed to minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions.		
8.2-SAC-05	**GDC 18** is satisfied as it relates to the inspection and testing of the offsite electric power system.	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.2-SAC-06	**GDCs 33, 34, 35, 38, 41, and 44** are satisfied as they relate to the operation of the offsite electric power system, encompassed in **GDC 17**, to ensure that the safety functions of the systems described in GDC's 33, 34, 35, 38, 41, and 44 are accomplished, assuming a single failure where applicable.	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.2-SAC-07	**10 CFR 50.63** is satisfied as it relates to an AAC power source (as defined in	Y	8.2.2.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	10 CFR 50.2) provided for safe shutdown in the event of a station blackout (non-DBA), and the guidelines of **Regulatory Guide 1.155** are followed as they relate to the adequacy of the AAC source and the independence of the AAC power source from the offsite power system and onsite power system and sources. Except for passive reactor designs described in subsection II(2) above, new applications must provide an adequate AAC source of diverse design (with respect to ac onsite emergency sources) that is consistent with the guidance in **Regulatory Guide 1.155** and capable of powering at least one complete set of normal safe shutdown loads. These issues are reviewed in detail in **SRP Section 8.4**.		8.3.1.2 8.3.2.2 8.4.2.2 8.4.2.6
8.2-SAC-08	**10 CFR 50.65, Section 50.65(a)(4)**, as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: A. **Regulatory Guide 1.160**, as related to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). B. **Regulatory Guide 1.182**, as related to implementing the provisions of 10 CFR 50.65 (a)(4) by endorsing **Section 11 to NUMARC 93-01**, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.	N/A-COL	8.2.2.8 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 8.3.1	**A-C Power Systems (Onsite) (R3, 03/2007)**		
8.3.1-AC-01	**GDC 2** as it relates to SSCs of the ac power system being capable of withstanding the effects of natural phenomena without the loss of the capability to perform their safety functions.	Y	8.1.4.3 8.2.2.1 8.3.1.2 8.3.2.2
8.3.1-AC-02	**GDC 4** as it relates to SSCs of the ac power system being capable of withstanding the effects of missiles and environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.3.1-AC-03	**GDC 5** as it relates to sharing of SSCs of the ac power systems.	Y	8.3.1.2 8.3.2.2
8.3.1-AC-04	**GDC 17** as it relates to the onsite ac power system's (a) capacity and capability to permit functioning of SSCs important to safety; (b) independence, redundancy, and testability to perform its safety function assuming a single failure; and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.3.1-AC-05	**GDC 18** as it relates to inspection and testing of the onsite power systems.	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.3.1-AC-06	**GDCs 33, 34, 35, 38, 41, and 44** as they relate to the operation of the onsite electric power system, encompassed in **GDC 17**, to ensure that the safety functions of the systems described in GDCs 33, 34, 35, 38, 41, and 44 are accomplished.	Y	8.2.2.4 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
8.3.1-AC-07	**GDC 50** as it relates to the design of containment electrical penetrations containing circuits of the ac power system and the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations.	Y	8.3.1.2 8.3.2.2
8.3.1-AC-08	**10 CFR 50.63**, as it relates to the establishment of a reliability program for emergency onsite ac power sources and the use of the redundancy and reliability of diesel generator units as a factor in limiting the potential for station blackout events.	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2
8.3.1-AC-09	**10 CFR 50.65, Section 50.65(a)(4)**, as it relates to the requirements to assess and manage the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. Acceptance is based on meeting the following specific guidelines: A. **Regulatory Guide 1.160**, as it relates to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase loss of offsite power (LOOP) frequency, or reduce the capability to cope with a LOOP or station blackout (SBO)). B. **Regulatory Guide 1.182**, as it relates to implementing the provisions of 10 CFR 50.65 (a)(4) by endorsing **Section 11 to NUMARC 93-01**, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.	N/A-COL	8.2.2.8 8.3.1.2 8.3.2.2
8.3.1-AC-10	**10 CFR 50.55a(h)** as it relates to protection systems for plants with	Y	8.1.4.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	construction permits issued after January 1, 1971, but before May 13, 1999, which must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or IEEE Std. 603-1991, "Criteria for Safety Systems for 8.3.1-13 Revision 3 - March 2007 Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std 279-1971. Nuclear power plants with applications filed on or after May 13, 1999 for preliminary and final design approvals (**10 CFR Part 52 Appendix O**), design certification, construction permits, operating licenses, and combined licenses that do not reference a final design approval or design certification must meet the requirements for safety systems in **IEEE Std 603-1991 and the correction sheet dated January 30, 1995**.		8.3.1.2 8.3.2.2
8.3.1-AC-11	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
8.3.1-AC-12	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
8.3.1-SAC-01	**GDC 2** is satisfied as it relates to SSCs of the onsite ac power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapter 3 of the SAR, and reviewed by the organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering.	Y	8.1.4.3 8.2.2.1 8.3.1.2 8.3.2.2
8.3.1-SAC-02	**GDC 4** is satisfied as it relates to SSCs of the ac power system being capable of withstanding the effects of missiles and environmental conditions associated with normal operation and postulated accidents, as established in Chapter 3 of the SAR and	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.3.1-SAC-03	**GDC 5** is satisfied as it relates to the sharing of SSCs of the ac power system and the following guidelines: A. **Regulatory Guide 1.32**, as it relates to the sharing of SSCs of the Class 1E power system at multi-unit stations. B. **Regulatory Guide 1.81**, as it relates to the sharing of SSCs of the ac power system, **positions C.2 and C.3**.	Y	8.1.4.3 8.2.2.4 8.3.1.2 8.3.2.2
		N/A-OTHER (for RG 1.81 per Single Unit Design)	8.1.4.3 8.2.2.4 8.3.1.2 8.3.2.2
8.3.1-SAC-04	**GDC 17** is satisfied as it relates to the onsite ac power system's: (a) capacity and capability to permit functioning of SSCs important to safety; (b) independence, redundancy, and testability to perform its safety function assuming a single failure; and (c) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or	Y	8.2.2.4 8.3.1.2 8.3.2.2 8.4.2.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 **Electric Power**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network. Acceptance is based on meeting the following specific guidelines: A. **Regulatory Guide 1.6**, as it relates to the independence of the onsite ac power system, **positions D.1, D.2, D.4, and D.5**. B. **Regulatory Guide 1.9** (see also **IEEE Std 387**). C. **Regulatory Guide 1.32** (see also **IEEE Std 308**), as it relates to design criteria for onsite ac power systems. D. **Regulatory Guide 1.53** (see also **IEEE Stds** 279 and **603**), as it relates to the application of the single-failure criterion to safety systems. E. **Regulatory Guide 1.75** (see also **IEEE Std 384**), as it relates to the onsite ac power system. F. **Regulatory Guide 1.153** (see also **IEEE Std 603**), as it relates to criteria for electrical portions of safety-related systems. G. **Regulatory Guide 1.155**, as it relates to the use of onsite emergency ac power sources for station blackout. H. **Regulatory Guide 1.204** (see also **IEEE Stds 665, 666, 1050, and C62.23**), as it relates to the lightning and surge protection for the onsite ac power system. I. **NUREG/CR-0660** is incorporated as it relates to the following recommendations: i. The diesel generator sets should be capable of operation at less than full load for extended periods of time without degradation of performance or reliability. With offsite power available, no-load operation of the diesel generators will occur following a safety injection signal. Extended no-load operation of this equipment should		8.4.2.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	be minimized. Operating procedures should be provided that limit extended no-load operation of the diesel generators. The procedures should include loading the diesel engine to a minimum of 25% of full load for 1 hour after 8 hours of continuous no-load operation or to a load as recommended by the engine manufacturer. ii. A complete formal training program should be provided for all personnel who will be responsible for the maintenance and availability of the diesel generators. The depth and quality of training shall be at least equivalent to that provided by major diesel engine manufacturers' training programs. iii. A preventive maintenance program should be provided which encompasses investigative testing of components which have a history of repeated malfunctioning and a plan for the replacement of those components that require constant attention and repair with other products of proven reliability. iv. Repair and maintenance procedures should provide for a final equipment check prior to an actual start-run-load test to ensure that all electrical circuits are functional (i.e., fuses in place, no loose wires, test leads removed, etc.) and all valves are in the proper position. The test procedure(s) should explicitly state that upon satisfactory test completion the diesel generator unit should be returned to a ready automatic standby service under the control of the control room operator. v. Except for sensors and other equipment that need to be directly mounted on the engine or associated piping, the controls and monitoring instruments should be installed on a free-standing, floor-mounted panel located on a vibration-free floor area.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	[NOTE: If the floor is not vibration free, the panel should be equipped with vibration mounts.] J. Acceptance criteria for the interface between the onsite ac power system and the offsite power system to satisfy the requirements of **GDC 17** in evolutionary light water reactor design applications are documented in **SECY-91-078**, which states that the design should include at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon the failure of any non-safety bus. The evolutionary light water reactor design should also include an alternate power source to non-safety loads, unless it can be demonstrated that existing design margins will ensure that transients for loss of non-safety power events are no more severe than those associated with the turbine-trip-only event specified in current plant designs. Passive light water reactor design applications provide passive safety systems that do not need Class 1E ac electric power, other than that provided by the Class 1E dc batteries and their inverters, to accomplish the plant's safety-related functions for 72 hours. However, in accordance with SECY-94-084, SECY-95-132, and Regulatory Guide 1.206 Section C.IV.10, ac power system features will be evaluated using the process for regulatory treatment of non-safety systems (RTNSS) for electrical distribution issues on passive designs. The AP1000 passive plant design certification, for example, includes an exemption to the requirement of GDC 17 for two physically independent offsite circuits, by providing safety-related passive safety systems for core cooling and containment integrity. However, even for this design, one offsite power source with		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	sufficient capacity and capability from the transmission network should be provided to power the safety-related systems and all other auxiliary systems under normal, abnormal, and accident conditions. The offsite power source should be designed to minimize to the extent practical the likelihood of its failure under normal, abnormal, and accident conditions. Detailed reviews of the offsite ac power system and its interface with the onsite power system for ALWR design applications are covered in Section 8.2, "Offsite Power System."		
8.3.1-SAC-05	**GDC 18** is satisfied as it relates to the testability of the onsite ac power system, and the following guidelines: A. **Regulatory Guide 1.32** (see also **IEEE Std 308**), as it relates to capability for testing of the onsite ac power system. B. **Regulatory Guide 1.47**, with respect to indicating the bypass or inoperable status of portions of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the system it actuates to perform their safety-related functions. C. **Regulatory Guide 1.118** (see also **IEEE Std 338**), as it relates to the capability for testing the onsite ac power system. D. **Regulatory Guide 1.153** (see also **IEEE Std 603**), as it relates to the onsite ac power system.	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.3.1-SAC-06	The design requirements for an onsite ac power supply for systems covered by **GDCs 33, 34, 35, 38, 41, and 44** are encompassed in **GDC 17**.	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.3.1-SAC-07	**GDC 50** is satisfied as it relates to the design of containment electrical	Y	8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	penetrations containing circuits of the ac power system, and the guidelines of **Regulatory Guide 1.63** are followed (see also **IEEE Stds 242, 317, and 741**), as related to the capability of electric penetration assemblies in containment structures to withstand a LOCA without loss of mechanical integrity and the external circuit protection for such penetrations, as well as to ensure that electrical penetrations will withstand the full range of fault current (minimum to maximum) available at the penetration.		8.3.2.2
8.3.1-SAC-08	**10 CFR 50.63**, as it relates to use of the redundancy and reliability of diesel generator units as a factor in limiting the potential for station blackout events. Acceptance is based on meeting the following specific guidelines: A. **Regulatory Guide 1.9**, as it relates to the adequacy of the diesel generator surveillance criteria provided to attain and maintain the target reliability levels of diesel generator units. B. **Regulatory Guide 1.155**, as it relates to use of the reliability of emergency onsite ac power sources as a factor in determining the coping duration for station blackout and the establishment of a reliability program for attaining and maintaining source target reliability levels. Determination of station blackout coping time is reviewed in detail in SRP Section 8.4. Except for passive reactor designs described in the acceptance criteria of SRP 8.3.1 subsection II.4.J above, new applications should provide an adequate AAC source of diverse design (with respect to onsite ac emergency sources) that is consistent with the guidance in **Regulatory Guide 1.155** and capable of powering at least one complete set of normal safe shutdown loads. These issues are reviewed in detail under **SRP Section 8.4**.	Y (for Design Criteria)	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2 8.4.2.5 8.4.2.6
		N/A –COL (for Construction, Operation, and Maintenance Criteria)	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2 8.4.2.5 8.4.2.6
8.3.1-SAC-09	**10 CFR 50.65, Section 50.65(a)(4)**, as it relates to the requirements to assess and manage the increase in risk that may result from proposed	Y (for Design Criteria)	8.2.2.8 8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	maintenance activities before performing the maintenance activities.		8.3.2.2
	Acceptance is based on meeting the following specific guidelines: A. **Regulatory Guide 1.160**, as it relates to the effectiveness of maintenance activities for onsite emergency ac power sources including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase loss of offsite power (LOOP) frequency, or reduce the capability to cope with a LOOP or station blackout (SBO)). B. **Regulatory Guide 1.182**, as it relates to implementing the provisions of 10 CFR 50.65 (a)(4) by endorsing **Section 11 to NUMARC 93-01**, “Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, February 22, 2000.	N/A –COL (for Construction, Operation, and Maintenance Criteria)	8.2.2.8 8.3.1.2 8.3.2.2
8.3.1-SAC-10	**10 CFR 50.55a(h)** as it relates to protection systems for plants with construction permits issued after January 1, 1971, but before May 13, 1999, which must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std 279-1971. Nuclear power plants with applications filed on or after May 13, 1999 for preliminary and final design approvals (**10 CFR Part 52, Appendix O**), design certification, construction permits, operating licenses, and combined licenses that do not reference a final design approval or design certification, must meet the requirements for	Y	8.1.4.3 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	safety systems in **IEEE Std 603-1991** and the correction sheet dated January 30, 1995.		
SRP 8.3.2	**D-C Power Systems (Onsite) (R3, 03/2007)**		
8.3.2-AC-01	**GDC 2**, as it relates to the ability of dc power system SSCs to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, as established in Chapter 3 of the SAR and reviewed by organizations with primary responsibility for the reviews of plant systems, civil engineering and geosciences, and mechanical engineering	Y	8.1.4.3 8.2.2.1 8.3.1.2 8.3.2.2
8.3.2-AC-02	**GDC 4**, as it relates to the ability of dc power system SSCs to withstand the effects of missiles and environmental conditions associated with normal operation and postulated accidents, as established in Chapter 3 of the SAR and reviewed by the organizations with primary responsibility for the reviews of plant systems, materials, and chemical engineering.	Y	8.1.4.3 8.3.1.2 8.3.2.2
8.3.2-AC-03	**GDC 5**, as it relates to sharing dc power system SSCs.	Y	8.3.1.2 8.3.2.2
8.3.2-AC-04	**GDC 17**, as it relates to (a) the capacity and capability of the onsite dc power system to enable the functioning of SSCs important to safety and (b) the independence and redundancy of the onsite dc power system in performing its safety function, assuming a single failure.	Y	8.2.2.4 8.3.1.2 8.3.2.2
8.3.2-AC-05	**GDC 18**, as it relates to the testability of the onsite dc power system.	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.3.2-AC-06	**GDCs 33, 34, 35, 38, 41, and 44** as they relate to the operation of the onsite electric power system, encompassed in **GDC 17** to ensure that the safety	Y	8.2.2.4 8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	functions of the systems described in GDCs 33, 34, 35, 38, 41, and 44 are accomplished.		8.3.2.2
8.3.2-AC-07	**GDC 50**, as it relates to the design of containment electrical penetrations containing circuits of safety-related and nonsafety-related dc power systems.	Y	8.3.1.2 8.3.2.2
8.3.2-AC-08	**10 CFR 50.63**, as it relates to the ability of the onsite dc power system to support the plant in withstanding or coping with, and recovering from, an SBO event.	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2
8.3.2-AC-09	**10 CFR 50.55a(h)**, as it relates to the incorporation of **Institute for Electrical and Electronics Engineers (IEEE) Standard (Std) 603-1991** (including the correction sheet dated January 30, 1995) and IEEE Std. 279 for protection and safety systems.	Y	8.3.2.2
8.3.2-AC-10	**10 CFR 50.65(a)(4)**, as it relates to the assessment and management, before the performance of maintenance activities, of the increase in risk that may result from proposed maintenance activities. These activities include, but are not limited to, surveillances, postmaintenance testing, and corrective and preventive maintenance. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.	Y	8.2.2.8 8.3.1.2 8.3.2.2
8.3.2-AC-11	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Energy Act, and the NRC's regulations;		
8.3.2-AC-12	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
8.3.2-SAC-01	**Regulatory Guide 1.6 positions D.1, D.3, and D.4**, as they relate to the independence between redundant onsite dc power sources and between their distribution systems.	Y	8.1.5 8.3.1.2 8.3.2.2
8.3.2-SAC-02	**Regulatory Guide 1.32**, as it relates to the design, operation, and testing of the safety-related portions of the onsite dc power system. Except for sharing of safety-related dc power systems in multi-unit nuclear power plants, RG 1.32 endorses **IEEE Std. 308-2001**.	Y	8.1.5 8.2.2.4 8.3.1.1 8.3.1.2 8.3.2.2
8.3.2-SAC-03	**Regulatory Guide 1.75**, as it relates to the physical independence of the circuits and electrical equipment that comprise or are associated with the onsite dc power system.	Y	8.1.5 8.3.1.1 8.3.1.2 8.3.2.2
8.3.2-SAC-04	**Regulatory Guide 1.81**, as it relates to the sharing of structures, systems, and components of the dc power system. Regulatory Position C.1 states that multi-unit sites should not share dc systems.	N/A-OTHER	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
8.3.2-SAC-05	**Regulatory Guide 1.128**, as it relates to the installation of vented lead-acid storage batteries in the onsite dc power system.	Y	8.1.5 8.3.2.2
8.3.2-SAC-06	**Regulatory Guide 1.129**, as it relates to maintenance, testing, and replacement of vented lead-acid storage batteries in the onsite dc power system.	Y	8.1.5 8.3.2.2
8.3.2-SAC-07	**Regulatory Guide 1.118**, as it relates to the capability to periodically test the onsite dc power system.	Y	8.1.5 8.3.1.2
8.3.2-SAC-08	**Regulatory Guide 1.153**, as it relates to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems of nuclear plants, including the application of the single failure criterion in the onsite dc power system. As endorsed by Regulatory Guide 1.153, **IEEE Std. 603** provides a method acceptable to the staff to evaluate all aspects of the electrical portions of the safety-related systems, including basic criteria for addressing single failures. However, as stated in 10 CFR 55a(h), all plants are not required to comply with IEEE Std. 603. Only applications filed on or after May 13, 1999, for preliminary and final design approvals (**10 CFR Part 52, Appendix O**), design certification, and construction permits; operating licenses and combined licenses that do not reference a final design approval or design certification must meet the requirements for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. Operating nuclear power plants are encouraged, but not required to, comply with IEEE Std. 603 for future system-level modifications.	Y	8.1.5 8.3.1.2 8.3.2.2
8.3.2-SAC-09	**Regulatory Guide 1.53**, as it relates to the application of the single-failure criterion.	Y	8.1.5 8.3.1.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
			8.3.1.2 8.3.2.2
8.3.2-SAC-10	**Regulatory Guide 1.63**, as it relates to the capability of electric penetration assemblies in containment structures to withstand a loss of coolant accident without loss of mechanical integrity and the external circuit protection for such penetrations.	Y	8.1.5 8.3.1.2 8.3.2.2
8.3.2-SAC-11	**Regulatory Guide 1.155**, as it relates to the capability and the capacity of the onsite dc power system for an SBO, including batteries associated with the operation of the alternate ac (AAC) power source(s) (if used).	Y	8.4.2.6
8.3.2-SAC-12	The guidelines of **Regulatory Guide 1.160**, as they relate to the effectiveness of maintenance activities for dc power systems. Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.	N/A-COL	17.6
8.3.2-SAC-13	The guidelines of **Regulatory Guide 1.182**, as they relate to conformance to the requirements of **10 CFR 50.65(a)(4)** for assessing and managing risk when performing maintenance.	N/A-COL	8.2.2.8 8.3.1.2 8.3.2.2.
SRP 8.4	**Station Blackout (03/2007)**		
8.4-AC-01	**GDC 17**, as it relates to (a) the capacity and capability of onsite and offsite power systems to permit functioning of SSCs important to safety in the event of anticipated operational occurrences and postulated accidents and (b) provisions to minimize the probability of losing electric power from the transmission network (grid) as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies. Plants not licensed in accordance with the GDC in	Y	8.2.2.4 8.3.1.2 8.3.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8
Electric Power

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Appendix A to 10 CFR Part 50 were licensed to satisfy plant-specific principal design criteria presented in the updated final		
8.4-AC-02	**GDC 18**, as it relates to periodic testing and inspection of offsite and onsite power systems important to safety.	Y	8.2.2.5 8.3.1.2 8.3.2.2
8.4-AC-03	**10 CFR 50.63**, as it relates to the capability to withstand and recover from an SBO.	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2
8.4-AC-04	**10 CFR 50.65(a)(4)**, as it relates to the assessment and management of the increase in risk that may result from proposed maintenance activities before performing the maintenance activities. These activities include, but are not limited to, surveillances, postmaintenance testing, and corrective and preventive maintenance. Compliance with the maintenance rule, including verification that appropriate maintenance activities are	Y	8.2.2.8 8.3.1.2 8.3.2.2
8.4-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
8.4-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
8.4-SAC-01	The guidelines of **RG 1.155**, as they relate to compliance to **10 CFR 50.63**. **NUMARC-8700, Revision 0**, also provides guidance acceptable to the staff for meeting these requirements. **Table 1 of RG 1.155** provides a cross-reference to NUMARC-8700, Revision 0, and notes when the RG takes precedence.	Y	8.2.2.7 8.3.1.2 8.3.2.2 8.4.2.2 8.4.2.6
8.4-SAC-02	The guidelines and criteria of **SECY-90-016** and **SECY-94-084** (Ref. 25), as they relate to the use of AAC power sources and RTNSS at plants provided with passive safety systems.	N/A-PAS	8.4.2.6
8.4-SAC-03	The guidelines of **RGs 1.9** (Ref. 6) **and 1.155**, as they relate to the reliability program implemented to ensure that the target reliability goals for onsite EAC power sources (typically diesel generator units) are adequately maintained.	N/A-COL	8.4.2.5
8.4-SAC-04	The guidelines of **RG 1.160** (Ref. 8), as they relate to the effectiveness of maintenance activities for onsite EAC power sources, including grid-risk-sensitive maintenance activities (i.e., activities that tend to increase the likelihood of a plant trip, increase LOOP frequency, or reduce the capability to cope with a LOOP or SBO). Compliance with the maintenance rule, including verification that appropriate maintenance activities are covered therein, is reviewed under SRP Chapter 17.	N/A-COL	17.6
8.4-SAC-05	The guidelines of **RG 1.182** (Ref. 9), as they relate to conformance to the requirements of **10 CFR 50.65(a)(4)** for assessing and managing risk when	N/A-COL	8.2.2.8 8.3.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 8 Electric Power			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	performing maintenance		8.3.2.2
SRP Appendix 8-A	General Agenda, Station Site Visits (R1, 03/2007)	N/A-COL	N/A
BTP 8-1	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.a	
BTP 8-2	Use of Diesel-Generator Sets for Peaking (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.b	
BTP 8-3	Stability of Offsite Power Systems (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.c	
BTP 8-4	Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.d	
BTP 8-5	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.e	
BTP 8-6	Adequacy of Station Electric Distribution System Voltages (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.f	
BTP 8-7	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status (R3, 03/2007)	Refer to SRP 8.1, 8.1-AC-01.4.g	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 9.1.1	**Criticality Safety of Fresh and Spent Fuel Storage and Handling (R3, 03/2007)**		
9.1.1-AC-01	**GDC 62**, as it relates to the prevention of criticality by physical systems or processes using geometrically safe configurations.	N/A-COL	9.1.1
9.1.1-AC-02	**10 CFR 50.68**, as it relates to preventing a criticality accident and to mitigating the radiological consequences of a criticality accident.	N/A-COL	9.1.1
9.1.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.1.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.1.1-SAC-01	The criteria for **GDC 62** are specified in **American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.1**, **ANSI/ANS 57.2**, and **ANSI/ANS 57.3**, as they relate to the prevention of criticality accidents in fuel storage and handling.	N/A-COL	9.1.1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 9.1.2	**New and Spent Fuel Storage (R4, 03/2007)**		
9.1.2-AC-01	**GDC 2 of Appendix A to 10 CFR Part 50** as to structures housing the facility and the facility itself withstanding the effects of natural phenomena like earthquakes, tornadoes, and hurricanes.	Y	9.1.2.1
9.1.2-AC-02	**GDC 4** as to structures housing the facility and the facility itself withstanding the effects of environmental conditions, externally-generated missiles, internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks so safety functions are not precluded.	Y	9.1.2.1
9.1.2-AC-03	**GDC 5** as to shared structures, components and systems (SSCs) important to safety performing required safety functions.	Y	9.1.2.1
9.1.2-AC-04	**GDC 61** as to the facility design for fuel storage and handling of radioactive materials.	Y	9.1.2.1
9.1.2-AC-05	**GDC 63** as to monitoring systems for detecting conditions that could cause the loss of decay heat removal capabilities for spent fuel assemblies, detecting excessive radiation levels, and initiating appropriate safety actions.	Y	9.1.2.1
9.1.2-AC-06	**10 CFR 20.1101(b)** as to radiation doses kept as low as reasonably achievable (ALARA).	Y	9.1.2.1
9.1.2-AC-07	**10 CFR 50.68** as to criticality monitoring or design to preclude criticality accidents.	N/A-COL	9.1.1 9.1.2
9.1.2-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance	ITAAC	Tier 1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.1.2-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.1.2-SAC-01	Acceptance for meeting the relevant aspect of **GDC 2** is based on compliance with **positions C.1 and C.2 of Regulatory Guide (RG) 1.13** and applicable portions of **RG 1.29**, and **RG 1.117**. For the spent fuel storage facility, additional guidance acceptable for meeting this criterion is found in **American Nuclear Society (ANS) 57.2, paragraphs 5.1.1, 5.1.3, 5.1.12.9, and 5.3.2**. For the new fuel storage facility, additional guidance acceptable for meeting this criterion is found in **ANS 57.3, paragraphs 6.2.1.3(2), 6.2.3.1, 6.3.1.1, 6.3.3.4, and 6.3.4.2**.	Y	9.1.2.1
9.1.2-SAC-02	Acceptance for meeting the relevant aspect of **GDC 4** is based on **positions C.2 and C.3 of RG 1.13**, and **RG 1.115** and **RG 1.117**	Y	9.1.2.1
9.1.2-SAC-03	**GDC 5** is met by sharing the SSCs important to safety between the units in a manner that does not degrade the performance of their safety functions.	Y	9.1.2.1
9.1.2-SAC-04	Acceptance for meeting the relevant aspect of **GDC 61** for the spent fuel storage facility is based on compliance with **positions C.4, C.6, C.10,**	Y	9.1.2.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	C.11 and C.12 of RG 1.13 and the appropriate paragraphs of **ANS 57.2**. Acceptance for meeting this criterion for the new fuel storage facility is based on compliance with the appropriate paragraphs of **ANS 57.3**. Acceptance is also based on meeting the fuel storage capacity requirements noted in **subsection III.1 of this SRP section**. The following design considerations are evaluated: A. Provisions for periodic inspections of components important to safety. B. Suitable shielding for radiation protection, including adequate water levels. C. Appropriate containment and confinement systems. D. Residual heat removal capability by effective coolant flow through the storage racks for spent fuel assemblies. E. Prevention of reduction in fuel storage coolant inventory under accident conditions.		
9.1.2-SAC-05	Acceptance for meeting the relevant aspect of **GDC 63** for spent fuel storage is based on compliance with **position C.7 of RG 1.13** and **paragraph 5.4 of ANS 57.2**. Acceptance for meeting this criterion for the dry storage of new fuel is based on radiation monitoring pursuant to **10 CFR 70.24** or acceptable prevention of an increase in effective multiplication factor (Keff) beyond safe limits as described in **10 CFR 50.68**.	Y (Spent Fuel Storage)	9.1.2.1
		N/A-COL (New Fuel Storage)	9.1.2.1
9.1.2-SAC-06	In meeting the requirements of **10 CFR 20.1101(b)**, **positions C.2.f(2) and C.2.f(6) of RG 8.8** are the bases for acceptance with respect to provisions for decontamination. For spent fuel storage, **paragraph 5.1.5**	Y	9.1.2.1.1 9.1.2.1.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	of ANS 57.2 and appropriate positions of **RG 1.13** are the bases for acceptance. For new fuel storage, **paragraphs 6.3.3.7 and 6.3.4 of ANS 57.3** are the bases for acceptance.		
9.1.2-SAC-07	**10 CFR 50.68** allows the applicant to follow the guidelines of **10 CFR 70.24** for criticality monitors or the guidelines described therein for significant margins of subcriticality.	N/A-COL	9.1.2.1
SRP 9.1.3	**Spent Fuel Pool Cooling and Cleanup System (R2, 03/2007)**		
9.1.3-AC-01	**General Design Criterion (GDC) 2** contained in Appendix A to 10 CFR Part 50, as related to structures housing the system and the system itself being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes. Acceptance for meeting this criterion is based on conformance to **positions C.1, C.2, C.6, and C.8 of RG 1.13** and **position C.1 of RG 1.29** for safety-related **and position C.2 of RG 1.29** for nonsafety-related portions of the system. This criterion does not apply to the cleanup portion of the system and need not apply to the cooling system if the fuel pool makeup water system and its source meet this criterion, the fuel pool building and its ventilation and filtration system meet this criterion, and the ventilation and filtration system meets the guidelines of **RG 1.52**. The cooling and makeup system should be designed to Quality Group C requirements in accordance with **RG 1.26**. However, when the cooling system is not designated Category I it need not meet the requirements of ASME Section XI for inservice inspection of nuclear plant components.	Y	9.1.3
9.1.3-AC-02	**GDC 4** with respect to the capability of the system and the structure housing the system to withstand the effects of external missiles. Acceptance is based on meeting **position C.2 of RG 1.13**.	Y	9.1.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	This criterion does not apply to the cleanup system and need not apply to the cooling water system if the makeup system, its source, the building, and its ventilation and filtration system are tornado protected, and the ventilation and filtration system meets the guidelines of **RG 1.52**.		
9.1.3-AC-03	**GDC 5** as related to shared systems and components important to safety being capable of performing required safety functions.	Y	9.1.3
9.1.3-AC-04	**GDC 61** as related to the system design for fuel storage and handling of radioactive materials, including the following elements: A. The capability for periodic testing of components important to safety. B. Provisions for containment. C. Provisions for decay heat removal that reflect its importance to safety. D. The capability to prevent reduction in fuel storage coolant inventory under accident conditions. E. The capability and capacity to remove corrosion products, radioactive materials and impurities from the pool water and reduce occupational exposures to radiation.	Y	9.1.3
9.1.3-AC-05	**GDC 63** as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal, to detect excessive radiation levels, and to initiate appropriate safety actions.	Y	9.1.3
9.1.3-AC-06	**10 CFR 20.1101(b)** as it relates to radiation doses being kept as low as is reasonably achievable (ALARA). In meeting this regulation, **RG 8.8, positions C.2.f(2) and C.2.f(3)** can be used as a basis for acceptance.	Y	9.1.3
9.1.3-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if	ITAAC	Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.1.3-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
SRP 9.1.4	**Light Load Handling System (Related to Refueling) (R3, 03/2007)**		
9.1.4-AC-01	**GDC 2** as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes.	Y	9.1.4
9.1.4-AC-02	**GDC 5** as it relates to the capability of shared equipment and components to perform safety functions.	Y	9.1.4
9.1.4-AC-03	**GDC 61** as it relates to radioactivity release as a result of fuel damage and the avoidance of excessive personnel radiation exposure.	Y	9.1.4
9.1.4-AC-04	**GDC 62** as it relates to prevention of criticality accidents.	Y	9.1.4
9.1.4-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and	ITAAC	Tier 1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.1.4-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.1.4-SAC-01	Acceptance for meeting the relevant aspects of **GDC 2** is based on **RG 1.29, Positions C.1 and C.2**.	Y	9.1.4
9.1.4-SAC-02	Acceptance for meeting the relevant aspects of **GDC 5** is embodied within the other acceptance criteria	Y	9.1.4
9.1.4-SAC-03	Acceptance for meeting the relevant aspects of **GDC 61** is based in part on the guidelines of **American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1-1992**.	Y	9.1.4
9.1.4-SAC-04	Acceptance for meeting the relevant aspects of **GDC 62** is based in part on **ANSI/ANS 57.1-1992**.	Y	9.1.4
SRP 9.1.5	**Overhead Heavy Load Handling Systems (R1, 03/2007)**		
9.1.5-AC-01	**GDC 1** of Appendix A to 10 CFR Part 50 as to the design, fabrication, and testing of SSCs important to safety to maintain quality standards.	Y	9.1.5.1
9.1.5-AC-02	**GDC 2** as to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes.	Y	9.1.5.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
9.1.5-AC-03	**GDC 4** as to protection of safety-related equipment from the effects of internally- generated missiles (i.e., dropped loads).	Y	9.1.5.3
9.1.5-AC-04	**GDC 5** as to the sharing of equipment and components important to safety.	Y	9.1.5.1.1
9.1.5-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.1.5-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.1.5-SAC-01	Acceptance for meeting the relevant aspects of **GDC 1** is based in part on **NUREG-0554** for overhead handling systems and **ANSI N14.6** or **ASME B30.9** for lifting devices.	Y	9.1.5.1
9.1.5-SAC-02	Acceptance for meeting the relevant aspects of **GDC 2** is based in part on **position C.2 of RG 1.29** and **Section 2.5 of NUREG-0554**.	Y	9.1.5.1
9.1.5-SAC-03	Acceptance for meeting the relevant aspects of **GDC 4** is based in part	Y	9.1.5.2

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	on **position C.5 of RG 1.13**.		
9.1.5-SAC-04	Acceptance for meeting the relevant aspects of **GDC 5** is embodied within the other acceptance criteria.	Y	9.1.5.1.1
SRP 9.2.1	**Station Service Water System (R5, 03/2007)**		
9.2.1-AC-01	**GDC 2** as to capability of the structures housing the service water system and the system itself to withstand the effects of earthquakes.	Y	9.2.1 FSAR Table 3.2-1
9.2.1-AC-02	**GDC 4** as to effects of missiles inside and outside of containment, pipe whip, jets, and environmental conditions from high- and moderate-energy line breaks and dynamic effects of flow instabilities and loads (e.g., water hammer) during normal plant operation as well as during upset or accident conditions.	Y	9.2.1
9.2.1-AC-03	**GDC 5** as to the capability of shared systems and components important to safety to perform required safety functions.	Y	9.2.1
9.2.1-AC-04	**GDC 44** as to heat transfer from SSCs important to safety to an ultimate heat sink. Acceptance is based on the following: A. The capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions. B. Component redundancy for safety function performance assuming a single, active component failure coincident with the loss of offsite power. C. The capability to isolate components, subsystems, or piping if required so that the system safety function will not be compromised.	Y	9.2.1
9.2.1-AC-05	**GDC 45** as to design provisions for inservice inspection of safety-related components and equipment.	Y	9.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.2.1-AC-06	**GDC 46** as to design provisions for operational functional testing of safety-related systems and components.	Y	9.2.1
9.2.1-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.2.1-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.2.1-SAC-01	Protection Against Natural Phenomena. Information that addresses the requirements of **GDC 2** regarding the capability of structures housing the SWS and the SWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of **Regulatory Guide (RG) 1.29, Position C.1** for safety-related portions of the SWS **and Position C.2** for nonsafety-related portions of the SWS are appropriately addressed.	Y	9.2.1 FSAR Table 3.2-1
9.2.1-SAC-02	Environmental and Dynamic Effects. Information that addresses the requirements of **GDC 4** regarding	Y	9.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the SWS, are met: **SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2, and SRP Section 3.6.1**. In addition, the information will be considered acceptable if the design provisions presented in **GL 96-06** and to **GL 96-06, Supplement 1** are appropriately addressed.		
9.2.1-SAC-03	Sharing of Structures, Systems, and Components. Information that addresses the requirements of **GDC 5** regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the SWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s). In addition, the information will be considered acceptable if the provisions **GL 89-13** and **GL 91-13** are appropriately addressed.	Y	9.2.1
9.2.1-SAC-04	Cooling Water System. Information that addresses the requirements of **GDC 44** regarding consideration of the cooling water system will be considered acceptable if a system to transfer heat from SSCs important to safety to an ultimate heat sink is provided. In addition, the SWS can transfer the combined heat load of these SSCs under normal operating and accident conditions, assuming loss of offsite power and a single failure, and that system portions can be isolated so the safety function of the system is not compromised.	Y	9.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.2.1-SAC-05	Cooling Water System Inspection. Information that addresses the requirements of **GDC 45** regarding the inspection of cooling water systems will be considered acceptable if the design of the SWS permits inservice inspection of safety-related components and equipment and operational functional testing of the system and its components.	Y	9.2.1
9.2.1-SAC-06	Cooling Water System Testing. Information that addresses the requirements of **GDC 46** regarding the testing of cooling water systems will be considered acceptable if the SWS is designed for testing to detect degradation in performance or in the system pressure	Y	9.2.1
SRP 9.2.2	**Reactor Auxiliary Cooling Water Systems (R4, 03/2007)**		
9.2.2-AC-01	**GDC 2** as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.	Y	9.2.2
9.2.2-AC-02	**GDC 4** as to effects of missiles inside and outside of containment, effects of pipe whip, jets, environmental conditions from high- and moderate-energy line breaks, and dynamic effects of flow instabilities and attendant loads (i.e., water hammer) during normal plant operation as well as upset or accident conditions.	Y	9.2.2
9.2.2-AC-03	**GDC 5** as to capability of shared systems and components important to safety to perform required safety functions.	Y	9.2.2
9.2.2-AC-04	**GDC 44** as to: A. The capability to transfer heat loads from safety-related SSCs to a	Y	9.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

<table>
<tr><th colspan="4">CHAPTER 9
Auxiliary Systems</th></tr>
<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>heat sink under both normal operating and accident conditions.
B. Component redundancy for performance of safety functions assuming a single, active component failure coincident with the loss of offsite power.
C. The capability to isolate components, systems, or piping, if required, so system safety functions are not compromised.
D. Remote manual isolation of the RCP seal coolant water by the main control room operator for continued long-term pump operation in an actual event.
E. Whether a single CWS failure results in fuel damage or reactor coolant leakage in excess of normal coolant-makeup capability. Single failure includes, but is not limited to, operator error, spurious activation of a valve operator, and loss of a cooling water pump.
F. Whether a moderate-energy leakage crack or an accident from a CWS piping failure results in excessive fuel damage or reactor coolant leakage in excess of normal coolant makeup capability. A single, active failure is considered in evaluations of the consequences of this accident. Moderate leakage cracks are determined in accordance with the guidelines of Branch Technical Position 3-3, “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.”
G. Demonstration by testing that RCPs withstand a complete loss of cooling water for 20 minutes and instrumentation in accordance with Institute of Electrical and Electronics Engineers Standard (IEEE Std) 603, as endorsed by RG 1.153 with control room alarms detecting loss of cooling water so a period of 20 minutes is available</td><td></td><td></td></tr>
</table>

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	for the operator to have sufficient time to initiate manual protection of the plant. Alternatively, if it is not demonstrated by the necessary pump testing that the RCPs will operate for 20 minutes without operator corrective action, then the following requirements apply: i. Instrumentation in accordance with **IEEE Std 603**, as endorsed by **RG 1.153** consistent with the criteria for the protection system to initiate automatic protection of the plant upon loss of cooling water to a pump. For this case, the component cooling water supply to the seal and bearing of the pump may be designed to nonseismic Category I requirements and Quality Group D; or ii. The component cooling water supply to each pump is designed to withstand a single, active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 **(BTP 3-3)** and to seismic Category I, Quality Group C, and American Society of Mechanical Engineers (ASME) Section III Class 3 requirements.		
9.2.2-AC-05	Not Used	N/A	N/A
9.2.2-AC-06	**GDC 45** as to design provisions for inservice inspection of safety-related components and equipment.	Y	9.2.2
9.2.2-AC-07	**GDC 46** as to design provisions for operational functional testing of safety-related systems or components for: A. Structural integrity and system leak-tightness. B. Operability and adequate performance of active system components. C. Capability of the integrated system to perform required functions during normal, shutdown, and accident situations.	Y	9.2.2
9.2.2-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.2.2-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.2.2-SAC-01	Protection Against Natural Phenomena. Information that addresses the requirements of **GDC 2** regarding the capability of structures housing the reactor auxiliary CWS and the reactor auxiliary CWS itself to withstand the effects of natural phenomena will be considered acceptable if the guidance of **Regulatory Guide (RG) 1.29, Position C.1** for safety-related portions of the reactor auxiliary CWS **and Position C.2** for nonsafety-related portions of the reactor auxiliary CWS are appropriately addressed.	Y	9.2.2
9.2.2-SAC-02	Environmental and Dynamic Effects. Information that addresses the requirements of **GDC 4** regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as	Y	9.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	they apply to the reactor auxiliary CWS, are met: **SRP Sections 3.5.1.1, 3.5.1.4, 3.5.2**, and **SRP Section 3.6.1**. In addition, the information will be considered acceptable if the design provisions presented in **GL 96-06** and **GL 96-06, Supplement 1** are appropriately addressed.		
9.2.2-SAC-03	Sharing of Structures, Systems, and Components. Information that addresses the requirements of **GDC 5** regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the reactor auxiliary CWS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s).	Y	9.2.2
9.2.2-SAC-04	Cooling Water System. Information that addresses the requirements of **GDC 44** regarding consideration of the cooling water system will be considered acceptable if the reactor auxiliary CWS and its components will continue to perform their required safety functions, assuming a single, active failure or a moderate-energy line crack as defined in Branch Technical Position ASB 3-1 **(BTP 3-1)** and to seismic Category I, Quality Group C, and American Society of Mechanical Engineers (ASME) Section III Class 3 requirements concurrent with the loss of offsite power. In addition, the information will be considered acceptable based on appropriate application of **IEEE Std 603**, as endorsed by **RG 1.153**, and appropriate application of **RG 1.155, Position C.3.3.4**.	Y	9.2.2
9.2.2-SAC-05	Cooling Water System Inspection. Information that addresses the requirements of **GDC 45** regarding the inspection of cooling water systems will be considered acceptable if the	Y	9.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	periodic inspection of important reactor auxiliary CWS components ensures system integrity and capability to perform design safety functions.		
9.2.2-SAC-06	Cooling Water System Testing. Information that addresses the requirements of **GDC 46** regarding the testing of cooling water systems will be considered acceptable if periodic system pressure and function testing of the reactor auxiliary CWS will ensure the leak-tight integrity and operability of its components, as well as the operability of the system as a whole, at conditions as close to the design basis as practical.	Y	9.2.2
SRP 9.2.4	**Potable and Sanitary Water Systems (R3, 03/2007)**		
9.2.4-AC-01	**General Design Criterion 60 (GDC 60)**, as it relates to design provisions provided to control the release of liquid effluents containing radioactive material from contaminating the PSWS.	Y	9.2.4
9.2.4-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.2.4-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.2.4-SAC-01	Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section.	Y	9.2.4
SRP 9.2.5	**Ultimate Head Sink (R3, 03/2007)**		
9.2.5-AC-01	**GDC 2** as to capability of structures housing the system and the system itself to withstand the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods.	Y	9.2.5 FSAR Table 3.2-1
9.2.5-AC-02	**GDC 5** as to capability of shared systems and components important to safety to perform required safety functions.	Y	9.2.5
9.2.5-AC-03	**GDC 44** as to: A. The capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. B. Suitable component redundancy so that safety functions can be performed assuming a single, active component failure coincident with loss of offsite power. C. The capability to isolate components, systems, or piping if required so safety functions are not compromised.	Y	9.2.5
9.2.5-AC-04	**GDC 45** as to the design provisions to permit inservice inspection of safety-related components and equipment.	Y	9.2.5
9.2.5-AC-05	**GDC 46** as to the design provisions to permit operation functional testing	Y	9.2.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	of safety-related systems or components.		
9.2.5-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.2.5-AC-07	**10 CFR 52.80(a),** which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.2.5-SAC-01	**GDC 2** requires that nuclear power plant SSCs important to safety be designed to withstand the effects of natural phenomena like earthquake, tornado, hurricane, flood, tsunami, and seiche without loss of capability to perform intended safety functions. **GDC 2** applies to this SRP section because the reviewer considers UHS capability to withstand natural phenomena. The UHS must be able to provide an adequate supply of cooling water to cool the reactor and its essential support systems under all plausible conditions. **RG 1.27** describes methods acceptable to the staff for ensuring UHS capability to withstand the effects of natural phenomena, including earthquakes.	Y	9.2.1 FSAR Table 3.2-1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	GDC 2 requirements provide assurance that SSCs comprised by the plant's ultimate heat sink have been designed to withstand the most severe natural phenomena likely to occur.		
9.2.5-SAC-02	**GDC 5** requires that SSCs important to safety not be shared by nuclear power units unless such sharing can be shown not to impair their capability to perform intended safety functions. **RG 1.27** describes staff positions on UHS design for sharing of SSCs. **GDC 5** applies to any multi-unit facility in which a UHS portion is shared by two or more units. **GDC 5** requirements provide assurance that, in an active or a passive failure at a multi-unit site, the sharing of UHS SSCs will not affect the safe shutdown of any unit.	Y	9.2.5
9.2.5-SAC-03	**GDC 44** requires systems to transfer heat from SSCs important to safety to a UHS. Systems must be able to function under normal and accident conditions, assuming a single failure. **GDC 44** applies to this SRP section because the reviewer evaluates the UHS design, including assumptions for heat loads, redundancy of components, capability to isolate components, and single failures. **RGs 1.27 and 1.72** describe guidance acceptable to the staff for UHS design and fiberglass piping for spray pond applications. In addition, **ANSI/ANS-5.1** describes methods acceptable to the staff for calculating residual decay energy. **GDC 44** requirements provide assurance that the UHS will function as designed to transfer heat from SSCs as required under normal and accident conditions, assuming a single failure.	Y	9.2.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.2.5-SAC-04	**GDC 45** requires that the cooling water system be designed to permit appropriate periodic inspection of important components (e.g., heat exchangers and piping) to ensure the integrity and capability of the system. Meeting the requirements of **GDC 45** provides assurance that components and equipment of the ultimate heat sink can and will be inspected, thereby ensuring that the system will perform its intended safety function.	Y	9.2.5
9.2.5-SAC-05	**GDC 46** requires that the cooling water system be designed to permit appropriate periodic pressure and functional testing to ensure the leaktight integrity and operability of its components, as well as the operability of the system as a whole, under conditions as close to the design basis as practical. Meeting the requirements of **GDC 46** provides assurance that components and equipment of the ultimate heat sink can and will be tested, thereby ensuring that the system will perform its intended safety function.	Y	9.2.5
SRP 9.2.6	**Condensate Storage Facilities (R3, 03/2007)**	N/A-OTHER	9.2.6
SRP 9.3.1	**Compressed Air System (R2, 03/2007)**	N/A-CLASS	9.3.1 FSAR Table 3.2-1
SRP 9.3.2	**Process and Post-Accident Sampling Systems (R3, 03/2007)**		
9.3.2-AC-01	**10 CFR 20.1101(b)**, as it relates to providing engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public as low as is reasonably achievable	Y	9.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(ALARA).		
9.3.2-AC-02	**General Design Criterion (GDC) 1**, found in Appendix A to 10 CFR Part 50, as it relates to the design of the PSS and components in accordance with standards commensurate with the importance of their safety functions.	Y	9.3.2
9.3.2-AC-03	**GDC 2**, as it relates to the ability of the PSS to withstand the effects of natural phenomena.	Y	9.3.2
9.3.2-AC-04	**GDC 13**, as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary.		
9.3.2-AC-05	**GDC 14**, as it relates to assuring the integrity of the reactor coolant pressure boundary by sampling for chemical species that can affect the reactor coolant pressure boundary.	Y	9.3.2
9.3.2-AC-06	**GDC 26**, as it relates to reliably controlling the rate of reactivity changes by sampling boron concentration.	Y	9.3.2
9.3.2-AC-07	**GDC 41**, as it relates to reducing the concentration and quality of fission products released to the environment following postulated accidents by sampling the chemical additive tank for chemical additive concentrations to ensure an adequate supply of chemicals for meeting the material compatibility requirements and the elemental iodine removal requirements of the containment spray and recirculation solutions following a postulated accident.	Y	9.3.2
9.3.2-AC-08	**GDC 60**, as it relates to the capability of the PSS to control the release of radioactive materials to the environment.	Y	9.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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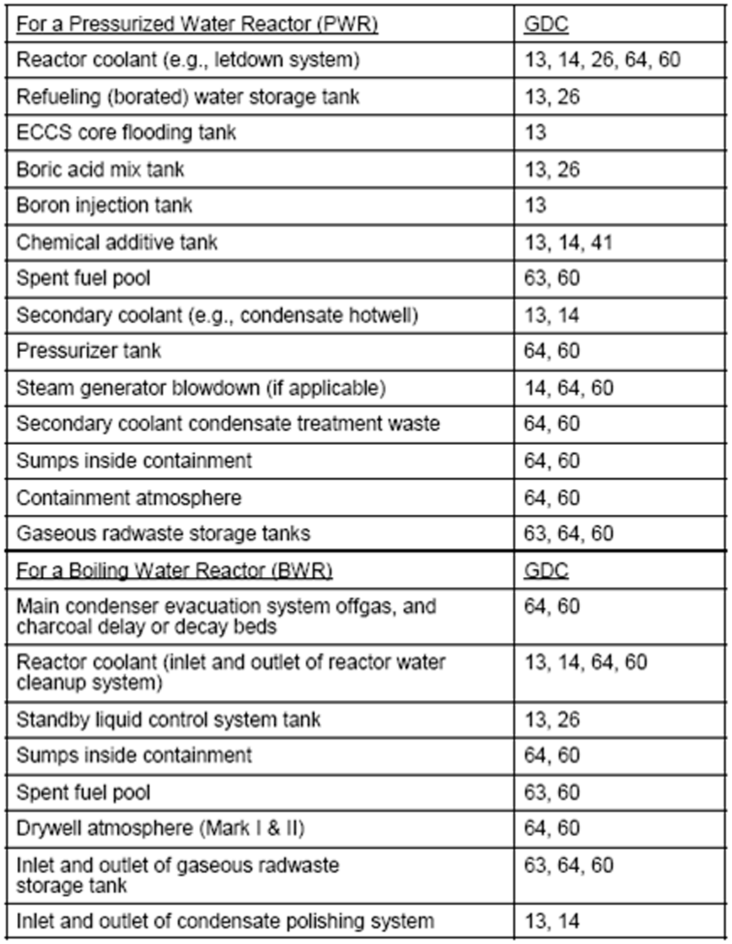
SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
9.3.2-AC-09	**GDC 63**, as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.	Y	9.3.2
9.3.2-AC-10	**GDC 64**, as it relates to monitoring the containment atmosphere and plant environs for radioactivity.	Y	9.3.2
9.3.2-AC-11	Three Mile Island (TMI) Action Plan Item III.D.1.1 in NUREG-0737, as it relates to the provisions for a leakage control program to minimize the leakage from those portions of the PSS outside of the containment that contain or may contain radioactive material following an accident. **10 CFR 50.34(f)(2)(xxvi)** provides equivalent requirements for those applicants subject to 10 CFR 50.34(f).	Y	9.3.2
9.3.2-AC-12	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC’s regulations.	ITAAC	Tier 1
9.3.2-AC-13	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC’s regulations.	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.3.2-SAC-01	The applicant’s design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of **Regulatory Guide (RG) 1.21, Position C.2**, the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines, and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC.	N/A-OTHER (Alternative design concept relative to sampling of containment sumps)	9.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

<table>
<tr><th colspan="4">CHAPTER 9
Auxiliary Systems</th></tr>
<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>
<table>
<tr><th>For a Pressurized Water Reactor (PWR)</th><th>GDC</th></tr>
<tr><td>Reactor coolant (e.g., letdown system)</td><td>13, 14, 26, 64, 60</td></tr>
<tr><td>Refueling (borated) water storage tank</td><td>13, 26</td></tr>
<tr><td>ECCS core flooding tank</td><td>13</td></tr>
<tr><td>Boric acid mix tank</td><td>13, 26</td></tr>
<tr><td>Boron injection tank</td><td>13</td></tr>
<tr><td>Chemical additive tank</td><td>13, 14, 41</td></tr>
<tr><td>Spent fuel pool</td><td>63, 60</td></tr>
<tr><td>Secondary coolant (e.g., condensate hotwell)</td><td>13, 14</td></tr>
<tr><td>Pressurizer tank</td><td>64, 60</td></tr>
<tr><td>Steam generator blowdown (if applicable)</td><td>14, 64, 60</td></tr>
<tr><td>Secondary coolant condensate treatment waste</td><td>64, 60</td></tr>
<tr><td>Sumps inside containment</td><td>64, 60</td></tr>
<tr><td>Containment atmosphere</td><td>64, 60</td></tr>
<tr><td>Gaseous radwaste storage tanks</td><td>63, 64, 60</td></tr>
<tr><td>For a Boiling Water Reactor (BWR)</td><td>GDC</td></tr>
<tr><td>Main condenser evacuation system offgas, and charcoal delay or decay beds</td><td>64, 60</td></tr>
<tr><td>Reactor coolant (inlet and outlet of reactor water cleanup system)</td><td>13, 14, 64, 60</td></tr>
<tr><td>Standby liquid control system tank</td><td>13, 26</td></tr>
<tr><td>Sumps inside containment</td><td>64, 60</td></tr>
<tr><td>Spent fuel pool</td><td>63, 60</td></tr>
<tr><td>Drywell atmosphere (Mark I & II)</td><td>64, 60</td></tr>
<tr><td>Inlet and outlet of gaseous radwaste storage tank</td><td>63, 64, 60</td></tr>
<tr><td>Inlet and outlet of condensate polishing system</td><td>13, 14</td></tr>
</table>

SRP Section 11.5 gives other sample points that may be included in the PSS but do not require remote sampling.</td><td></td><td></td></tr>
</table>

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.3.2-SAC-02	The plant Technical Specifications include the required analysis and frequencies.	Y	9.3.2 Chapter 16
9.3.2-SAC-03	The following guidelines should be used to determine the acceptability of the PSS functional design: A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of **Regulatory Position C.6 in RG 1.21** are followed to meet these criteria. B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with **American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999**. The guidelines of **Regulatory Position C.6 in RG 1.21** are followed to meet this criterion. C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of **Regulatory Position C.7 in RG 1.21** are followed to meet this criterion. D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of **10 CFR 20.1101(b)** to keep radiation exposures at ALARA levels. The guidelines of **Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8** are followed to meet this criterion.	Y	9.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	E. Isolation valves should fail in the closed position, in accordance with the requirements of **GDC 60** to control the release of radioactive materials to the environment. F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of **10 CFR 20.1101(b)** to keep radiation exposures to ALARA levels and the requirements of **GDC 60** to control the release of radioactive materials to the environment. The guidelines of **Regulatory Position 2.i.(6) in RG 8.8** should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.		
9.3.2-SAC-04	To meet the requirements of **GDCs 1 and 2**, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with **Regulatory Positions C.1, C.2, and C.3 in RG 1.26**; **Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29**, and the guidelines of **RG 1.97**. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with **Regulatory Position C.3 in RG 1.26**.	Y	9.3.2

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 9.3.3	**Equipment and Floor Drainage System (R3, 03/2007)**		
9.3.3-AC-01	**GDC 2** as to safety-related system portions capable of withstanding the effects of natural phenomena.	Y	9.3.3
9.3.3-AC-02	**GDC 4** as to capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures).	Y	9.3.3
9.3.3-AC-03	**GDC 60** as to suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences. This criterion applies as the EFDS usually consists of two subsystems, radioactive and nonradioactive. The inadvertent transfer of radioactive wastes to the nonradioactive portion of the system could result in radioactive releases to the environs.	Y	9.3.3
9.3.3-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.3.3-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.3.3-SAC-01	Protection Against Natural Phenomena. Information that addresses the requirements of **GDC 2** regarding the capability of safety-related system portions of the EFDS to withstand the effects of natural phenomena. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section. If no portion is safety-related, the EFDS need not meet GDC 2.	Y	9.3.3
9.3.3-SAC-02	Environmental and Dynamic Effects. Information that addresses the requirements of **GDC 4** regarding the capability to withstand the effects of and to be compatible with the environmental conditions (flooding) of normal operation, maintenance, testing, and postulated accidents (pipe break, tank ruptures) will be considered acceptable if the EFDS is designed to prevent flooding that could affect SSCs important to safety (i.e., necessary for safe shutdown, accident prevention, or accident mitigation) adversely.	Y	9.3.3
9.3.3-SAC-03	Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of **GDC 60** regarding the suitable control of the release of radioactive materials in liquid effluent, including anticipated operational occurrences will be considered acceptable if the EFDS is designed to prevent the inadvertent transfer of contaminated fluids to a noncontaminated drainage system for disposal.	Y	9.3.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 9.3.4	**Chemical and Volume Control System (PWR) (Including Boron Recovery System) (R3, 03/2007)**		
9.3.4-AC-01	**General Design Criterion (GDC) 1**, as it relates to system components being assigned quality group classifications and application of quality standards in accordance with the importance of the safety function to be performed.	Y	9.3.4
9.3.4-AC-02	**GDC 2**, as it relates to structures housing the facility and the system itself being capable of withstanding the effects of earthquakes.	Y	6.8 9.3.4
9.3.4-AC-03	**GDC 5**, as it relates to shared systems and components important to safety being capable of performing required safety functions.	Y	6.8 9.3.4
9.3.4-AC-04	**GDC 14**, as it relates to assuring reactor coolant pressure boundary material integrity by means of the CVCS being capable of maintaining RCS water chemistry necessary to meet PWR RCS water chemistry technical specifications.	Y	9.3.4
9.3.4-AC-05	**GDC 29**, as it relates to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the RCS in the event of anticipated operational occurrences, if the plant design relies on the CVCS to perform the safety function of boration for mitigation of design basis events.	Y	6.8 9.3.4
9.3.4-AC-06	**GDCs 33 and 35**, as they relate to the CVCS capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary (RCPB), to function as part of ECCS assuming a single active failure coincident with the loss of offsite power, and to meet ECCS technical specifications, if the plant design relies on	Y	9.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the CVCS to perform the safety function of safety injection as part of ECCS.		
9.3.4-AC-07	**GDCs 60 and 61**, as they relate to CVCS components having provisions for venting and draining through closed systems.	Y	9.3.4
9.3.4-AC-08	**10 CFR 50.34(f)(2)(xxvi)**, with respect to the provisions for a leakage detection and control program to minimize the leakage from those portions of the CVCS outside of the containment that contain or may contain radioactive material following an accident.	Y	9.3.4
9.3.4-AC-09	**Paragraph (a)(2) in 10 CFR 50.63**, “Loss of All Alternating Current Power,” as it relates to the ability of the CVCS to provide sufficient capacity and capability to ensure that the core is cooled in the event of a station blackout.	Y	9.3.4
9.3.4-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC’s regulations.	ITAAC	Tier 1
9.3.4-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Energy Act, and the NRC's regulations.		
9.3.4-SAC-01	The CVCS safety-related functional performance should be maintained in the event of adverse environmental phenomena such as earthquakes, tornadoes, hurricanes, and floods, or in the event of certain pipe breaks or loss of offsite power. For compliance with **GDC 29, 33 and 35**, the CVCS should provide sufficient pumping capacity to supply borated water to the RCS, maintain RCS water inventory within the allowable pressurizer level range for all normal modes of operation, and function as part of the ECCS, if so designed, to supply reactor coolant makeup in the event of small pipe breaks assuming a single active failure coincident with the loss of offsite power. Also, **Regulatory Guide 1.155** describes a means acceptable to the NRC staff for meeting the requirements of **10 CFR 50.63**, "Loss of all alternating current power." If the CVCS is necessary to support a plant SBO coping capability as required by 10 CFR 50.63, the positions in Regulatory Guide 1.155 regarding CVCS design provide an acceptable method for showing compliance.	Y	9.3.4
9.3.4-SAC-02	**SECY-77-439** describes the concept of single failure criteria and the application of the single failure criterion that involves a systematic search for potential single failure points and their effects on prescribed missions. Application of the single failure assumption in system design and analysis provides redundancy and defense-in-depth to ensure functional performance of the CVCS. Also, the requirements of **GDC 5** prohibiting the sharing among nuclear units the SSCs important to safety would be met by the use of a separate CVCS for each unit.	Y	6.8 9.3.4 15.0.0.3.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9
Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
9.3.4-SAC-03	**10 CFR 50.55(a)** requires that components of the RCPB be designed, fabricated, erected, and tested in accordance with the requirements for Class 1 components of S**ection III of the ASME Boiler and Pressure Vessel Code** or equivalent quality standards. **Regulatory Guide 1.26** describes a quality classification system that may be used to determine quality standards acceptable to the NRC staff for satisfying **GDC 1** for other safety related components containing water, steam, or radioactive materials in light-water-cooled nuclear power plants. **RG 1.29** describes a method acceptable to the NRC staff for identifying and classifying those features of LWRs that should be designed to withstand the effects of the safe shutdown earthquake (SSE). The requirements of **GDC 1** regarding the quality standard are met by acceptable application of quality group classifications and application of quality standards as described in **RG 1.26**. The requirement of **GDC 2** regarding the protection against natural phenomena are met by meeting the guidance of **RG 1.29, Position C.1**, for safety-related portions of the system **and Position C.2** for nonsafety-related portion.	Y	6.8 9.3.4
9.3.4-SAC-04	The CVCS design and arrangement should be that all components and piping that can contain boric acid will either be heat traced or will be located within heated rooms to prevent precipitation of boric acid. As additional specific criteria used to review the CVCS and BRS design, the CVCS should include provisions for monitoring: (a) temperature upstream of the demineralizer to assure that resin temperature limits are not exceeded, and (b) filter demineralizer differential pressure to assure that pressure differential limits are not exceeded. In addition, the CVCS should have provision for automatically diverting or isolating the CVCS	Y	9.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	flow to the demineralizer in the event the demineralizer influent temperature exceeds the resin temperature limit.		
9.3.4-SAC-05	**10 CFR 50.34(f)(2)(xxvi)**, as applicable, specifies the provisions regarding detection of reactor coolant leakage outside containment. These requirements will be met, in part, by providing leakage control and detection systems in the CVCS and implementation of appropriate leakage control program.	Y	6.8 9.3.4
9.3.4-SAC-06	Implementation of Action 1 specified in **Bulletin 80-05** provides an acceptable means for the system to prevent the CVCS holdup tanks, which can contain radioactive release, from the formation of such vacuum conditions that could cause wall inward buckling and failure. The requirements of GDC 60 and 61 can be met, in part, by providing in the CVCS appropriately designed venting and draining closed systems to confine the radioactivity associated with the effluents.	Y	9.3.4
9.3.4-SAC-07	**10 CFR 52.47(a)(1)(vi)** specifies that the application of a design certification should contain proposed ITAAC necessary and sufficient to assure the plant is built and will operate in accordance with the design certification. **10 CFR 52.97(b)(1)** specifies that the COL identifies the ITAAC necessary and sufficient to assure that the facility has been constructed and will be operated in conformity with the license. **SRP 14.3** provides guidance for reviewing the ITAAC. The requirements of **10 CFR 52.47(a)(1)(vi)** and **10 CFR 52.97(b)(1)** will be met, in part, by identifying inspections, tests, analyses, and acceptance criteria of the top-level design features of the CVCS in the design certification application and the combined license, respectively.	ITAAC N/A-COL	Tier 1 N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 9.3.5	**Standby Liquid Control System (BWR) (R3, 03/2007)**	N/A-BWR	N/A
SRP 9.4.1	**Control Room Area Ventilation System (R3, 03/2007)**		
9.4.1-AC-01	**GDC 2**, “Design Bases for Protection Against Natural Phenomena,” as it relates to system capability to withstand the effects of earthquakes.	Y	9.4.1
9.4.1-AC-02	**GDC 4**, “Environmental and Dynamic Effects Design Bases,” as it relates to the CRAVS being appropriately protected against dynamic effects and being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal operation, maintenance, testing, and postulated accidents. The GDC 4 evaluation includes the adequacy of environmental support for safety-related SSCs within areas served by the CRAVS.	Y	9.4.1
9.4.1-AC-03	**GDC 5**, “Sharing of Structures, Systems, and Components,” as it relates to shared SSCs among nuclear power units.	Y	9.4.1
9.4.1-AC-04	**GDC 19**, “Control Room,” as it relates to providing adequate protection to permit access to and occupancy of the control room under accident conditions.	Y	9.4.1
9.4.1-AC-05	**GDC 60**, “Control of Release of Radioactive Materials to the Environment,” as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.	Y	9.4.1
9.4.1-AC-06	**10 CFR 50.63**, as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability for cope with a station blackout event. An analysis to determine capability for withstanding (if an acceptable alternate ac source is provided) or coping with a station blackout event is required. The analysis should address, as appropriate, the potential failures of equipment/systems during the event	Y	9.4.1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(e.g., loss or degraded operability of heating, ventilating, and air conditioning systems, including the CRAVS, as appropriate), the expected environmental conditions associated with the event, the operability and reliability of equipment necessary to cope with the event under the expected environmental conditions, and the habitability of plant areas requiring operator access during the event and associated recovery period.		
9.4.1-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.4.1-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.4.1-SAC-01	Protection Against Natural Phenomena. Information that addresses the requirements of **GDC 2** regarding the capability of structures housing the CRAVS and the CRAVS itself to withstand the effects of natural phenomena will be considered acceptable	Y	9.4.1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	if the guidance of **Regulatory Guide (RG) 1.29, Position C.1** for safety-related portions of the CRAVS **and Position C.2** for nonsafety-related portions of the CRAVS are appropriately addressed.		
9.4.1-SAC-02	Environmental and Dynamic Effects. Information that addresses the requirements of **GDC 4** regarding consideration of environmental and dynamic effects will be considered acceptable if the acceptance criteria in the following SRP sections, as they apply to the CRAVS, are met: **SRP Sections 3.5.1.1, 3.5.2, and 3.6.1**.	Y	9.4.1
9.4.1-SAC-03	Sharing of Structures. Systems, and Components. Information that addresses the requirements of **GDC 5** regarding the capability of shared systems and components important to safety to perform required safety functions will be considered acceptable if the use of the CRAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	Y	9.4.1
9.4.1-SAC-04	Control Room. Information that addresses the requirements of **GDC 19** regarding the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain the plant in a safe condition under accident conditions, including loss-of-coolant accidents will be considered acceptable if adequate protection against radiation and hazardous chemical releases are provided to permit access to and occupancy of the control room under accident conditions. **RG 1.78** provide guidance acceptable to the staff for meeting these control room occupancy	Y	9.4.1

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	protection requirements.		
9.4.1-SAC-05	Control of Releases of Radioactive Material to the Environment. Information that addresses the requirements of **GDC 60** regarding the suitable control of the release of gaseous radioactive effluents to the environment will be considered acceptable if the guidance of **RGs 1.52 and 1.140** as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants are appropriately addressed. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.	Y	9.4.1
9.4.1-SAC-06	Loss of All Alternating Current Power. Information that addresses the requirements of **10 CFR 50.63** regarding the necessary support systems providing sufficient capacity and capability for coping with a station blackout event will be considered acceptable if the guidance of **RG1.155**, including position C.3.2.4 is applied appropriately.	Y	9.4.1
SRP 9.4.2	**Spent Fuel Pool Area Ventilation System (R3, 03/2007)**		
9.4.2-AC-01	**GDC 2**, “Design Bases for Protection Against Natural Phenomena,” as related to the system being capable of withstanding the effects of earthquakes.	Y	9.4.2
9.4.2-AC-02	**GDC 5**, “Sharing of Structures, Systems, and Components,” as related to shared systems and components important to safety.	Y	9.4.2

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.4.2-AC-03	**GDC 60**, “Control of Release of Radioactive Materials to the Environment,” as related to the system’s capability to control suitably the release of radioactive materials in gaseous effluents to the environment.	Y	9.4.2
9.4.2-AC-04	**GDC 61**, “Fuel Storage and Handling and Radioactivity Control,” as related to the system’s capability to provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment from the fuel storage facility under normal and postulated accident conditions.	Y	9.4.2
9.4.2-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC’s regulations.	ITAAC	Tier 1
9.4.2-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC’s regulations.	N/A-COL	N/A
9.4.2-SAC-01	For **GDC 2**, acceptance is based on the guidance of **RG 1.29, Position C.1** for safety-related portions **and Position C.2** for nonsafety-related portions.	Y	9.4.2

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.4.2-SAC-02	For **GDC 5**, acceptance is based on the determination that the use of the SFPAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	Y	9.4.2
9.4.2-SAC-03	For **GDC 60**, acceptance is based on the guidance of **RGs 1.52 and 1.140** as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.	Y	9.4.2
9.4.2-SAC-04	For **GDC 61**, acceptance is based on the guidance of **RG 1.13** as to the design of the ventilation system for the spent fuel storage facility, Position C.4.	Y	9.4.2
SRP 9.4.3	**Auxiliary and Radwaste Area Ventilation System (R3, 03/2007)**		
9.4.3-AC-01	**GDC 2**, “Design Bases for Protection Against Natural Phenomena,” as related to the system being capable of withstanding the effects of earthquakes.	N/A-CLASS	9.4.3
9.4.3-AC-02	**GDC 5,** “Sharing of Structures, Systems, and Components,” as related to shared systems and components important to safety.	N/A-CLASS	9.4.3
9.4.3-AC-03	**GDC 60**, “Control of Release of Radioactive Materials to the Environment,” as related to the capability of the system to suitably control release of gaseous radioactive effluents to the environment.	N/A-CLASS (For	9.4.3

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Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
		safety-related)	
		Y (For non-safety-related)	9.4.3
9.4.3-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.4.3-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.4.3-SAC-01	For **GDC 2**, acceptance is based on the guidance of **RG 1.29, Position C.1** for safety-related portions, **and Position C.2** for nonsafety-related portions.	N/A-CLASS (For safety-related)	9.4.3
		Y (For non-safety-	9.4.3

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Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
		related)	
9.4.3-SAC-02	For **GDC 5**, acceptance is based on the determination that the use of the ARAVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	N/A-CLASS	9.4.3
9.4.3-SAC-03	For **GDC 60**, acceptance is based on the guidance of **RGs 1.52 and 1.140** as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.	N/A-CLASS (For safety-related)	9.4.3
		Y (For non-safety-related)	9.4.3
SRP 9.4.4	**Turbine Area Ventilation System (R3, 03/2007)**	N/A-CLASS	9.4.4 FSAR Table 3.2-1
SRP 9.4.5	**Engineered Safety Feature Ventilation System (R3, 03/2007)**		
9.4.5-AC-01	**GDC 2** as related to the system being capable of withstanding the effects of earthquakes.	Y	9.4.5
9.4.5-AC-02	**GDC 4** with respect to the ESFVS being appropriately protected against dynamic effects and being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The evaluation with respect to GDC 4 also includes evaluation of the adequacy of environmental support provided to structures, systems, and	Y	9.4.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	components important to safety located within areas served by the ESFVS.		
9.4.5-AC-03	**GDC 5** as related to shared systems and components important to safety.	Y	9.4.5
9.4.5-AC-04	**GDC 17** as related to ensuring proper functioning of the essential electric power system.	Y	9.4.5
9.4.5-AC-05	**GDC 60** as related to the system being capable to suitably control release of gaseous radioactive effluents to the environment.	Y	9.4.5
9.4.5-AC-06	**10 CFR 50.63** as related to necessary support systems providing sufficient capacity and capability for coping with a station blackout event. An analysis to determine capability for withstanding (if an acceptable alternate ac source is provided) or coping with a station blackout event is required. The analysis should address, as appropriate, the potential failures of equipment/systems during the event (e.g., loss of or degraded operability of HVAC systems, including the ESFVS, as appropriate), the expected environmental conditions associated with the event, the operability and reliability of equipment necessary to cope with the event under the expected environmental conditions, and the habitability of plant areas requiring operator access during the event and associated recovery period.	Y	9.4.5
9.4.5-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the Atomic Energy Act, and the NRC's regulations;		
9.4.5-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.4.5-SAC-01	For **GDC 2**, acceptance is based on the guidance of **RG 1.29, Position C.1**, for safety-related portions **and Position C.2** for nonsafety-related portions.	Y	9.4.5
9.4.5-SAC-02	For **GDC 4**, acceptance is based on meeting the acceptance criteria in the following SRP sections, as they apply to the ESFVS: S**RP Sections 3.5.1.1, 3.5.1.4, 3.5.2**, and **SRP Section 3.6.1**.	Y	9.4.5
9.4.5-SAC-03	For **GDC 5**, acceptance is based on the determination that the use of the ESFVS in multiple-unit plants during an accident in one unit does not significantly affect the capability to conduct a safe and orderly shutdown and cool-down in the remaining unit(s).	Y	9.4.5
9.4.5-SAC-04	For **GDC 17**, acceptance is based on the guidance of item 2 under Subsection A and item 1 under Subsection C of the **NUREG-CR/0660** section "Recommendations" for protection of essential electrical components from failure due to the accumulation of dust and particulate materials.	Y	9.4.5
9.4.5-SAC-05	For **GDC 60**, acceptance is based on the guidance of **RGs 1.52 and**	Y	9.4.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 **Auxiliary Systems**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	1.140 as related to design, inspection, testing, and maintenance criteria for post-accident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and adsorption units of light-water-cooled nuclear power plants. For RG 1.52 rev 2, the applicable regulatory position is C.2. For RG 1.52 rev 3, the applicable regulatory position is C.3. For RG 1.140 rev 1, the applicable regulatory positions are C.1 and C.2. For RG 1.140 rev 2, the applicable regulatory positions are C.2 and C.3.		
9.4.5-SAC-06	For **10 CFR 50.63**, acceptance is based on the applicable guidance of **RG 1.155**, including Position C.3.2.4.	Y	9.4.5
SRP 9.5.1	**Fire Protection Program (R5, 03/2007)**		
9.5.1-AC-01	**10 CFR 50.48**, "Fire protection," which requires that operating nuclear power plants have a fire protection plan that satisfies **General Design Criterion (GDC) 3** and also provides general requirements regarding the content of the fire protection plan and the applicability of 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."	Y	9.5.1
9.5.1-AC-02	**10 CFR 50.48(f)** establishes the criteria for a fire protection plan for those plants that have submitted the certifications required for license termination under **10 CFR 50.82(a)(1)**.	N/A-CP/OL	N/A
9.5.1-AC-03	**10 CFR Part 50, Appendix A, GDC 3**, "Fire Protection," establishes the criteria for the fire and explosion protection of SSCs important to safety. GDC 3 also establishes the criteria for fire detection and firefighting systems and for the use of noncombustible and heat-resistant materials throughout the unit.	Y	9.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.5.1-AC-04	**10 CFR Part 50, Appendix A, GDC 5**, “Sharing of Structures, Systems, and Components,” as it applies to shared fire protection systems and potential fire impacts on shared SSCs important to safety.	Y (No shared SSCs)	9.5.1
9.5.1-AC-05	**10 CFR Part 50, Appendix A, GDC 19**, “Control Room,” as it applies to providing the capability both inside and outside the control room to operate plant systems necessary to achieve and maintain safe-shutdown conditions.	Y	9.5.1
9.5.1-AC-06	**10 CFR Part 50, Appendix A, GDC 23**, “Protection System Failure Modes,” as it applies to safe-failure states of the protection system when exposed to adverse conditions associated with fire events or inadvertent operation of fire protection systems.	Y	9.5.1
9.5.1-AC-07	10 CFR Part 50, Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” which establishes the FPP requirements for nuclear power plants operating prior to January 1, 1979, subject to the provisions in 10 CFR 50.48(b). Appendix R establishes, along with other fire protection requirements, the requirement to demonstrate that one success path of SSCs necessary to achieve and maintain safe shutdown of the reactor is protected from the effects of fire. The substantive provisions of Appendix R, or portions thereof, may apply to plants licensed to operate after January 1, 1979, to the extent incorporated in or provided for in the fire protection licensing basis for the individual plants.	N/A-CP/OL	N/A
9.5.1-AC-08	**10 CFR Part 52**, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,” which establishes regulatory requirements applicable to new reactors.	Y	9.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.5.1-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
9.5.1-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.1-AC-11	**10 CFR Part 72**, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,” which establishes regulatory requirements applicable to spent nuclear fuel and waste storage.	N/A-COL	N/A
9.5.1-SAC-01	**RG 1.174**, Revision 1, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” as it applies to the use of PRA in support of changes to the fire protection licensing basis for nuclear power plants. Appropriate techniques for performing a Fire PRA are presented in **NUREG/CR-6850** (EPRI TR-1011989), “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities.”	Y	9.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9
Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
9.5.1-SAC-02	**RG 1.188**, Revision 1, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,” as it applies to FPP considerations for license renewal such as equipment aging issues. This RG endorses the guidance in Nuclear Energy Institute (NEI) document, **NEI 95-10**, Revision 6, “Industry Guideline for Implementing the Requirements of **10 CFR Part 54** - The License Renewal Rule.”	N/A-COL	N/A
9.5.1-SAC-03	**RG 1.189**, Revision 1, “Fire Protection for Nuclear Power Plants,” which provides comprehensive staff positions and guidelines on fire protection for nuclear power plants.	Y	9.5.1
9.5.1-SAC-04	**RG 1.191**, “Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown,” which establishes the fire protection objectives and staff positions for implementing fire protection for those nuclear power plants that have submitted the necessary certifications for license termination under 10 CFR Part 50.82(a).	N/A-COL	N/A
9.5.1-SAC-05	**Regulatory Guide 1.206**, “Combined License Applications for Nuclear Power Plants (LWR Edition),” as it applies to the FPP of any new reactor COL application submitted in accordance with 10 CFR Part 52.	N/A-COL	N/A
9.5.1-SAC-06	Enhanced fire protection criteria for new reactor designs as documented in **SECY 90-016**, **SECY 93-087**, and **SECY 94-084**. SECY 90-016 established enhanced fire protection criteria for evolutionary light water reactors. SECY 93-087 recommended that the enhanced criteria be extended to include passive reactor designs. SECY 90 016 and SECY 93-087 were approved by the Commission in staff requirements memoranda (SRM). SECY 94-084, in part, establishes criteria defining safe-shutdown conditions for passive light water reactor designs.	Y	9.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.5.1-SAC-07	For COL reviews, the description of the operational program and proposed implementation milestone(s) for the fire protection program are reviewed in accordance with 10 CFR 50.48. The operational program for fire protection should be fully implemented prior to fuel receipt at the plant site.	N/A-COL	N/A
SRP 9.5.2	**Communications Systems (R3, 03/2007)**		
9.5.2-AC-01	**Appendix E to 10 CFR Part 50**, “Emergency Planning and Preparedness for Production and Utilization,” particularly part IV.E(9), as it relates to the provision of at least one onsite and one offsite communications system, each with a backup power source.	Y	9.5.2
9.5.2-AC-02	**10 CFR 50.34(f)(2)(xxv)**, “Emergency Response Facilities,” (TMI Action Plan Item III A.1.2).	Y	9.5.2
9.5.2-AC-03	**10 CFR 50.47(a)(8)**, “Equipment and Facilities to Support Emergency Response.”	Y	9.5.2
9.5.2-AC-04	**10 CFR 50.55a**, “Codes and Standards.”	Y	9.5.2
9.5.2-AC-05	**General Design Criteria (GDC) 1**, “Quality Standards and Records.”	Y	9.5.2
9.5.2-AC-06	**GDC 2**, “Design Basis for Protection Against Natural Phenomena.”	Y	9.5.2
9.5.2-AC-07	**GDC 3**, “Fire Protection.”	Y	9.5.2
9.5.2-AC-08	**GDC 4**, “Environmental and Missile Design Bases.”	Y	9.5.2
9.5.2-AC-09	**GDC 19**, “Control Room.”	Y	9.5.2
9.5.2-AC-10	**10 CFR 73.45(e)(2)(iii)**, “Performance Capabilities for Fixed Site Physical Protection Systems - Communications Subsystems.”	Y	9.5.2
9.5.2-AC-11	**10 CFR 73.45(g)(4)(i)**, “Provide Communications Networks.”	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
9.5.2-AC-12	**10 CFR 73.46(f)**, "Fixed Site Physical Protection Systems, Subsystems, Components, and Procedures - Communications Subsystems."	Y	9.5.2
9.5.2-AC-13	**10 CFR 73.55(e)**, "Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage - Detection Aids."	Y	9.5.2
9.5.2-AC-14	**10 CFR 73.55(f)**, "Communications Subsystems."	Y	9.5.2
9.5.2-AC-15	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
9.5.2-AC-16	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.2-SAC-01	Information regarding the requirements of **Appendix E to 10 CFR Part 50, Part IV.E(9)**, will be found acceptable if adequate provisions are made and described for emergency facilities and equipment, including: at least one onsite and one offsite communications system; each system	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 **Auxiliary Systems**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	shall have a backup power source.		
9.5.2-SAC-02	For those applicants subject to either 10 CFR 50.34(f) or the TMI Action Plan, information regarding the requirements of **10 CFR 50.34(f)(2)(xxv)** and TMI Action Plan Item III A.1.2 will be found acceptable if provisions are made for an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility.	Y	9.5.2
9.5.2-SAC-03	Information regarding the requirements of **10 CFR 50.47(a)(8)** will be found acceptable if adequate emergency facilities and equipment to support the response are provided and maintained.	Y	9.5.2
9.5.2-SAC-04	Information regarding the requirements of **10 CFR 50.55a** will be found acceptable if SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	Y	9.5.2
9.5.2-SAC-05	Information regarding the requirements of **GDC 1** will be found acceptable if SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these SSCs will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9
Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	the nuclear power unit licensee throughout the life of the unit.		
9.5.2-SAC-06	Information regarding the requirements of **GDC 2** will be found acceptable if SSCs important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.	Y	9.5.2
9.5.2-SAC-07	Information regarding the requirements of **GDC 3** will be found acceptable if SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on SSCs important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.	Y	9.5.2
9.5.2-SAC-08	Information regarding the requirements of **GDC 4** will be found acceptable if SSCs important to safety are designed to accommodate the	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.		
9.5.2-SAC-09	Information regarding the requirements of **GDC 19** will be found acceptable if equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls (I&C) to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.	Y	9.5.2
9.5.2-SAC-10	Information regarding the requirements of **10 CFR 73.45(e)(2)(iii)** will be found acceptable if communications subsystems and procedures are provided for notification of an attempted unauthorized or unconfirmed removal of strategic special nuclear material so that response can be such as to prevent the removal and satisfy the general performance objective and requirements of **§ 73.20(a)**.	Y	9.5.2
9.5.2-SAC-11	Information regarding the requirements of **10 CFR 73.45(g)(4)(i)** will be found acceptable if communications networks are provided to transmit rapid and accurate security information among onsite forces for routine security operation, assessment of a contingency, and response to a contingency.	Y	9.5.2
9.5.2-SAC-12	Information regarding the requirements of **10 CFR 73.46(f)** will be found	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	acceptable if each guard, watchman, or armed response individual on duty shall be capable of maintaining continuous communication with an individual in each continuously manned alarm station required by **10 CFR 73.46(e)(5)**, who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from law enforcement authorities; each alarm station required by 10 CFR 73.46(e)(5) shall have both conventional telephone service and radio or microwave transmitted two-way voice communication, either directly or through an intermediary, for the capability of communication with the law enforcement authorities; and non-portable communications equipment controlled by the licensee and required by 10CFR 73.46(f) shall remain operable from independent power sources in the event of the loss of normal power.		
9.5.2-SAC-13	Information regarding the requirements of **10 CFR 73.55(e)** will be found acceptable if all alarms required by 10 CFR 73.55 annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station not necessarily onsite, so that a single act cannot remove the capability of calling for assistance or otherwise responding to an alarm. The onsite central alarm station must be considered a vital area and its walls, doors, ceiling, floor, and any windows in the walls and in the doors must be bullet-resisting. The onsite central alarm station must be located within a building in such a manner that the interior of the central alarm station is not visible from the perimeter of the protected area. This station must not contain any operational activities that would interfere with the execution of the alarm response function. Onsite secondary power supply systems for alarm annunciator equipment and non-portable communications	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	equipment as required **10 CFR 73.55(f)** of this section must be located within vital areas. All alarm devices including transmission lines to annunciators shall be tamper indicating and self-checking, e.g., an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power. The annunciation of an alarm at the alarm stations shall indicate the type of alarm (e.g., intrusion alarms, emergency exit alarm, etc.) and location. All emergency exits in each protected area and each vital area shall be alarmed.		
9.5.2-SAC-14	Information regarding the requirements of **10 CFR 73.55(f)** will be found acceptable if each guard, watchman or armed response individual on duty is capable of maintaining continuous communication with an individual in each continuously manned alarm station required by **10 CFR 73.55(e)(1)**, who shall be capable of calling for assistance from other guards, watchmen, and armed response personnel and from local law enforcement authorities. The alarm stations required by 10 CFR 73.55(e)(1) shall have conventional telephone service for communication with the law enforcement authorities as described in **10 CFR 73.55(f)(1)**. To provide the capability of continuous communication, radio or microwave transmitted two-way voice communication, either directly or through an intermediary, shall be established, in addition to conventional telephone service, between local law enforcement authorities and the facility and shall terminate in each continuously manned alarm station required by 10 CFR 73.55(e)(1). Non-portable communications equipment controlled by the licensee and required by 10 CFR 73.55 shall remain operable from independent power sources in the event of the loss of normal power.	Y	9.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 9.5.3	**Lighting Systems (R3, 03/2007)**		
9.5.3-AC-01	There are no general design criteria or other requirements that directly apply to the normal and emergency or supplementary plant lighting systems.	N/A-INFO	N/A
9.5.3-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
9.5.3-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.3-SAC-01	Acceptance criteria of the design of the normal and emergency lighting systems, as described in the applicant's safety analysis report (SAR), is based in part on the degree of similarity of the systems design with those for previously reviewed plants with satisfactory operating experience.	Y	9.5.3
9.5.3-SAC-02	The normal lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate station lighting in all areas, from power	Y	9.5.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	sources described in **Section 8.2 of the SRP** that are required for control and maintenance of equipment and plant access routes during normal plant operations.		
9.5.3-SAC-03	The emergency lighting system(s) is acceptable if the integrated design of the system(s) will provide adequate emergency station lighting in all areas, required for fire fighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas.	Y	9.5.3
9.5.3-SAC-04	The lighting systems designs will be acceptable if they conform to the lighting levels recommended in **NUREG-0700**, which is based on the **Illuminating Engineering Society of North America (IESNA) Lighting Handbook** (Reference 2) as related to systems design and illumination levels recommended for industrial facilities.	Y	9.5.3
SRP 9.5.4	**Emergency Diesel Engine Fuel Oil Storage and Transfer System (R3, 03/2007)**		
9.5.4-AC-01	**GDC 2** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of such natural phenomena as earthquakes, tornadoes, hurricanes, and floods, as established in Chapters 2 and 3 of the SAR.	Y	9.5.4
9.5.4-AC-02	**GDC 4** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks.	Y	9.5.4
9.5.4-AC-03	**GDC 5** as it relates to the capability of shared systems and components important to safety between units to perform required safety functions.	Y	9.5.4
9.5.4-AC-04	**GDC 17** as it relates to the capability of the diesel engine fuel oil system	Y	9.5.4

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	to meet independence and redundancy criteria.		
9.5.4-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
9.5.4-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.4-SAC-01	**GDC 2** requirements for which SSCs must be protected from, or be capable of withstanding, the effects of such natural phenomena as earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDEFSS SSCs. The identification of SSC required to withstand earthquakes without the loss of capability to perform safety functions is listed in **RG 1.29**. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.4
9.5.4-SAC-02	**GDC 4** requirements for which SSCs must be protected from, or be capable of withstanding the effects of externally-and internally-generated	Y	9.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDEFSS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.		
9.5.4-SAC-03	**GDC 5** requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator has an independent fuel oil system.	Y	9.5.4
9.5.4-SAC-04	**GDC 17** as to the capability of the fuel oil system to meet independence and redundancy criteria and the guidance and positions of the following:	Y	9.5.4
	A. **RG 1.137** as to the diesel engine fuel oil system design, fuel oil quality, and tests which are specified in regulatory positions C1 and C2. The regulatory position C1 addresses the design criteria for the fuel oil system such as materials, physical arrangement, and applicable codes and regulations. The physical arrangements of the fuel oil system should provide for inservice inspection and testing in accordance with **ASME Boiler and Pressure Vessel Code Section XI** ,"Rules for Inservice Inspections." Criteria for oil quality are addressed in the position C2. The fuel oil stored in the fuel supply tank or used for filling or refilling the supply tank should meet the Federal Fuel Oil, ASTM, or diesel-generator manufacturer requirements. The quality of fuel oil is determined by performing suitable tests and when it does not meet the prescribed standards it is replaced. Also, prior to adding new fuel oil to the supply tank the test for specific gravity, water sediment and viscosity testing should be performed and the fuel oil not meeting the test requirements should not be added to the tank.	Y	9.5.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B. **NUREG/CR-0660**, "Enhancement of Onsite Emergency Diesel Generator Reliability"	Y	9.5.4
	C. Each diesel engine with its own EDEFSS.	Y	9.5.4
	D. **ANSI/ANS-59.51** regarding the onsite fuel oil storage for each diesel generator being sufficient to operate the diesel generator following any design basis event and a continuous loss of off-site power either for seven days, or for the time required to replenish the fuel from sources outside the plant site following any design event without interruption of the operation of the diesel generator, whichever is longer.	Y	9.5.4
SRP 9.5.5	**Emergency Diesel Engine Cooling Water System (R3, 03/2007)**		
9.5.5-AC-01	**GDC 2** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods as established in SAR Chapters 2 and 3.	Y	9.5.5
9.5.5-AC-02	**GDC 4** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks.	Y	9.5.5
9.5.5-AC-03	**GDC 5** as it relates to the capability of systems and components important to safety shared between units to perform required safety functions.	Y	9.5.5
9.5.5-AC-04	**GDC 17** as it relates to EDECWS capability to meet independence and redundancy criteria.	Y	9.5.5
9.5.5-AC-05	**GDC 44** for a cooling system with suitable redundancy to transfer heat to	Y	9.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	an ultimate heat sink under normal operating and accident conditions.		
9.5.5-AC-06	**GDC 45** for design provisions to permit periodic inspection of safety-related system components and equipment.	Y	9.5.5
9.5.5-AC-07	**GDC 46** for design provisions to permit appropriate functional testing of safety-related systems or components for structural integrity, leak-tightness, operability, and performance of active components and system capability function as intended under accident conditions.	Y	9.5.5
9.5.5-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
9.5.5-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.5-SAC-01	**GDC 2** requirements for which SSCs must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDECWS SSCs. The identification of SSC required to withstand	Y	9.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	earthquakes without the loss of capability to perform safety functions is listed in **RG 1.29**. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.		
9.5.5-SAC-02	**GDC 4** requirements for which SSCs must be protected from, or be capable of withstanding, the effects of externally-and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDECWS SSCs. Comprehensive compliance with **GDC 2** is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.5
9.5.5-SAC-03	**GDC 5** requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator has an independent and reliable cooling water system.	Y (No shared SSCs)	9.5.5
9.5.5-SAC-04.	GDC 17 requirements for the capability of the cooling water system to meet independence and redundancy criteria are met when:		
	A. Each diesel generator has a separate and independent EDECWS.	Y	9.5.5
	B. NRC recommendations specified in **NUREG/CR-0660**, "Enhancement of Onsite Emergency Diesel Generator Reliability," are implemented.	Y	9.5.5
9.5.5-SAC-05.	**GDC 44** requirements are met when the EDECWS has:		
	A. The capability to transfer heat from systems and components to a heat sink under transient or accident conditions.	Y	9.5.5
	B. Redundancy of components for performance of safety functions under accident conditions, assuming a single active component failure, or each diesel generator has a separate and independent	Y	9.5.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	EDECWS.		
	C. The capability to isolate system or piping components if required to maintain the system safety function.	Y	9.5.5
9.5.5-SAC-06	**GDC 45** as to design provisions for periodic inspection of safety-related system components and equipment.	Y	9.5.5
9.5.5-SAC-07	**GDC 46** as to design provisions for appropriate functional testing of safety-related systems or components for structural integrity and leak-tightness, operability, performance of active components, and the capability of the system to function as intended under accident conditions.	Y	9.5.5
SRP 9.5.6	**Emergency Diesel Engine Starting System (R3, 03/2007)**		
9.5.6-AC-01	**GDC 2** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods as established in SAR Chapters 2 and 3.	Y	9.5.6
9.5.6-AC-02	**GDC 4** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of externally and internally generated missiles, pipe whip, and jet impingement forces of pipe breaks.	Y	9.5.6
9.5.6-AC-03	**GDC 5** as it relates to the capability of systems and components important to safety shared between units to perform required safety functions.	Y	9.5.6
9.5.6-AC-04	**GDC 17** as it relates to the capability of the diesel engine air starting system to meet independence and redundancy criteria.	Y	9.5.6
9.5.6-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
9.5.6-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.6-SAC-01	**GDC 2** requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDESS SSCs. The identification of SSC required to withstand earthquakes without loss of capability to perform safety functions is listed in **RG 1.29**. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.6
9.5.6-SAC-02	**GDC 4** requirements for SSCs to be protected against the effects of externally-and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDESS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.6

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CHAPTER 9
Auxiliary Systems

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
9.5.6-SAC-03	**GDC 5** requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s) and each diesel generator an independent starting system.	Y (No shared SSCs)	9.5.6
9.5.6-SAC-04	GDC 17 as to the capability of the diesel engine air starting system to meet independence and redundancy criteria. Specific criteria and guidance necessary to meet GDC 17 requirements are as follow:		
	A. **NUREG/CR-0660** "Enhancement of Onsite Emergency Diesel Generator Reliability."	Y	9.5.6
	B. Each diesel engine should have a dedicated air starting system consisting of an air compressor, an air dryer, one or more air receiver(s), piping, injection lines and valves, and devices to crank the engine as recommended by the engine manufacturer.	Y	9.5.6
	C. As a minimum, the air starting system should be capable of cranking a cold diesel engine five times without recharging the receiver(s). The air starting system capacity should be determined as follows: (i) each cranking cycle duration should be approximately three seconds, (ii) consist of two to three engine revolutions, or (iii) air start requirements per engine start provided by the engine manufacturer, whichever air start requirement is larger.	Y	9.5.6
	D. Alarms should alert operating personnel if the air receiver pressure falls below the minimum allowable value.	Y	9.5.6
	E. Provisions for the periodic or automatic blowdown of accumulated moisture and foreign material in the air receiver(s) and other system	Y	9.5.6

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	critical points.		
	F. Starting air should be dried to a dew point of not more than 10°C (50°F) when installed in a normally-controlled 21°C (70°F) environment; otherwise, the starting air dew point should be controlled to at least 5.5°C (10°F) less than the lowest expected ambient temperature.	Y	9.5.6
SRP 9.5.7	**Emergency Diesel Engine Lubrication System (R3, 03/2007)**		
9.5.7-AC-01	**GDC 2** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods, as established in SAR Chapters 2 and 3.	Y	9.5.7
9.5.7-AC-02	**GDC 4** as it relates SSCs that must be protected from, or be capable of withstanding, the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks.	Y	9.5.7
9.5.7-AC-03	**GDC 5** as it relates to the capability of systems and components important to safety shared between units to perform required safety functions.	Y	9.5.7
9.5.7-AC-04	**GDC 17** as it relates to EDELS capability to meet independence and redundancy criteria.	Y	9.5.7
9.5.7-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	ITAAC	Tier 1

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CHAPTER 9
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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the Atomic Energy Act, and the NRC's regulations;		
9.5.7-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
9.5.7-SAC-01	**GDC 2** requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods apply to safety-related EDELS SSCs. The identification of SSCs required to withstand earthquakes without the loss of capabilities to perform safety functions is listed in **RG 1.29**. Comprehensive compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.7
9.5.7-SAC-02	**GDC 4** requirements for SSCs to be protected against the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks apply to safety-related EDELS SSCs. Comprehensive compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.7
9.5.7-SAC-03	**GDC 5** requirements for sharing of SSCs important to safety among nuclear power units are met if each unit has its own diesel generator(s), each with an independent lubrication system.	Y	9.5.7
9.5.7-SAC-04	GDC 17 requirements of independence and redundancy criteria are applicable to the EDELS. Acceptance is based on the following specific		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria:		
	A. **NUREG/CR-0660**, "Enhancement of Onsite Emergency Diesel Generator Reliability."	Y	9.5.7
	B. System operating pressure, temperature differentials, flow rate, and heat removal rate external to the engine in accordance with engine manufacturer recommendations.	Y	9.5.7
	C. Sufficient system protective measures to maintain required oil quality during engine operation.	Y	9.5.7
	D. Protective measures (e.g., relief ports) to prevent unacceptable crankcase explosions and to mitigate consequences of such events.	Y	9.5.7
	E. A keep-warm oil lubricating system to maintain engine lubricating oil passages in a warmed and filled state when the diesel engine is in the standby mode.	Y	9.5.7
	F. System design to circulate lubricating oil to the diesel engine during standby to enhance starting capability in conditions under which the engine-driven oil pump can pressurize the system quickly following engine starts.	Y	9.5.7
	G. Each diesel engine lubricating oil system completely independent of other diesel engines so a single failure will not cause a loss of the required minimum diesel generator capacity as specified in **ANSI/ANS-59.52**.	Y	9.5.7
	H. Onsite lubricating oil storage capacity for each diesel engine sufficient for seven days operation after any design basis event and a continuous loss of off-site power as specified in **ANSI/ANS-59.52**.	Y	9.5.7

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CHAPTER 9 Auxiliary Systems			
SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 9.5.8	**Emergency Diesel Engine Combustion Air Intake and Exhaust System (R3, 03/2007)**		
9.5.8-AC-01	**GDC 2** as it relates to SSCs that must be protected from, or be capable of withstanding, the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods as established in SAR Chapters 2 and 3.	Y	9.5.8
9.5.8-AC-02	**GDC 4** as it relates to SSCs that must be protected from, or be capable of withstanding the effects of, externally- and internally-generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.	Y	9.5.8
9.5.8-AC-03	**GDC 5** as it relates to safety-related systems and components shared between units able to perform required safety functions.	Y	9.5.8
9.5.8-AC-04	**GDC 17** as it relates to the capabilities of the EDECAIES to meet independence and redundancy criteria.	Y	9.5.8
9.5.8-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
9.5.8-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
9.5.8-SAC-01	**GDC 2** requirements for SSCs to withstand or be protected from the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods, apply to safety-related EDECAIES SSCs. The identification of SSCs required to withstand earthquakes without the loss of capabilities to perform safety function is listed in **RG 1.29**. Compliance with GDC 2 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.8
9.5.8-SAC-02	**GDC 4** requirements of SSCs to be protected against the effects of externally- and internally-generated missiles, pipe whip, and jet impingement forces of pipe breaks, apply to safety-related EDECAIES SSCs. Compliance with GDC 4 is reviewed under other SRP sections as specified in subsection I of this SRP section.	Y	9.5.8
9.5.8-SAC-03	**GDC 5** requirements for sharing of SSC important to safety are met when each diesel generator has its own independent and reliable combustion air intake and exhaust system.	Y	9.5.8
9.5.8-SAC-04	GDC 17 as related to the capabilities of the diesel engine combustion and air intake exhaust system to meet independence and redundancy criteria. Acceptance is based on meeting the following specific criteria:		
	A. **NUREG/CR-0660**, "Enhancement of Onsite Emergency Diesel Generator Reliability." i. Engine combustion air should be through piping directly from	Y	9.5.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 9 Auxiliary Systems			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	outside the building with the air intake sufficiently (20 feet) above ground level and filtered to preclude any degradation of continuous engine function. ii. The piping for room ventilation air should be separate from that for engine combustion air. iii. Engine exhaust gas should not circulate back into the diesel generator room, fuel storage room, or any part of the power plant.		
	B. Each emergency diesel engine should have an independent and reliable combustion air intake and exhaust system sized and physically arranged for no degradation of engine function when the diesel generator set must operate continuously at the maximum rated power output.	Y	9.5.8
	C. The combustion air intake system must have a means of reducing airborne particulate material over the entire time period requiring emergency power, assuming the maximum airborne particulate concentration at the combustion air intake.	Y	9.5.8
	D. Suitable design precautions must preclude degradation of the diesel engine power output due to exhaust gases and other diluents that could reduce oxygen content below acceptable levels.	Y	9.5.8

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CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 10.2	**Turbine Generator (R3, 03/2007)**		
10.2-AC-01	**General Design Criterion (GDC 4)** as it relates to the TGS for the protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generation of turbine missiles.	Y	10.2 10.2A
10.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.2-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.2-SAC-01	Specific criteria necessary to meet the requirements of **GDC 4** are as follows: A. A turbine control and overspeed protection system should control turbine action under all normal or abnormal operating conditions and should ensure that a fullload turbine trip will not cause the turbine to overspeed beyond acceptable limits. Under these conditions, the control and protection system should permit an orderly reactor shutdown by use of either the turbine bypass system and main steam relief system or other engineered safety systems. The overspeed protection system should meet the single failure criterion and should be testable when the turbine is in operation. B. The turbine main steam stop and control valves and the reheat steam stop and intercept	Y	10.2 10.2A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	valves should protect the turbine from exceeding set speeds and should protect the reactor system from abnormal surges. The reheat stop and intercept valves should be capable of closure concurrent with the main steam stop valves, or of sequential closure within an appropriate time limit, to ensure that turbine overspeed is controlled within acceptable limits. The valve arrangements and valve closure times should be structured so that a failure of any single valve to close will not result in excessive turbine overspeed in the event of a TGS trip signal. C. The TGS should have the capability to permit periodic testing of components important to safety while the unit is operating at rated load.		
10.2-SAC-02	An inservice inspection program for main steam and reheat valves should be established and should include the following provisions: A. At intervals of approximately 3-1/3 years, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule required by **Section XI of the American Society of Mechanical Engineers (ASME) Code** for reactor components, at least one main steam stop valve, one main steam control valve, one reheat stop valve, and one reheat intercept valve should be dismantled, and visual and surface examinations should be conducted of valve seats, disks, and stems. If this process detects unacceptable flaws or excessive corrosion in a valve, all other valves of that type should be dismantled and inspected. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance. B. Main steam stop and control valves should be exercised at a frequency recommended by the turbine vendor or valve manufacturer.	Y	10.2 10.2A
10.2-SAC-03	The arrangement of connection joints between the low-pressure turbine exhaust and the main condenser should prevent adverse effects on any safety-related equipment in the turbine room in the event of a rupture (it is preferable not to locate safety-related equipment in the turbine room).	Y	10.2 10.2A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 10.2.3	**Turbine Rotor Integrity (R2, 03/2007)**		
10.2.3-AC-01	**General Design Criterion 4 of Appendix A to 10 CFR Part 50**, as it relates to structures, systems, and components important to safety being appropriately protected against the environmental and dynamic effects, including the effects of missiles, that may result from equipment failure.	Y	10.2.3 10.2A.3
10.2.3-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.2.3-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.2.3-SAC-01	Materials Selection. The turbine forged or welded rotor should be made from a material and by a process that tends to minimize flaw occurrence and maximize fracture toughness properties, such as a NiCrMoV alloy processed by vacuum melting or vacuum degassing. The material should be examined and tested to meet the following criteria: A. Chemical analysis should be performed for each forging. Elements that have a deleterious effect on toughness, such as sulfur and phosphorus, should be controlled to low levels.	Y	10.2.3 10.2A.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10
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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	B. The 50% fracture appearance transition temperature (FATT) as obtained from Charpy tests performed in accordance with specification **ASTM A-370** should be no higher than -18°C (O°F) for low-pressure turbine rotors. The nil-ductility transition (NDT) temperature obtained in accordance with specification **ASTM E-208** may be used in lieu of FATT. NDT temperatures should be no higher than -35°C (-30°F). C. The Charpy V-notch (C_v) energy at the minimum operating temperature of each low-pressure rotor in the tangential direction should be at least 8.3 kg-m (60 ft-lbs). A minimum of three C_v specimens should be tested in accordance with specification **ASTM A-370**.		
10.2.3-SAC-02	Fracture Toughness. The low-pressure turbine disk forged or welded rotor fracture toughness properties are acceptable if the following criteria are met. The ratio of the fracture toughness (K_{Ic}) of the rotor material to the maximum tangential stress at speeds from normal to design overspeed should be at least 10 √mm (2 √in), at minimum operating temperature. Bore stress calculations should include components due to centrifugal loads, interference fit, and thermal gradients. Sufficient warmup time should be specified in the turbine operating instructions to ensure that toughness will be adequate to prevent brittle fracture during startup. Fracture toughness properties can be obtained by any of the following methods: A. Testing of the actual material of the turbine rotor to establish the K_{Ic} value at normal operating temperature. B. Testing of the actual material of the turbine rotor with an instrumented Charpy machine and a fatigue precracked specimen to establish the K_{Ic} (dynamic) value at normal operating temperature. If this method is used, K_{Ic} (dynamic) shall be used in lieu of K_{Ic} (static) in meeting the toughness criteria above. C. Estimating of K_{Ic} values at various temperatures from conventional Charpy and tensile data on the rotor material using methods are presented in J. A. Begley and W. A.	Y	10.2.3 10.2A.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Logsdon, Scientific Paper 71-1E7-AMSLRF-P1. This method of obtaining K_{Ic} should be used only on materials which exhibit a well-defined Charpy energy and fracture appearance transition curve and are strain-rate insensitive. The staff should review the test data and the calculated toughness curve submitted by the applicant. D. Estimating "lower bound" values of K_{Ic} at various temperatures using the equivalent energy concept developed by F. J. Witt and T. R. Mager, ORNL-TM-3894. The staff should review the load-displacement data from the compact tension specimens and the calculated toughness data submitted by the applicant.		
10.2.3-SAC-03	<u>Pre-service Inspection</u>. The applicant's pre-service inspection program is acceptable if it meets the following criteria: A. Forged or welded rotors should be rough machined prior to heat treatment. B. Each finished forged or welded rotor should be subjected to 100% volumetric (ultrasonic), surface, and visual examinations using procedures and acceptance criteria equivalent to those specified for Class 1 components in the **ASME Boiler and Pressure Vessel Code, Sections III and V**. Before welding and/or brazing, all surfaces prepared for welding and/or brazing should be surface examined. After welding and/or brazing, all surfaces exposed to steam should be surface examined, giving particular attention to stress risers and welds. Welds should be ultrasonically examined in the radial and radial-tangential sound beam directions. C. Finish machined bores, keyways, and drilled holes should be subjected to magnetic particle or liquid penetrant examination. No flaw indications in keyway or hole regions are allowed. D. Each turbine rotor assembly should be spin tested at 5% above the maximum speed anticipated during a turbine trip following loss of full load.	Y	10.2.3 10.2A.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
10.2.3-SAC-04	Turbine Rotor Design. The turbine assembly should be designed to withstand normal conditions, anticipated transients, and accidents resulting in a turbine trip without loss of structural integrity. The design of the turbine assembly should meet the following criteria: A. The design overspeed of the turbine should be 5% above the highest anticipated speed resulting from a loss of load. The staff should review the basis for the assumed design overspeed. B. The combined stresses of low-pressure turbine rotor at design overspeed due to centrifugal forces, interference fit, and thermal gradients should not exceed 0.75 of the minimum specified yield strength of the material, or 0.75 of the measured yield strength in the weak direction of the materials if appropriate tensile tests have been performed on the actual rotor material. C. The turbine shaft bearings should be able to withstand any combination of the normal operating loads, anticipated transients, and accidents resulting in a turbine trip. D. The natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20% overspeed should be controlled in the design and operation stages so as to cause no distress to the unit during operation. E. The turbine rotor design should facilitate inservice inspection of all high stress regions, including bores and keyways, without the need for removing the disks from the shaft.	Y	10.2.3 10.2A.3
10.2.3-SAC-05	Inservice Inspection. The applicant's inservice inspection program is acceptable if it meets the following criteria: The inservice inspection program for the steam turbine assembly should provide assurance that rotor flaws that might lead to brittle failure of a rotor at speeds up to design speed will be detected. The inservice inspection and maintenance program for the turbine assembly should comply with the manufacturer’s recommendations.	Y	10.2.3 10.2A.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 **Steam and Power Conversion System**			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Inservice inspection and maintenance activities may be performed during plant shutdown coinciding with the inservice inspection schedule as required by **ASME Boiler and Pressure Vessel Code, Section XI**, and should include complete inspection of all significant turbine components, such as couplings, coupling bolts, turbine shafts, low-pressure turbine blades, low-pressure rotors, and high-pressure rotors. This inspection should consist of visual, surface, and volumetric examinations, as required by the code.		
SRP 10.3	**Main Steam Supply System (R4, 03/2007)**		
10.3-AC-01	**GDC 2**, as it relates to safety-related portions of the system being capable of withstanding the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.	Y	10.3 & FSAR Table 3.2-1
10.3-AC-02	**GDC 4**, with respect to safety-related portions of the system being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.	Y	10.3
10.3-AC-03	**GDC 5**, as it relates to the capability of shared systems and components important to safety to perform required safety functions.	Y	10.3
10.3-AC-04	**GDC 34**, as it relates to the system function of transferring residual and sensible heat from the reactor system in indirect-cycle plants.	Y	10.3
10.3-AC-05	**10 CFR 50.63**, as it relates to the ability of a plant to withstand for a specified duration and then recover from an SBO.	Y	10.3
10.3-AC-06	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
10.3-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.3-SAC-01	Acceptance of **GDC 2** is based on meeting the guidance of **Regulatory Guide 1.29, Position C.1** for safety-related portions and **Position C.2** for nonsafety-related portions.	Y	10.3 & FSAR Table 3.2-1
10.3-SAC-02	Acceptance of **GDC 4** is based on the guidance of **Regulatory Guide 1.115, Position C.1**, as it relates to the protection of SSCs important to safety from the effects of turbine missiles. In addition, the system design should adequately consider water (steam) hammer and relief valve discharge loads to assure that system safety functions can be performed and should assure that operating and maintenance procedures include adequate precautions to prevent water (steam) hammer and relief valve discharge loads. The system design should also include protection against water entrainment.	Y	10.3
10.3-SAC-03	Compliance with **GDC 5** requires that structures, systems, and components important to safety shall not be shared by nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their intended safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. Meeting the requirements of GDC 5 provides assurance that the main steam system and its associated components will continue performing their required safety functions even if they are shared by multiple nuclear power units.	Y	10.3
10.3-SAC-04	Acceptance of **GDC 34** is based on the following:		
	A. The positions in **Branch Technical Position 5-4**, as they relate to the design	Y	10.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	requirements for residual heat removal (RHR).		
	B. Issue Number 1 of **NUREG-0138**, as it relates to credit being taken for all valves downstream of the main steam isolation valves (MSIVs) to limit blowdown of a second steam generator if a steamline were to break upstream of the MSIV.	Y	10.3
10.3-SAC-05	Acceptance of **10 CFR 50.63** is based on meeting **Regulatory Guide 1.155** as it relates to the MSSS design.	Y	10.3
10.3-SAC-06	**Regulatory Guide 1.29, Positions C.1.a, C.1.e, C.1.f, C.2 and C.3**, as it relates to the seismic design classification of system components.	Y	10.3
10.3-SAC-07	**Regulatory Guide 1.117, Appendix Position 2 and 4**, as it relates to the protection of SSCs important to safety from the effects of tornado missiles.	Y	10.3
10.3-SAC-08	**SECY 93-087**, as it applies to BWR plants that do not incorporate an MSIVLCS and for which main steamline fission product holdup and retention are credited in the analysis of design-basis accident radiological consequences as follows: A. Seismic Category I is the classification for the main steamlines extending from the outermost containment isolation valve to the seismic interface restraint and connected piping up to the first normally closed valve. B. The nonseismic Category I classification can apply to the main steamlines from the seismic interface restraint up to, but not including, the turbine stop valve (including connected piping to the first normally closed valve) if the following criteria are met: i. A dynamic seismic analysis method analyzed the lines to demonstrate their structural integrity under SSE loading conditions. ii. All pertinent quality assurance requirements of **Appendix B to 10 CFR Part 50** are applied. iii. For lines used as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can	N/A-BWR	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	establish the flowpath, assuming a single active failure. C. Main steamlines and other main steam system components are assigned a quality group classification in accordance with the criteria of **Branch Technical Position 3-1**.		
SRP 10.3.6	**Steam and Feedwater System Materials (R3, 03/2007)**		
10.3.6-AC-01	**10 CFR 50.55a**, "Codes and Standards," which requires that SSCs shall be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.	Y	10.3.6
10.3.6-AC-02	**10 CFR Part 50, Appendix A, GDC 1**, "Quality Standards and Records," which requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. GDC 1 also requires that appropriate records of the design, fabrication, erection, and testing of SSCs important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.	Y	10.3.6
10.3.6-AC-03	10 CFR Part 50, Appendix A, **GDC 35** - "Emergency Core Cooling," which requires that a system be provided to supply abundant emergency core cooling such that damage to reactor core components is minimal following any loss of reactor coolant. GDC 35 also requires that the system will have containment capabilities to assure that the emergency core cooling function can be accomplished, assuming a single failure. For pressure-containing components of a critical nature, their containment capability, i.e., their structural integrity, including freedom from brittle fracture, can only be assured by requiring minimum fracture toughness performance of the materials of which they are fabricated. This is a standard industrial practice which is frequently used in construction codes of significant steel structures.	Y	10.3.6

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
10.3.6-AC-04	**10 CFR Part 50, Appendix B**, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which establishes quality assurance requirements for the design, construction, and operation of those SSCs of nuclear power plants that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.	Y	10.3.6
10.3.6-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.3.6-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.3.6-SAC-01	Materials Selection and Fabrication of Class 2 and 3 Components	Y	10.3.6
	A. The materials specified for use in Class 2 and 3 components should conform to **Appendix I to Section III of the Code** and to **Parts A, B, and C of Section II of the Code**.		
	B. **Regulatory Guide 1.84**, describes acceptable Code Cases that may be used in conjunction with the above specifications. **Appendix IV to Section III of the Code** provides requirements for approval of new materials.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	C. **Regulatory Guide 1.71** provides the following guidelines for assuring the integrity of welds in locations of restricted direct physical and visual accessibility. i. The performance qualification should require testing of the welder under simulated conditions when conditions of accessibility to production welds are less than 30 to 35 cm (12 to 14 inches) in any direction from the joint. ii. Requalification should be required for significantly different restricted accessibility conditions or when any essential welding variables listed in Code Section IX are changed.		
	D. **Regulatory Guide 1.50** provides methods to control preheat temperatures for welding low alloy steel. For carbon steel and low alloy steel materials, **Section III, Appendix D, Article D-1000 of the ASME Code** specifies preheat temperatures.		
	E. **Regulatory Guide 1.37** and **ANSI Standard N45.2.1-1973**, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," describe acceptable procedures for cleaning and handling Class 2 and 3 components of the steam and feedwater systems.		
	F. Acceptance criteria for nondestructive examination of tubular products are provided in the relevant paragraphs of **Subsections NC and ND of Section III of the ASME Code**.		
10.3.6-SAC-02	The fracture toughness properties of the ferritic materials of these components should meet the following requirements of the editions and addenda of **Section III of the Code**, as specified in **10 CFR 50.55a**:	Y	10.3.6
	A. **NC-2300**, "Fracture Toughness Requirements for Material" (Class 2)		
	B. **ND-2300**, "Fracture Toughness Requirements for Material" (Class 3)		
SRP 10.4.1	**Main Condensers (R3, 03/2007)**		
10.4.1-AC-01	Acceptability of the design of the MC system, as described in the applicant's safety analysis	Y	10.4.1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	report (SAR), is based on meeting the requirements of **General Design Criterion 60 (GDC 60)** and on the similarity of the design to that of plants previously reviewed and found acceptable. The design of the MC is acceptable if the integrated design of the system meets the requirements of GDC 60 as related to failures in the design of the system which do not result in excessive releases of radioactivity to the environment.		
10.4.1-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.1-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.1-SAC-01	The requirements of **GDC 60** are met when the MC design includes provisions to prevent excessive releases of radioactivity to the environment which may result from a failure of a structure, system or component in the MC. Acceptance is based on meeting the following:		
	A. SECY 93-087 gives guidance for new BWR plants that do not incorporate an MSIVLCS and for which MC holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequence. It states that seismic analyses are to be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after an SSE.	N/A-BWR	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B. If there is a potential for explosive mixtures to exist, the MC is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive mixtures, as outlined in SRP Section 11.3, subsection II, “Acceptance Criteria,” SRP Acceptance Criteria, Item 6.	Y	10.4.1
SRP 10.4.2	**Main Condenser Evacuation System (R3, 03/2007)**		
10.4.2-AC-01	**General Design Criterion 60** as it relates to the MCES design for the control of releases of radioactive materials to the environment.	Y	10.4.2
10.4.2-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.2-AC-03	10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.2-SAC-01.A	A. If there is a potential for explosive mixtures to exist, the MCES is designed to withstand the effects of an explosion and instrumentation is provided to detect and annunciate the buildup of potentially explosive mixtures, dual instrumentation is provided to detect, annunciate, and effect control measures to prevent the buildup of potentially explosive	Y	10.4.2

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	mixtures, as outlined in **SRP Section 11.3, subsection II**, "Acceptance Criteria," SRP Acceptance Criteria, Item 6. Such a potential does not exist on systems designed to maintain the steam content above 58% by volume in hydrogen-air mixtures or nitrogen content above 92% by volume in hydrogen-oxygen mixtures in all MCES components. The design pressure and normal operational absolute pressure should be provided for MCES components containing potentially explosive mixtures.		
SRP 10.4.3	**Turbine Gland Sealing System (R3, 03/2007)**		
10.4.3-AC-01	**General Design Criterion 60**, "Control of Releases of Radioactive Materials to the Environment," as it relates to the TGSS design for the control of releases of radioactive materials to the environment.	Y	10.4.3
10.4.3-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.3-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.3-SAC	**GDC 60** requires the TGSS to be designed to provide for the collection and condensation of sealing steam and the venting and treatment of noncondensables. Additional acceptance	Y	10.4.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria and review procedures are contained in the SRP sections referenced in the "review interfaces" section of this SRP.		
SRP 10.4.4	**Turbine Bypass System (R3, 03/2007)**		
10.4.4-AC-01	**General Design Criterion 4 (GDC 4)**, "Environmental and Dynamic Effects Design Basis," in that failure of the TBS due to a pipe break or malfunction of the TBS should not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).	Y	10.4.4
10.4.4-AC-02	**General Design Criterion 34 (GDC 34)**, "Residual Heat Removal," as related to the ability to use the system for shutting down the plant during normal operations. The operation of the TBS eliminates the need to rely solely on safety systems, which are required to meet the redundancy and power source requirements of this criterion.	Y	10.4.4
10.4.4-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.4-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.4-SAC-01	Piping Failures. The requirements of **GDC 4** related to the ability of structures, systems and components	Y	10.4.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	important to safety to meet environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions is met by demonstrating that failure of the TBS due to a pipe break or malfunction of the TBS will not adversely affect essential systems or components (i.e., those necessary for safe shutdown or accident prevention or mitigation).		
10.4.4-SAC-02	Residual Heat Removal. The requirements of **GDC 34** related to providing a reliable system that removes residual heat during normal plant shutdown is met by demonstrating the ability to use the turbine bypass system for shutting down the plant during normal operations. The operation of the TBS eliminates the need to rely solely on safety systems, which are required to meet the redundancy and power source requirements of this criterion.	Y	10.4.4
10.4.4-SAC-03	MSIV Alternate Leakage Path (ALP). For BWR plants that do not incorporate an MSIVLCS and for which TBS holdup and plateout of fission products is credited in the analysis of design basis accident radiological consequences, guidance from SECY 93-087 is applicable. Specifically, the turbine bypass lines from the first valve up to the condenser inlet do not need to be classified as seismic category I if the following criteria are met: A. They have been analyzed using a dynamic seismic analysis method to demonstrate their structural integrity under SSE loading conditions. B. All pertinent QA requirements of Appendix B to 10 CFR Part 50 are applied. C. For lines utilized as an MSIV leakage path to the condenser, reliable power sources must be available for control and isolation valves so that a control operator can establish the flow path assuming a single active failure. In addition, the TBS lines and other components utilized as an MSIV leakage path to the condenser are assigned a quality group classification in accordance with the criteria of Branch Technical Position 3-1.	N/A-BWR	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 10.4.5	**Circulating Water System (R3, 03/2007)**		
10.4.5-AC-01	**General Design Criterion 4 (GDC 4)**, "Environmental and Dynamic Effects Design Bases," as it relates to design provisions provided to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS.	Y	10.4.5
10.4.5-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.5-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.5-SAC-01	The requirements of **GDC 4** are met when the circulating water system design includes provisions to accommodate the effects of discharging water that may result from a failure of a component or piping in the CWS. Acceptance is based on meeting the following: A. Means should be provided to prevent or detect and control flooding of safety-related areas so that the intended safety function of a system or component will not be precluded due to leakage from the CWS. B. Malfunction or a failure of a component or piping of the CWS, including an expansion joint, should not have unacceptable adverse effects on the functional performance capabilities of safety-related systems or components.	Y	10.4.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 10.4.6	**Condensate Cleanup System (R3, 03/2007)**		
10.4.6-AC-01	**General Design Criterion (GDC) 14** found in Appendix A to 10 CFR Part 50, as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	Y	10.4.6
10.4.6-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.6-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.6-SAC-01	For direct cycle (boiling-water reactor (BWR)) plants, SRP Section 5.4.8 provides the criteria for acceptable water purity. SRP Section 5.4.8 refers to the guidelines provided in the latest version in the Electric Power Research Institute (EPRI) report series, “BWR Water Chemistry Guidelines,” and the technical specifications for the water chemistry of BWR reactor coolant systems.	N/A-BWR	N/A
10.4.6-SAC-02	For indirect cycle (pressurized-water reactor (PWR)) plants, **SRP Section 5.4.2.1** provides the criteria for acceptable secondary water chemistry. SRP Section 5.4.2.1 refers to the guidelines provided in the latest version in the EPRI report series, “PWR Secondary Water	Y	10.3.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Chemistry Guidelines.”		
SRP 10.4.7	**Condensate and Feedwater System (R4, 03/2007)**		
10.4.7-AC-01	**General Design Criterion 2 (GDC 2)**, "Design Bases for Protection Against Natural Phenomena," as related to the system being capable of withstanding the effects of earthquakes.	Y	10.4.7
10.4.7-AC-02	**General Design Criterion 4 (GDC 4)**, "Environmental and Dynamic Effects Design Bases," as related to the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions.	Y	10.4.7
10.4.7-AC-03	**General Design Criterion 5 (GDC 5)**, "Sharing of Structures, Systems, and Components," as related to the capability of shared systems and components important to safety to perform required safety functions.	Y	10.4.7
10.4.7-AC-04	**General Design Criterion 44 (GDC 44)**, "Cooling Water," as it relates to: A. The capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions. B. Redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure. (This may be coincident with the loss of offsite power for certain events.) C. The capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.	Y	10.4.7
10.4.7-AC-05	**General Design Criterion 45 (GDC 45)**, "Inspection of Cooling Water System," as related to design provisions to permit periodic inservice inspection of system components and equipment.	Y	10.4.7
10.4.7-AC-06	**General Design Criterion 46 (GDC 46)**, "Testing of Cooling Water System," as related to design provisions to permit appropriate functional testing of the system and components to	Y	10.4.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	ensure structural integrity and leak-tightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.		
10.4.7-AC-07	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.7-AC-08	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.7-SAC-01	Seismic Events. The requirements of GDC 2 are met by demonstrating that structures, systems, and components important to safety will be designed to withstand the effects of natural phenomena such as earthquakes. Acceptance is based on meeting the guidance of **Regulatory Guide 1.29, Position C.1** for safety-related portions **and Position C.2** for nonsafety-related portions.	Y	10.4.7
10.4.7-SAC-02	Fluid Instabilitites. The requirements of **GDC 4** as related to protecting structures, systems and components against the dynamic effects associated with possible fluid flow instabilities (e.g., water hammers) during normal plant operation as well as during upset or accident conditions are met by:	Y	10.4.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	A. Meeting the guidance contained in the **Branch Technical Position 10-2**, "Design Guidelines for Avoiding Water Hammers in Steam Generators," for reducing the potential for water hammers in steam generators; and B. Meeting the guidance related to feedwater-control-induced water hammer. Guidance for water hammer prevention and mitigation is found in **NUREG-0927**, Revision 1.		
10.4.7-SAC-03	<u>Sharing of Structures, Systems, and Components</u>. The requirements of **GDC 5** are met by demonstrating the capability of important to safety components in the CFS which are shared by multiple units to perform their required safety functions.	Y	10.4.7
10.4.7-SAC-04	<u>Heat Removal Capability</u>. The requirements of **GDC 44**, as related to the capability to transfer heat from structures, systems and components important to safety to an ultimate heat sink are met by demonstrating that the CFS is capable of providing heat removal under both normal operating and accident conditions. Sufficient redundancy of components is demonstrated so that under accident conditions the safety function can be performed assuming a single active component failure (which may be coincident with the loss of offsite power for certain events.) The system demonstrates capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.	Y	10.4.7
10.4.7-SAC-05	<u>Inspection</u>. The requirements of **GDC 45** are met by demonstrating that the design contains provisions to permit periodic inservice inspection of system components and equipment.	Y	10.4.7
10.4.7-SAC-06	<u>Testing</u>. The requirements of **GDC 46** are met by demonstrating that the design contains provisions to permit appropriate functional testing of the system and components to ensure structural integrity and leak-tightness, operability and performance of active components, and	Y	10.4.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 **Steam and Power Conversion System**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	capability of the integrated system to function as intended during normal, shutdown, and accident conditions.		
10.4.7-SAC-07	Flow Accelerated Corrosion. Piping system designs, including material standards and inspection programs, shall incorporate adequate considerations to avoid erosion and corrosion. Guidance for acceptable inspection programs is found in **Generic Letter 89-08** and in **EPRI NP-3944**, "Erosion/Corrosion in Nuclear Plant Steam Piping: Causes and Inspection Guidelines."	Y	10.4.7
10.4.7-SAC-08	Feedwater Nozzle Design. For BWRs, feedwater nozzle design, inspection, and testing procedures, and CFS operating procedures are adequate to minimize nozzle cracking at low feedwater flow. The review criteria for this issue are stated in NUREG-0619 and in associated Generic Letters 80-95 and 81-11.	N/A-BWR	N/A
SRP 10.4.8	**Steam Generator Blowdown System (PWR) (R3, 03/2007)**		
10.4.8-AC-01	**General Design Criterion 1 (GDC 1)** as it relates to system components being designed, fabricated, erected, and tested for quality standards.	Y	10.4.8
10.4.8-AC-02	**General Design Criterion 2 (GDC 2)** as it relates to system components designed to seismic Category 1 requirements.	Y	10.4.8
10.4.8-AC-03	**General Design Criterion 13 (GDC 13)** as it relates to monitoring system variables that can affect the reactor coolant pressure boundary and maintaining them within prescribed operating ranges.	Y	10.4.8
10.4.8-AC-04	**General Design Criterion 14 (GDC 14)** as it relates to secondary water chemistry control to maintain the integrity of the primary coolant pressure boundary.	Y	10.4.8
10.4.8-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
10.4.8-AC-06	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.8-SAC-01	The requirements of **GDC 1** and **GDC 2** are met when the design of the SGBS includes the following:		
	A. The design is seismic Category I and Quality Group B, from its connection to the steam generator inside primary containment up to and including the first isolation valve outside containment.	Y	10.4.8
	B. The design is in accordance with the provisions of **Regulatory Guide 1.143, Position C.1.1** downstream of the outer containment isolation valves.	Y	10.4.8
10.4.8-SAC-02	The requirements of **GDC 13** are met when the SGBS design includes provisions to monitor system parameters and maintain them within a range that allows the system to perform its impurity removal function and thereby assist in maintaining the integrity of the reactor coolant pressure boundary.	Y	10.4.8
10.4.8-SAC-03	The requirements of **GDC 14** are met when the SGBS design includes provisions to control secondary water chemistry to maintain the integrity of the primary coolant boundary. Acceptance is based on meeting the following:	Y	10.4.8
	A. The SGBS is sized to accommodate the design blowdown flow needed to maintain	Y	10.4.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	secondary coolant chemistry for normal operation, including anticipated operational occurrences.		
	B. Equipment capacities are based on design blowdown flow rates and are such that temperature limits for heat-sensitive processes are not exceeded.	Y	10.4.8
SRP 10.4.9	**Auxiliary Feedwater System (PWR) (R3, 03/2007)**		
10.4.9-AC-01	**GDC 2**, as related to structures housing the system and the system itself being capable of withstanding the effects of earthquakes, tornados and floods.	Y	10.4.9
10.4.9-AC-02	**GDC 4**, with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks.	Y	10.4.9
10.4.9-AC-03	**GDC 5**, as related to the capability of shared systems and components important to safety to perform required safety functions.	Y	10.4.9
10.4.9-AC-04	**GDC 19**, as related to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown.	Y	10.4.9
10.4.9-AC-05	**GDC 34 and 44**, to assure: A. Capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions. B. Redundancy of components for performance of the safety function under accident conditions, assuming a single active component failure (perhaps coincident with the loss of offsite power for certain events).	Y	10.4.9
10.4.9-AC-06	**GDC 45**, as related to design provisions made to permit periodic in service inspection of system components and equipment.	Y	10.4.9
10.4.9-AC-07	**GDC 46**, as related to design provisions made to permit appropriate functional testing of the system and components to assure structural integrity and leak-tightness, operability and	Y	10.4.9-

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.		
10.4.9-AC-08	**10 CFR 50.62**, as related to the design provisions for automatic initiation of the AFWS in an ATWS.	Y	10.4.9
10.4.9-AC-09	**10 CFR 50.63**, as related to the design provisions for withstanding and recovering from a station blackout, including an acceptable degree of independence from the ac power system and the capability for removal of decay heat at an appropriate rate for an appropriate duration.	Y	10.4.9-
10.4.9-AC-10	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
10.4.9-AC-11	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
10.4.9-SAC-01	Acceptance for meeting the relevant aspects of **GDC 2** is based in part on meeting the guidance of **Position C.1 of Regulatory Guide 1.29** if any portion of the system is deemed to be safety related **and** the guidance of **Position C.2** for nonsafety-related portions. Also, acceptance is based in part on (1) meeting the guidance of **Regulatory Guide 1.117** with respect to identifying portions of the system that should be protected from tornadoes and (2) meeting the guidance of **Regulatory Guide 1.102** with respect to identifying portions of the	Y	10.4.9

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 **Steam and Power Conversion System**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	system that should be protected from flooding.		
10.4.9-SAC-02	Acceptance for meeting the relevant aspects of **GDC 4** is based on identification of essential portions of the system as protected from dynamic effects including internal and external missiles. In part, this information should be consistent with the guidance of **Regulatory Guide 1.117** with respect to identifying portions of the system that should be protected from tornado missiles and the guidance of **BTP 3-3** with respect to identifying portions of the system that should be protected from the dynamic effects of pipe breaks.	Y	10.4.9
10.4.9-SAC-03	Acceptance of **GDC 5** is based on provision of information that addresses the capability of shared portions of the AFW system to perform required safety functions during an accident in one unit such that the capability to conduct a safe and orderly shutdown and cool-down in the unaffected unit(s) is not significantly affected.	Y	10.4.9
10.4.9-SAC-04	Acceptance of **GDC 19** is based on meeting **BTP 5-4** with regards to cold shutdown from the control room using only safety grade equipment.	Y	10.4.9
10.4.9-SAC-05	Acceptance of **GDC 34 and 44** is based on the system having sufficient flow capacity so that the system can remove residual heat over the entire range of reactor operation and cool the plant to the decay heat removal system cut-in temperature and the system design conforming to the guidance of **BTP 10-1** as it relates to AFW pump drive and power supply diversity. In addition, the recommendations of **NUREG-0611** and **NUREG-0635** shall also be met. TMI Action Plan item II.E.1.1 of NUREG 0737 and **10 CFR 50.34(f)(1)(ii)** for applicants subject to 10 CFR 50.34(f) require an AFWS reliability analysis. An acceptable AFWS should have an unreliability in the range of 10^{-4} to 10^{-5} per demand exclusive of station blackout scenarios. Compensating factors (e.g., other methods of accomplishing AFWS safety functions of the AFWS or other reliable methods for cooling the reactor core during abnormal conditions) may be considered to justify a larger AFWS unavailability.	Y EXCEPTION (BTP 10-1 for AFW pump diversity)	10.4.9 10.4.9

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 10 Steam and Power Conversion System			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
10.4.9-SAC-06	Acceptance of **GDC 45** is based on provision of information describing how the design of the AFW system permits inservice inspection of safety-related components and equipment.	Y	10.4.9
10.4.9-SAC-07	Acceptance of **GDC 46** is based on provision of information describing how the design of the AFW system, including instrumentation, permits periodic operational functional testing of safety-related components and equipment.	Y	10.4.9
10.4.9-SAC-08	Acceptance of **10 CFR 50.62** is based on design provisions for automatic initiation of the AFW system in an ATWS.	Y	10.4.9
10.4.9-SAC-09	Acceptance of **10 CFR 50.63** is based on conformance with the guidance of **RG 1.155** as related to the AFWS design.	Y	10.4.9-
BTP 10-1	**Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants (R3, 03/2007)**	See SRP 10.4.9, 10.4.9-SAC-05	
BTP 10-2	**Design Guidelines for Avoiding Water Hammers in Steam Generators (R4, 03/2007)**	See SRP 10.4.7. 10.4.7-SAC-02	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 11.1	**Source Terms (R3, 03/2007)**		
11.1-AC-01	**10 CFR Part 20**, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents to unrestricted areas. Part 20 is not applicable to an ESP application.	Y	11.1.2 Tables 11.2-6 & 11.3-6
11.1-AC-02	**10 CFR Part 50, Appendix I**, as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents considered in the context of numerical guides for design objectives and limiting conditions for operation to meet the criterion “as low as is reasonably achievable” (ALARA) for radioactive material in LWR effluents.	Y	11.1.2 Tables 11.2-5 & 11.3-5
11.1-AC-03	**General Design Criterion 60 (GDC)** as it relates to determining the operational source term that is used in calculations associated with potential radioactivity in effluents to unrestricted areas, such that a nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents provided during normal reactor operation, including anticipated operational occurrences. GDC 60 is not applicable to an ESP application.	Y	11.1.2 11.3.1.2.3 11.2.1.2.4 11.5.1.2
11.1-SAC-01	All normal and potential sources of radioactive effluent delineated above in Subsection I will be considered.	Y	11.2 11.3
11.1-SAC-02	For each source of liquid and gaseous waste considered above in Subsection I.1, the volumes and concentrations of radioactive material given for normal operation and anticipated operational occurrences should be consistent with those given in NUREG-0016 or NUREG-0017.	Y	Table 11.2-3
11.1-SAC-03	Decontamination factors for in-plant control measures used to reduce	Y	9.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	gaseous effluent releases to the environment, such as iodine removal systems and high-efficiency particulate air (HEPA) filters for building ventilation exhaust systems and containment internal cleanup systems should be consistent with those given in **Regulatory Guide 1.140**. The building mixing efficiency for containment internal cleanup should be consistent with NUREG-0017.		
11.1-SAC-04	Decontamination factors for in-plant control measures used to reduce liquid effluent releases to the environment, such as filters, demineralizers and evaporators, should be consistent with those given in NUREG-0016 or NUREG-0017.	Y	FSAR Table 11.2-3
		EXCEPTION (NUREG-0017 - Different decontamination factors used for shim bleed, equipment drains, and clean wastes.)	11.1
11.1-SAC-05	Radwaste augments used in the calculation of effluent releases to the environment are consistent with the findings of a cost-benefit analysis, which may be performed using the guidance of **Regulatory Guide 1.110**. The provisions that require a cost-benefit analysis are stated in **Section II.D of Appendix I to 10 CFR Part 50**.	Y	11.2.4 11.3.4
11.1-SAC-06	Effluent concentration limits at the boundary of the unrestricted area do not exceed the values specified in **Table 2 of Appendix B to 10 CFR Part 20**.	Y	Tables 11.2-7 & 11.3-6
11.1-SAC-07	The source terms result in meeting the design objectives for doses in unrestricted areas as set forth in **Appendix I to 10 CFR Part 50**.	Y	Tables 11.2-5 & 11.3-5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
11.1-SAC-08	For evaluating the source terms, the applicant should provide the relevant information in the SAR as required by **10 CFR 50.34**, and **10 CFR 50.34a**. This technical information should include all the basic data listed in Appendix A (BWRs) and **Appendix B** (PWRs) **to Regulatory Guide 1.112** in order to calculate the releases of radioactive material in liquid and gaseous effluents (the source terms). An acceptable method for satisfying the criteria given in items 1 through 5 consists of using the Gaseous and Liquid Effluent (GALE) Computer Code and the source term parameters given in NUREG-0016 or NUREG-0017 for BWRs and PWRs, respectively. Complete listings of the GALE Computer Codes for BWRs and PWRs are given in NUREG-0016 and NUREG-0017, respectively.	Y	9.41 11.1.2.1 11.2.3.2 Table 11.2-3
11.1-SAC-09	If the applicant’s calculational technique or any source term parameter differs from that given in ANSI/ANS 18.1-1999, NUREG-0016, or NUREG-0017, they should be described in detail and the bases for the methods and/or parameters used should be provided.	Y	11.1.2.1 11.1.3 Tables 11.1-6 & 11.1-7
SRP 11.2	**Liquid Waste Management System (R3, 03/2007)**		
11.2-AC-01	**10 CFR 20.1302**, as it relates to radioactivity in liquid effluents released to unrestricted areas and doses to offsite receptors.	Y	Table 11.2-7
11.2-AC-02	**10 CFR 20.1406**, as it relates to the design and operational procedures to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste.	Y	11.2.1 12.3.6
11.2-AC-03	**10 CFR 50.34a**, as it relates to the availability of sufficient design information to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the environment	Y	11.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	have been met.		
11.2-AC-04	**Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 60**, as it relates to the ability of the LWMS design to control releases of radioactive materials to the environment.	Y	11.2.1
11.2-AC-05	**GDC 61**, as it relates to the ability of the LWMS design to ensure adequate safety under normal and postulated accident conditions.	Y	11.2.1
11.2-AC-06	**Appendix I to 10 CFR Part 50, Sections II.A and II.D**, as they relate to the numerical guides for dose design objectives and limiting conditions for operation to meet the “as low as is reasonably achievable” (ALARA) criterion.	Y	11.2.1 Tables 11.2-4 & 11.2-6
11.2-AC-07	**40 CFR Part 190** (the U.S. Environmental Protection Agency’s (EPA) generally applicable environmental radiation standards), as implemented under **10 CFR 20.1301(e)**, as it relates to limits on annual doses from all sources of radioactivity and radiation from the site (with single or multiple units).	Y	11.2.1 Table 11.2-6
11.2-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
11.2-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
11.2-AC-10	For an ESP application, the relevant requirement is limited to **Appendix I to 10 CFR Part 50**, such that the guidelines in Section II.A can be met.	N/A-ESP	N/A
11.2-SAC-01	The LWMS should have the capability to meet the dose design objectives and include provisions to treat liquid radioactive wastes such that the following is true:		
	A. The calculated annual total quantity of all radioactive materials released from each reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 0.03 millisievert (mSv) (3 millirem (mrem)) to the total body or 0.1 mSv (10 mrem) to any organ. **Regulatory Guides 1.109, 1.112, and 1.113** provide acceptable methods for performing this analysis.	Y	11.2.3.4 Table 11.2-6
	B. In addition to 1.A, the LWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return for a favorable cost-benefit ratio, can effect reductions in doses to the population reasonably expected to be within 80 kilometers (km) (50 miles (mi)) of the reactor. **Regulatory Guide 1.110** provides an acceptable method for performing this analysis.	Y	11.2.4 Tables 11.2-10 & 11.2-11

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	C. The concentrations of radioactive materials in liquid effluents released to unrestricted areas should not exceed the concentration limits in **Table 2, Column 2, of Appendix B, to 10 CFR Part 20**	Y	11.2.3.5 Table 11.2-7
11.2-SAC-02	The LWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, interconnections between subsystems, redundant equipment, mobile equipment, and reserve storage capacity will be considered.	Y	11.2.1
11.2-SAC-03	The seismic design of structures housing LWMS components, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of **Regulatory Guide 1.143** for liquids and liquid wastes produced during normal operation and anticipated operational occurrences. For the purpose of this SRP, the dose limit cited in **Section 5 of Regulatory Guide 1.143**, addressing unmitigated releases of radioactive materials, is revised to be consistent with that of **10 CFR Part 20.1301**. The annual dose limit of Part 20.1301 is 100 mrem for members of the public located in unrestricted areas.	Y	11.2.1 11.2.1.2.2
11.2-SAC-04	System designs should contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of **Regulatory Guide 1.143** and industry standards cited in this regulatory	Y	11.2.2.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	guide for liquids and liquid wastes produced during normal operation and anticipated operational occurrences.		
11.2-SAC-05	System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste, in accordance with the guidelines of **Regulatory Guide 1.143**, for liquids and liquid wastes produced during normal operation and anticipated operational occurrences, and the requirements of **10 CFR 20.1406**, or the DC application, update in the SAR, or the COL application, to the extent not addressed in a referenced certified design.	Y	11.2.1 12.3.6
11.2-SAC-06	For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from liquid effluents using radiological exposure models are developed based on **Regulatory Guides 1.109, 1.111, and 1.113**, and appropriate computer codes, such as the LADTAP II computer code (NUREG/CR-4013) for liquid effluents.	N/A-ESP	N/A
11.2-SAC-07	The relevant regulatory guides and Branch Technical Position are as follows:		
	1. **Regulatory Guide 1.110**, as it relates to performing a cost-benefit analysis for reducing cumulative dose to the population by using available technology.	Y	11.2.4 Tables 11.2-10 & 11.2-11
	2. **Regulatory Guide 1.112**, as it relates to the use of acceptable methods for calculating annual average releases of radioactive materials in liquid effluents.	Y	11.1.2.1
	3. **Regulatory Guide 1.109**, as it relates to the use of acceptable	Y	11.2.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	methods for calculating annual doses to the maximally exposed individual in demonstrating compliance with **10 CFR Part 50, Appendix I** dose objectives.		Table 11.2-6
	4. **Regulatory Guide 1.113**, as it relates to the use of acceptable methods for estimating aquatic dispersion and transport of liquid effluents in demonstrating compliance with **10 CFR Part 50, Appendix I** dose objectives.	Y	11.2.3.4 Table 11.2-6
	5. **Regulatory Guide 1.143**, as it relates to the seismic design and quality group classification of components used in the LWMS and structures housing the systems and the provisions used to control leakages of liquids and liquid wastes produced during normal operation and anticipated operational occurrences.	Y	11.2.1 11.2.1.2.2
	6. **Regulatory Guide 1.143**, as it relates to the definition of the boundary of the LWMS beginning at the interface from plant systems to the point of controlled discharge to the environment, as defined in the ODCM, or at the point of recycling to the primary or secondary water system storage tanks for liquids and liquid wastes produced during normal operation and anticipated operational occurrences.	Y	11.2.2 Figures 11.2-1, 11.2-2, & 11.2-3
	7. **Branch Technical Position BTP 11-6** as it relates to the assessment of a potential release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment, and impacts of the release of radioactive materials at the nearest potable water supply, located in an unrestricted area, for direct human consumption or indirectly through animals, crops, and food processing.	Y	11.2.3.7 Table 11.2-8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 11.3	**Gaseous Waste Management System (R3, 03/2007)**		
11.3-AC-01	**10 CFR 20.1302**, as it relates to radioactivity in gaseous effluents released to unrestricted areas.	Y	11.5.1.2 Table 11.3-6
11.3-AC-02	**10 CFR 20.1406**, as it relates to the design and operational procedures (for applications other than renewals, after August 20, 1997) for minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive waste.	Y	11.3 12.3.6
11.3-AC-03	**10 CFR 50.34a**, as it relates to the provision of sufficient design information to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the unrestricted areas are kept as low as reasonably achievable.	Y	11.3 11.3.1.1 11.3.1.2 Table 11.3-6
11.3-AC-04	**General Design Criterion 3 (GDC) 3**, as it relates to the design of gaseous waste handling and treatment systems to minimize the effects of explosive mixtures of hydrogen and oxygen.	Y	11.3.1.2
11.3-AC-05	**GDC 60**, as it relates to the design of the GWMS to control releases of radioactive materials to the environment.	Y	11.3.1.2
11.3-AC-06	**GDC 61**, as it relates to radioactivity control in the GWMS associated with fuel storage and handling areas.	Y	9.4.2 11.3.1.2 11.3.3
11.3-AC-07	**10 CFR Part 50**, Appendix I, Sections II.B, II.C, and II.D, as they relate to the numerical guides for design objectives and limiting conditions for operation to meet the “as low as is reasonably achievable” criterion.	Y	11.3.4 Table 11.3-5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
11.3-AC-08	**40 CFR Part 190** (EPA generally applicable environmental radiation standards), as implemented under **10 CFR 20.1301(e)**, as it relates to limits on total annual doses from all sources of radioactivity and external radiation from the site (with single or multiple units).	Y N/A-COL (Total Site Doses)	11.3 Table 11.3-5
11.3-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1
11.3-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
11.3-AC-11	For an ESP application, the relevant requirement is limited to **Appendix I to 10 CFR Part 50**, such that the guidelines in Sections II.B and II.C of Appendix I can be met.	N/A-ESP	N/A
11.3-SAC-01	The GWMS should have the capability to meet the dose design objectives and should include provisions to treat gaseous radioactive		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	wastes such that the following is true:		
	A. The calculated annual total quantity of all radioactive materials released from each reactor to the atmosphere will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 0.05 mSv (5 mrem) to the total body or 0.15 mSv (15 mrem) to the skin. **Regulatory Guides 1.109, 1.111, and 1.112** provide acceptable methods for performing this analysis.	Y	11.3.3.4 Table 11.3-5
	B. The calculated annual total quantity of radioactive materials released from each reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 0.01 cGy (10 millirads) for gamma radiation or 0.02 cGy (20 millirads) for beta radiation. **Regulatory Guides 1.109, 1.111, and 1.112** provide acceptable methods for performing this analysis.	Y	11.3.3.4 Table 11.3-5
	C. The calculated annual total quantity of radioiodines, carbon-14, tritium, and all radioactive materials in particulate form released from each reactor at the site in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such releases for any individual in an unrestricted area from all pathways of exposure in excess of 0.15 mSv (15 mrem) to any organ. **Regulatory Guides 1.109, 1.111, and 1.112** provide acceptable methods for performing this analysis.	Y	11.3.3.4 Table 11.3-5
	D. In addition to 1.A, 1.B, and 1.C, above, the GWMS should include all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, for	Y	11.3.4 Tables 11.3-8 & 11.3-9

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	a favorable cost-benefit ratio, can effect reductions in dose to the population reasonably expected to be within 80 km (50 mi) of the reactor. **Regulatory Guide 1.110** provides an acceptable method for performing this analysis.		
	E. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area should not exceed the limits specified in **Table 2, Column 1, of Appendix B to 10 CFR Part 20**.	Y	11.3.3.5 Table 11.3-6
	F. The regulatory position contained in **Regulatory Guide 1.140** is met, as it relates to the design testing and maintenance of normal ventilation exhaust system air filtration and adsorption units at nuclear power plants	Y	9.4 11.3
	G. The regulatory position contained in **Regulatory Guide 1.143** is met; as it relates to the seismic design and quality group classification of components used in the structures housing the GRS and the provisions used to control leakages of gaseous wastes produced during normal operation and anticipated operational occurrences	Y	11.3 11.3.1.2.1 11.3.1.2.2
	H. The regulatory position contained in **Regulatory Guide 1.143** is met, as it relates to the definition of the boundary of the GWMS, beginning at the interface from plant systems to the point of controlled discharges to the environment as defined in the ODCM, or at the point of storage in holdup tanks or decay beds for gaseous wastes produced during normal operation and anticipated operational occurrences.	Y	11.3 Figures 11.3-1 & 11.3-2
11.3-SAC-02	The GWMS should be designed to meet the anticipated processing requirements of the plant. Adequate capacity should be provided to process gaseous wastes during periods when major processing	Y	11.3.1.2 11.3.2.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	equipment may be down for maintenance (single failures) and during periods of excessive waste generation. Systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences, are acceptable. To meet these processing demands, the reviewer will consider shared systems, redundant equipment, mobile equipment, and reserve storage capacity.		
11.3-SAC-03	The seismic design and quality group classification of components used in the GWMS and structures housing the system should conform to **Regulatory Guide 1.143**. The design should include precautions to stop continuous leakage paths (i.e., to provide liquid seals downstream of rupture discs) and to prevent permanent loss of the liquid seals in the event of an explosion due to gaseous wastes produced during normal operation and anticipated operational occurrences.	Y	11.3 11.3.1.2.1 11.3.1.2.2
11.3-SAC-04	System designs should describe features that will minimize, to the extent practicable, contamination of the facility and environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste in accordance with **Regulatory Guide 1.143**, for gaseous wastes produced during normal operation and anticipated operational occurrences, and the requirements of **10 CFR 20.1406** or the DC application, update in the SAR, or the COL application to the extent not addressed in a referenced certified design.	Y	11.3 12.3.6
11.3-SAC-05	System designs should use the guidelines in **Regulatory Guide 1.140** for the design testing and maintenance of HEPA filters and charcoal adsorbers installed in normal ventilation exhaust systems. If decontamination factors for radioiodines that differ from those specified in	Y	9.4 11.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Regulatory Guide 1.140 are used for design purposes, they should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The test data should also support the effects of aging and poisoning by airborne contaminants.		
11.3-SAC-06	If the potential for explosive mixtures of hydrogen and oxygen exists, the GRS portion of the GWMS should either be designed to withstand the effects of a hydrogen explosion or be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures. The GRS is normally the only portion of the system that is vulnerable to potential hydrogen explosion.		
	A. For a system designed to withstand the effects of a hydrogen explosion, the design pressure of the system should be approximately 20 times the operating absolute pressure (including the intermediate stage condenser for BWR offgas systems).	N/A-OTHER (Not design to withstand explosion)	N/A
	B. Small allowances should be made to conform to standard design pressures for off-the-shelf components (e.g., if the system operating pressure is nominally 103 kPa (15 psia) but could approach 138 kPa (20 psia) by design, piping could be designed to 2413 kPa (350 psia), since the next higher standard pressure rating is 4137 kPa (600 psia)).	N/A-OTHER (Not design to withstand explosion)	N/A
	C. The process gas stream should be analyzed for potentially explosive mixtures and annunciated both locally and in the control room.	Y	11.3.2.3.15
	D. For systems not designed to withstand a hydrogen explosion, dual gas analyzers (with dual being defined as two independent gas analyzers continuously operating and providing two independent	Y	11.3.2.3.15
		EXCEPTION	11.3.2.3.15

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	measurements verifying that hydrogen and/or oxygen are not present in potentially explosive concentrations) with automatic control functions are required to preclude the formation or buildup of explosive hydrogen/oxygen mixtures	(Configuration of gas analyzers)	
	Gas analyzers should annunciate alarms both locally and in the control room. Analyzer "high alarm" setpoints should be set at approximately 2 percent and "high-high alarm" setpoints should be set at a maximum of 4 percent hydrogen or oxygen. Control features to reduce the potential for explosion should be automatically initiated at the "high-high alarm" setting. The automatic control features should be as follows: i. For systems designed to preclude explosions by maintaining either hydrogen or oxygen below 4 percent, the source of hydrogen or oxygen (as appropriate) should be automatically isolated from the system (valves should fail in closed position). ii. For systems using recombiners, if the downstream hydrogen and/or oxygen concentration exceeds 4 percent (as appropriate), acceptable control features include automatic switching to an alternate recombiner train. iii. Injection of diluents to reduce concentrations below the limits specified herein. Systems designed to operate below 4 percent hydrogen and below 4 percent oxygen may be analyzed for either hydrogen or oxygen; systems designed to operate below 4 percent hydrogen only (no oxygen restrictions) should be analyzed for hydrogen; and systems designed to operate above 4 percent hydrogen should be analyzed for oxygen.	Y	11.3.2.3.15

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	For BWR systems with steam dilution upstream of the recombiners, analysis for hydrogen (oxygen is not an acceptable alternative) should be downstream of the recombiners and upstream of the delay portions of the system (analysis upstream of the recombiners is not required if the system is designed to assure the availability of dilution steam during operation).	N/A-BWR	N/A
	For PWR systems using recombiners, analysis for hydrogen and/or oxygen should be downstream of the recombiners. In addition, unless the system design features preclude explosive gas mixtures of hydrogen and oxygen upstream of the recombiners, analysis for hydrogen and/or oxygen (as appropriate) should be upstream of the recombiners as well. The number of gas analyzers and control features at each location should be in accordance with this SRP section. One gas analyzer upstream and one gas analyzer downstream of the recombiners should not be construed as dual gas analyzers.	Y	11.3.2.3.15
	For systems involving pressurized storage tanks (excluding surge tanks), at least one gas analyzer is required between the compressor and the storage tanks Dual gas analyzers set to sequentially measure concentrations both upstream and downstream of a recombiner are acceptable for a PWR.	N/A-OTHER	N/A
	When two or more potentially explosive process streams are combined before entering a component, each stream or the combination thereof, is required to have dual gas analyzers.	Y	11.3.2.3.15
	If gas analyzers are to be used to sequentially measure several points in a system not designed to withstand a hydrogen explosion, at least one	N/A-OTHER	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	gas analyzer which is continuously on stream is required. The continuous gas analyzer should be located at a point common to streams and measured sequentially (i.e., the analyzer should be sampling the combined stream).		
	Gas analyzers should have daily sensor checks, monthly functional checks, and quarterly calibrations. Gas analyzers installed in systems designed to withstand a hydrogen explosion should be capable of withstanding a hydrogen explosion; gas analyzers installed in the systems not designed to withstand a hydrogen explosion need not be capable of withstanding a hydrogen explosion (similar requirements apply to radiation monitors which are internal to lines containing potentially explosive mixtures). All gas analyzer instrumentation systems shall be non-sparking.	Y	11.3.2.3.15
11.3-SAC-07	**Branch Technical Position (BTP) 11-5**, as it relates to potential releases of radioactive materials (noble gases) as a result of postulated leakage or failure of a waste gas storage tank or off-gas charcoal delay bed.	Y	11.3.3.6
11.3-SAC-08	For an ESP application, the dose estimates to a hypothetical maximally exposed member of the public from gaseous effluents using radiological exposure models are developed based on **Regulatory Guides 1.109 and 1.111**, and appropriate computer codes, such as the GASPAR II computer code (NUREG/CR-4653) for gaseous effluents.	N/A-ESP	N/A
SRP 11.4	**Solid Waste Management System (R3, 03/2007)**		
11.4-AC-01	**10 CFR 20.1302** and **10 CFR 20.1301(e)**, as they relate to radioactive materials released in gaseous and liquid effluents to unrestricted areas.	Y	11.5.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	These criteria apply to releases resulting from SWMS operation during normal plant operations and anticipated operational occurrences.		11.5.3.1 11.5.3.2 Tables 11.2-7 & 11.3-6
11.4-AC-02	**10 CFR 20.1406**, as it relates to the design and operational procedures (for applications other than license renewals, after August 20, 1997) for minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive waste.	Y	11..4 12.3.6
		N/A-COL (Decommissioning)	N/A
11.4-AC-03	**10 CFR 50.34a**, as it relates to the provision of sufficient information to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the unrestricted areas are kept as low as reasonably achievable.	Y	11.4.1.2 11.4.1.2.4 11.4.2.4
11.4-AC-04	**10 CFR Part 50, Appendix I, Sections II.A, II.B, II.C, and II.D**, as they relate to the numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion.	Y	Tables 11.2-6 & 11.3-5
11.4-AC-05	**40 CFR Part 190** (the U.S. Environmental Protection Agency (EPA), generally applicable environmental radiation standards, as implemented under **10 CFR 20.1301(e)**), as it relates to limits on total annual doses from all sources of radioactivity and radiation from the site (with single or multiple units).	Y	11.5.1.2 Tables 11.2-6 & 11.3-5
11.4-AC-06	**Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 60**, as it relates to the design of the SWMS to control the release of radioactive materials in liquid effluents from the SWMS and to handle solid wastes produced during normal plant operation, including anticipated operational occurrences.	Y	11.4.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
11.4-AC-07	**GDC 61**, as it relates to the ability of systems that may contain radioactivity to assure adequate safety under normal and postulated accident conditions.	Y	11.4.1.2
11.4-AC-08	**GDC 63**, as it relates to the ability of the SWMS to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions.	Y	11.4.1.2 11.4.1.4
11.4-AC-09	**10 CFR 61.55** and **10 CFR 61.56**, as they relate to classifying, processing, and disposing of dry solid and wet wastes at approved low-level radioactive waste disposal sites.	N/A-COL	11.4.3
11.4-AC-10	**10 CFR 20.2006** and **Appendix G to 10 CFR Part 20**, as they relate to the requirements for transferring and manifesting radioactive materials shipments to authorized facilities (e.g., disposal sites, waste processors).	N/A-COL	11.4.3
11.4-AC-11	**10 CFR 20.2007**, as it relates to compliance with other applicable Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes, such as mixed wastes characterized by the presence of hazardous chemicals and radioactive materials, that may be disposed under 10 CFR Part 20.	N/A-COL	11.4.3
11.4-AC-12	**10 CFR 20.2108**, as it relates to the maintenance of waste disposal records until the NRC terminates the pertinent license requirements.	N/A-COL	11.4.3
11.4-AC-13	**10 CFR Part 71 and 49 CFR Parts 171–180**, as they relate to the use of approved containers and packaging methods for the shipment of radioactive materials.	N/A-COL	11.4.3
11.4-AC-14	**49 CFR 173.443**, as it relates to methods and procedures used to monitor for the presence of removable contamination on shipping containers, and **49 CFR 173.441**, as it relates to methods and	N/A-COL	11.4.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	procedures used to monitor external radiation levels for shipping containers and vehicles.		
11.4-AC-15	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
11.4-AC-16	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
11.4-SAC-01	The SWMS design parameters are based on expected radionuclide distributions and concentrations consistent with reactor operating experience for similar designs, as evaluated under SRP Section 11.1.	Y	11.4.1.2.4 Table 11.4-1
11.4-SAC-02	Processing equipment is sized to handle the design SWMS inputs, that is, the types of liquid, wet, and solid wastes; radionuclide distributions and concentrations; radionuclide removal efficiencies and decontamination factors; waste volume reduction and increase factors; waste volumes; and waste generation rates.	Y	11.4.1.2.1 Table 11.4-1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
11.4-SAC-03	All liquid and wet wastes will be stabilized in accordance with a PCP before offsite shipment, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which free liquid is detected in accordance with the requirements of **Branch Technical Position (BTP) 11-3**.	Y	11.4.1 11.4.2.4
		N/A-COL	11.4.3
11.4-SAC-04	Other forms of wet wastes will be stabilized or dewatered (subject to the licensed disposal facility's waste acceptance criteria) in accordance with a PCP, or provisions will be made to verify the absence of free liquid in each container and procedures to reprocess containers in which excess water is detected in accordance with the requirements of **BTP 11-3**.	Y	11.4.1
		N/A-COL	11.4.3
11.4-SAC-05	SWMS design objectives, design criteria, treatment methods, expected effluent releases, process and effluent radiation monitoring and control instrumentation, and methods for establishing process and effluent instrumentation control set points, as they relate to the PCP and ODCM under this SRP Section and SRP Section 11.5.	Y	11.4.1 11.5.1 11.5.2
		N/A-COL	11.4.3 11.5.2
11.4-SAC-06	Waste containers, shipping casks, and methods of packaging wastes meet all applicable Federal regulations (e.g., **10 CFR Part 71**, addressing the packaging and transportation of radioactive materials; **10 CFR 20.2006** and **Appendix G to 10 CFR Part 20**, addressing the transfer and manifesting of radioactive waste shipments; and **49 CFR Parts 171–180**, addressing U.S. Department of Transportation (DOT) regulations for the shipment of radioactive materials); and **10 CFR Part 61** or corresponding State regulations addressing applicable waste acceptance criteria of the disposal facility or waste processors.	Y	11.4.2.4
		N/A-COL	11.4.3
11.4-SAC-07	Onsite waste storage facilities provide sufficient storage capacity to allow	Y	11.4.1.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	time for shorter lived radionuclides to decay before shipping in accordance with the requirements of **BTP 11-3**. The SAR should give the bases for determining the duration of the storage.		11.4.1.2.4 11.4.2.3.1
11.4-SAC-08	SWMS components and piping systems, as well as structures housing SWMS components, are designed in accordance with the provisions of **Regulatory Guide 1.143**, as it relates to the seismic design and quality group classification of components, and **BTP 11-3** for wastes produced during normal operation and anticipated operational occurrences.	Y	11.4.1 11.4.1.2.2 11.4.1.2.3
11.4-SAC-09	The SWMS contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of **Regulatory Guide 1.143** and **BTP 11-3**, as they relate to wastes produced during normal operation and anticipated operational occurrences.	Y	11.4.1
11.4-SAC-10	For long-term onsite storage (e.g., for several years, but within the operational life of the plant), the storage facility should be designed to the guidelines of **Appendix 11.4-A** to this SRP section, including updated guidance from **SECY 93-323** and **SECY 94-198**.	N/A-COL	11.4.3
11.4-SAC-11	Liquid, wet, and dry solid wastes will be processed and disposed of in accordance with **10 CFR 61.55** and **10 CFR 61.56** requirements for waste classification and characteristics and with the waste acceptance criteria of the chosen licensed radioactive waste disposal site. The PCP should present the process and methods used to meet these **10 CFR Part 61** requirements.	N/A-COL	11.4.3
11.4-SAC-12	Mixed wastes (characterized by the presence of hazardous chemicals and radioactive materials) will be processed and disposed in accordance with **10 CFR 20.2007**, as it relates to compliance with other applicable	N/A-COL	11.4.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Federal, State, and local regulations governing any other toxic or hazardous properties of radioactive wastes.		
11.4-SAC-13	All effluent releases (gaseous and liquid) associated with the operation (normal and anticipated operational occurrences) of the SWMS will comply with **10 CFR Part 20** and **Regulatory Guide 1.143**, as they relate to the definition of the boundary of the SWMS beginning at the interface from plant systems, including multiunit stations, to the points of controlled liquid and gaseous effluent discharges to the environment or designated onsite storage locations, as defined in the PCP and ODCM.	N/A-COL	11.5.6
11.4-SAC-14	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the PCP aspect of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with **10 CFR 20.1301** and **10 CFR 20.13.2**, **10 CFR 50.34a**, **10 CFR 50.36a**, and **10 CFR 50, Appendix I, section II and IV**. Its implementation is required by a license condition.	N/A-COL	11.4.3
SRP 11.5	**Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems (R4, 03/2007)**		
11.5-AC-01	**10 CFR 20.1302** and **10 CFR 20.1301(e)**, as they relate to the monitoring of radioactivity in plant radiological effluents to unrestricted areas. These criteria apply to all effluent releases resulting from operation during normal plant operations and anticipated operational occurrences.	Y	11.5.1.2 11.5.3.1 11.5.3.2
11.5-AC-02	**10 CFR 50.34a**, as it relates to equipment design and procedures used to control releases of radioactive material to the environment within the numerical guidance provided in **Appendix I to 10 CFR Part 50**.	Y	11.5.1 11.5.3.1 11.5.3.2
11.5-AC-03	**10 CFR 50.36a**, as it relates to operating procedures and equipment	N/A-COL	11.5.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	installed in the radioactive waste system pursuant to **10 CFR 50.34a** to ensure that releases of radioactive materials to unrestricted areas are kept ALARA.		11.5.2 11.5.3.1 11.5.3.2
11.5-AC-04	**Appendix I to 10 CFR Part 50**, as it relates to numerical guides for design objectives to meet the requirements of **10 CFR 50.34a** and **10 CFR 50.36a**, which specify that radioactive effluents released to unrestricted areas will be kept ALARA.	Y	11.5.1.2 Tables 11.2-6 & 11.3-5
11.5-AC-05	**10 CFR 20.1406**, as it relates to the design and operational procedures in minimizing contamination, facilitating eventual decommissioning, and minimizing the generation of radioactive waste.	Y	11.5.1 12.3.6
11.5-AC-06	**General Design Criterion (GDC) 60 of Appendix A to 10 CFR Part 50**, as it relates to controlling effluent releases from the LWMS, GWMS, and SWMS and designing these systems to handle radioactive materials produced during normal plant operation, including operational occurrences.	Y	11.5.1.2
11.5-AC-07	**GDC 63** and **GDC 64**, as they relate to designing the LWMS, GWMS, and SWMS to monitor radiation levels and radioactivity in effluents, as well as radioactive leakages and spills, during routine operation and anticipated operational occurrences.	Y	11.5.1.2
11.5-AC-08	Requirements specified in **10 CFR 50.34(f)(2)(xvii)** and **10 CFR 50.34(f)(2)(xxvii)** for monitoring gaseous effluents from all potential accident release points, consistent with the requirements of **GDC 63 and 64**.	Y	11.5.3.1 Tables 11.5-1
11.5-AC-09	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC)	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.		
11.5-AC-10	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
11.5-SAC-01	Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems. For **GDC 64** and the requirements specified in **10 CFR 50.34(f)(2)(xvii)** and **10 CFR 50.34(f)(2)(xxvii)**, the system designs should meet the provisions of **Regulatory Guide 1.21 (Position C and Appendix A)**, **Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable)**, **Regulatory Guide 4.15 (Position C)**, and **Appendix A to Regulatory Guide 1.33**. SRP **Branch Technical Position (BTP) 7-10** (see SRP Section 7.5) provides additional guidance on the application of **Regulatory Guide 1.97**.	Y	11.5.1 11.5.2 11.5.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	A. The gaseous and liquid process streams or effluent release points should be monitored and sampled according to **Tables 1 and 2 of this SRP**.	Y	Table 11.5-1
	B. For both boiling water-reactors (BWRs) and pressurized-water reactors (PWRs), liquid waste and gaseous waste (contained in tanks) should be sampled on a batch basis before their release, in accordance with **Regulatory Guide 1.21**. Open structures, such as PWR turbine buildings or atmospheric vents for liquid waste tanks containing treated or processed liquid waste and located outside of buildings, do not require continuous gaseous effluent monitors. For liquid and gaseous effluents that cannot be easily monitored or sampled on a batch basis, **one of the following representative sampling methods** should be provided:	Y	11.5.1 11.5.3.1 11.5.3.2
	i. Use of a continuous proportioning sampling system, with at least two sample collection tanks. The system should be designed to collect a sample at a fixed ratio established between the sample collection flow rate and the effluent stream discharge flow rate.	N/A-OPT (See SRP 11-5, 11.5-SAC-01.B.iv & 11.5-SAC-01.B.v)	N/A
	ii. Use of a periodic automatic grab sampling system, with at least two sample collection tanks. The system should be designed to collect a sample at a fixed volume established at a rate that is proportional to the effluent stream discharge flow rate.	N/A-OPT (See SRP 11-5, 11.5-SAC-01.B.iv & 11.5-SAC-01.B.v)	N/A
	iii. For radioactive materials, other than noble gases in gaseous effluents, a continuous sampling system should be used with replaceable particulate filters and radioiodine adsorbers. The system should be designed to automatically take representative samples at a known flow rate established in accordance with	N/A-OPT (See SRP 11-5, 11.5-SAC-01.B.iv & 11.5-SAC-01.B.v)	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	American National Standards Institute/Health Physics Society (ANSI/HPS) N13.1-1999.		
	iv. For intermittently operating effluent release points, the system should be designed to automatically take samples whenever flow is in the effluent stream using a known ratio between the discharge and sampling stream flow rates.	Y	11.3.1.2.3 11.5.3.1
	v. Periodic sampling and analysis frequencies and types of radiological analyses should be specified for all samples described above in the SREC, ODCM, and/or PCP.	N/A-COL	11.4.3 11.5.2
11.5-SAC-02	Provisions should be made for the installation of instrumentation and monitoring equipment and/or periodic or continuous sampling and analysis of radioactive waste process systems. For **GDC 60 and 63**, as they relate to radioactive waste systems, detection of excessive radiation levels, and initiation of appropriate safety actions, the design of systems should meet the guidelines of **Appendix 11.5-A**, **Regulatory Guide 1.21 (Position C, as applicable)**, **Regulatory Guide 1.97 (Position C and Table 1 or 2, as applicable)**, **Regulatory Guide 4.15 (Position C)**, and **Appendix A to Regulatory Guide 1.33**. SRP **BTP 7-10** (see SRP Section 7.5) provides additional guidance on the application of **Regulatory Guide 1.97**	Y	11.5.1 11.5.1.1 11.5.1.2
	A. Provisions should be made to ensure representative sampling from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks collection or sample test tanks) should be capable of recirculating at a rate of not less than two tank volumes in 8 hours. For gaseous and liquid process stream samples, provisions should be made for purging sampling lines and for reducing the plate-	Y	11.5.1 11.5.1.2 11.5.2 11.5.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	out of radioactive materials in sample lines. Provisions for gaseous sampling from ducts and stacks should be consistent with **ANSI/HPS N13.1-1999**.		
	B. When practicable, provisions should be made to collect samples from process waste streams at central sample stations to reduce leakage, spillage, and radiation exposures to operating personnel in accordance with **SRP Section 9.3.2** and **10 CFR 20.1406**.	Y	9.3.2 12.3.6
	C. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system.	Y	9.3.2 11.5.3.1
11.5-SAC-03	Provisions should be made for administrative and procedural controls for the installation of necessary auxiliary or ancillary equipment, for the inclusion of special features in instrumentation and radiological monitoring sampling systems, and for the analysis of process and effluent streams. For **GDC 63 and 64** (including the requirements specified **SRP 11.5-8 Revision 4** - March 2007 in **10 CFR 50.34(f)(2)(xvii)** and **10 CFR 50.34(f)(2)(xxvii)**), as they relate to radioactive waste process systems and effluent discharge paths, the design of systems and the implementation of administrative and procedural controls should meet the guidelines of **Appendix 11.5-A**, **Regulatory Guide 1.21 (Position C)**, **Regulatory Guide 4.15 (Position C)**, and **Appendix A to Regulatory Guide 1.33**. Instrumentation, sampling, and monitoring provisions should conform to the following:		
	A. Sampling frequencies, required analyses, instrument alarm/trip setpoints, calibration and sensitivities, and provisions for preparing composite samples for low-level radioactivity analyses should conform to **Regulatory Guides 1.21, 1.33, and 4.15**. The plant's SREC,	N/A-COL	11.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	ODCM, and/or PCP should indicate sampling frequencies and required analyses.		
	B. Provisions should be made for the necessary instrumentation and facilities to perform gross beta-gamma and gross alpha measurements, isotopic or radionuclide-specific analyses, and other routine analyses in conformance with **Regulatory Guides 1.21, 1.33, and 4.15**.	N/A-COL	11.5.2
	C. Provisions should be made to perform routine instrument calibration, maintenance, and inspections in conformance with **Regulatory Guides 4.15 and 1.33**. Instrumentation calibration procedures should consider whether instrumentation response is expected to change given that radionuclide distributions may vary with the operating status of the plant (i.e., normal operation, anticipated operational occurrences, and post-accident conditions). The plant's SREC, ODCM, and/or PCP should indicate the frequency of such actions. Provisions should also be made to replace or decontaminate instrumentation or sampling equipment without opening the process system or losing the capability of isolating the effluent stream.	N/A-COL	11.5.2
	D. Isolation valves, dampers, or diversion valves with automatic control features should fail in the closed or safe position. The plant's SREC, ODCM, and/or PCP should establish setpoints for actuation of automatic control features initiating actuation of isolation valves, dampers, or diversion valves. The bases for establishing instrumentation alarm or system activation setpoints should be provided, taking into consideration the following:	N/A-COL	11.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	i. For liquid effluents, in-plant effluent dilution factors and dilution factors beyond the point of discharge to the site boundary and nearest offsite dose receptors ii. For gaseous and particulate effluents from plant stacks and building vents, atmospheric dispersion (χ/Q) and deposition (D/Q) factors to the site boundary and offsite dose receptors		
	E. Non-ESF instrumentation provisions for automatic termination or diversion of releases should conform to the design guidance contained in **Appendix 11.5-A**. **SRP Sections 7.6 and 13.3** address the review the ESF instrumentation provisions for automatic termination or diversion of releases	Y	11.5.1 11.5.1.2
	F. The process used to develop, review, verify, validate, and audit digital computer software used in radiation monitoring and sampling equipment, including software used to terminate or divert process and effluent streams. This aspect addresses software developed by the applicant, purchased through a vendor, or included with the instrumentation.	Y	7.1.1.4.5 7.3 17.5
11.5-SAC-04	Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident. For **GDC 64**, as it relates to potential gaseous effluent release paths, the design of systems should meet the provisions of **NUREG-0718 and NUREG-0737 (item II.F.1 and Attachments 1 and 2)**, **10 CFR 50.34(f)(2)(vxii)** and **10 CFR 50.34(f)(2)(vxxii)**, **Appendix 11.5-A**, and **Regulatory Guide 1.97 (Position C)**. SRP **BTP 7-10** (see SRP Section 7.5) provides additional guidance on the application of **Regulatory Guide 1.97**. In addition, the design of the gaseous waste collection and processing system should	Y	11.5.1 11.5.1.1 11.5.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11
Radioactive Waste Management

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	meet the guidelines referenced in **SRP Sections 9.3.2, 11.3, and 13.3**, as well as the following conditions:		
	A. Administrative controls and procedures in conformance with Acceptance Criterion 3 of this SRP section are to be in effect to minimize inadvertent or accidental releases of radioactive gaseous and particulate effluents.	N/A-COL	11.5.2
	B. Gaseous and particulate radiological effluent monitors are to be provided for the automatic termination of releases in the event that effluent release set-points are exceeded, as provided in Acceptance Criterion 1 of this SRP section and as established in the plant's SREC, ODCM, and/or PCP.	Y	11.5.1.2 11.5.3.1 Table 11.5-1
11.5-SAC-05	Provisions should be made for monitoring instrumentation, sampling, and sample analysis for all identified liquid effluent release paths in the event of a postulated accident. These provisions should be in accordance with **GDC 64** and the requirements of **10 CFR 50.34(f)(2)(vxii)** and **10 CFR 50.34(f)(2)(vxxii)**, as they relate to postulated accidents and identified liquid effluent release paths. In addition, the design of the liquid waste collection and processing system should meet the guidelines referenced in **SRP Sections 9.3.2, 11.2, and 13.3**, as well as the following conditions:	Y	11.5.1 11.5.1.1 11.5.1.2 11.5.3.2
	A. Administrative controls and procedures in conformance with Acceptance Criterion 3 of this SRP section are to be in effect to minimize inadvertent or accidental releases of radioactive liquids.	N/A-COL	11.5.2
	B. Liquid effluent radiological monitors are to be provided for the automatic termination of releases in the event that effluent release set-points are exceeded, as provided in Acceptance Criterion 1 of this	Y	11.5.1.2 11.5.3.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 11 Radioactive Waste Management			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	SRP section and as established in the plant's SREC, ODCM, and/or PCP.		Table 11.5-1
11.5-SAC-06	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone for the RETS/SREC, ODCM and REMP aspects of the Process and Effluent Monitoring and Sampling Program are reviewed in accordance with **10 CFR 20.1301** and **10 CFR 20.13.2**, **10 CFR 50.34a**, **10 CFR 50.36a**, and **10 CFR Part 50, Appendix I, Section II and IV**. Its implementation is required by a license condition.	N/A-COL	11.5.2
BTP 11-3	**Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants (R3, 03/2007)**	See SRP 11.4, 11.4-SAC-03, 11.4-SAC-04, 11.4-SAC-07, 11.4-SAC-08, & 11.4-SAC-09	
BTP 11-5	**Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure (R3, 03/2007)**	See SRP 11.3, 11.3-SAC-07	
BTP 11-6	**Postulated Radioactive Releases Due to Liquid-Containing Tank Failures (03/2007)**	See SRP 11.2, 11.2-SAC-07.7	

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12
Radiation Protection

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 12.1	**Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (R3, 03/2007)**		
12.1-AC-01	**10 CFR 19.12**, as it relates to keeping workers who receive ORE informed as to the storage, transfer, or use of radioactive materials or radiation in such areas, and instructed as to the risk associated with ORE, precautions and procedures to reduce exposures, and the purpose and function of protective devices employed.	N/A-COL	12.5
12.1-AC-02	**10 CFR 20.1101** and the definition of ALARA in 10 CFR 20.1003, as they relate to those measures that ensure that radiation exposures resulting from licensed activities are below specified limits and ALARA.	Y	12.1.1.
		N/A-COL	12.5
12.1-AC-03	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
12.1-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
12.1-SAC-01	<u>Policy Considerations</u>. Acceptability will be based on evidence that a policy for ensuring that ORE will be ALARA has been formulated in accordance with the training requirements in **10 CFR 19.12** and the ALARA provisions of **10 CFR 20.1101(b)**, and that the policy has been described, displayed, and will be implemented in accordance with the provisions of **Regulatory Guides 8.8** (Regulatory Position C.1) **and 8.10** (Regulatory Position C.1) and **NUREG-1736**, as it relates to maintaining doses ALARA. A specific individual(s) will be designated and assigned responsibility and authority for implementing ALARA policy. Alternative proposed policies will be evaluated on the basis of a comparison with the above regulatory guides and NUREG-1736	Y N/A-COL	12.1.1.2 12.1.1.1 12.1.3 12.5
12.1-SAC-02	<u>Design Considerations</u>. Acceptability will be based on evidence that the design methods, approach, and interactions are in accordance with the ALARA provisions of **10 CFR 20.1101(b)** and **Regulatory Guide 8.8** (Regulatory Position C.2) and will include incorporation of measures for reducing the need for time spent in radiation areas; maintenance; measures to improve the accessibility to components requiring periodic maintenance or inservice inspection; measures to reduce the production, distribution, and retention of activated corrosion products throughout the primary system; measures for assuring that ORE during decommissioning will be ALARA; reviews of the design by competent radiation protection personnel; instructions to designers and engineers regarding ALARA design; experience from operating plants and past designs; and continuing facility design reviews. Alternative proposed design policies will be evaluated on the basis of a	Y	12.1.1.2 12.1.2 12.1.2.1.1 12.1.2.1.2 12.1.2.2 12.1.2.3.1 12.1.2.3.2 12.1.3 12.3.6.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	comparison with the design guidance in **Regulatory Guide 8.8** (Regulatory Position C.2).		
12.1-SAC-03	Operational Considerations. Acceptability will be based on evidence that the applicant has a program to develop plans and procedures in accordance with **Regulatory Guides 1.33, 1.8, 8.8, and 8.10** that can incorporate the experiences obtained from facility operation into facility and equipment design and operations planning and that will implement specific exposure control techniques	N/A-COL	12.1.3
12.1-SAC-04	Radiation Protection Considerations. Acceptability will be based on evidence that overall facility operations, as well as the radiation protection program, integrate the procedures necessary to ensure that radiation doses are ALARA, including work scheduling, work planning, design modifications, and radiological considerations.	N/A-COL	12.1.3 12.5
SRP 12.2	**Radiation Sources (R3, 03/2007)**		
12.2-AC-01	**10 CFR 20.1201**, **10 CFR 20.1202**, and **10 CFR 20.1206**, as they relate to limiting occupational radiation doses.	Y	12.1.2
		N/A-COL	12.1.3 12.5
12.2-AC-02	**10 CFR 20.1203** and **10 CFR 20.1204**, as they relate to limiting average concentrations of airborne radioactive materials to protect individuals and control the intake (inhalation or absorption) of such materials.	Y	12.3.1.7.3
		N/A-COL	12.5
12.2-AC-03	**10 CFR 20.1207**, as it relates to limiting exposure to minors to one-tenth of limits for adults.	N/A-COL	12.5
12.2-AC-04	**10 CFR 20.1301**, as it relates to the determination of radiation levels and	Y	11.2.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	radioactive materials concentrations within the components of waste treatment systems.		11.3.3
12.2-AC-05	**10 CFR 20.1801**, as it relates to securing licensed materials against unauthorized removal.	N/A-COL	12.5
12.2-AC-06	**General Design Criterion 61** found in Appendix A to 10 CFR Part 50, as it relates to systems that may contain radioactive materials. *Criterion 61--* *These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.*	Y	5.4.7 9.1.3 12.1.2.1.2 12.3.1 12.3.2.3 12.3.3
12.2-AC-07	**10 CFR 50.34(f)(2)(vii)** and **General Design Criterion 19**, as they relate to the acceptable radiation conditions in the plant under accident conditions, and the source term release assumptions used to estimate calculate those conditions.	Y	12.3.5.2 15.0.3.11
12.2-AC-08	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	ITAAC	Tier 1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
12.2-AC-09	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations	N/A-COL	N/A
12.2-SAC	Descriptions should be provided for all radiation sources that require (1) shielding, (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design process, (2) establishment of related facility design features, (3) development of plans and procedures, and (4) assessment of occupational exposure.	Y	12.2.1 12.2.2 12.3.2.2 12.3.2.3 Tables 12.2-1 through 12.2-16 & 12.2-18 Figures 12.3-13 through 12.2-57
		N/A-COL (ALARA Program)	12.1.3
	For contained sources, the description should include plan scale drawings of each floor of the plant that show all sources identified so that they can easily be related to tables containing the pertinent and necessary quantitative source parameters. Their position should be located accurately, indicating the approximate size and shape.	Y	12.2.1 Figures 12.3-13 through 12.2-57
	Neutron and gamma streaming into containment from the annulus	Y	12.2.1.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12
Radiation Protection

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	between the reactor pressure vessel and the biological shield should be analyzed to determine the radiation fields that could occur in areas that may require occupancy. Relevant experience from operating reactors may be used.		Tables 12.1-2 & 12.2-2
	Airborne sources that are created by leakage, opening formerly closed containers, storage of leaking fuel elements, and other mechanisms should be identified by location and magnitude so that they can be used for designing appropriate ventilation systems and in specifying appropriate monitoring systems.	Y	12.2.2
	Airborne radioactivity concentrations in frequently occupied areas should be a small fraction of the concentrations related to **10 CFR 20.1203**, **10 CFR 20.1204**, and **Appendix B to 10 CFR Part 20**.	Y	11.1.2 12.2.1 Table 12.2-18
	The assumptions made in arriving at quantitative values for these various sources should be specified, either in this section or by reference to **SAR Chapter 11**.	Y	12.2.2 Table 12.2-7
	Shielding and ventilation design fission product source terms will be acceptable if developed using these bases:		
	• An offgas rate of 370 MBq/s (100,000 µCi/s) after a 30-minute delay for BWRs.	N/A-BWR	N/A
	• 0.25-percent fuel cladding defects for PWRs.	Y	11.2.2.2
	• Postaccident shielding (for vital area access, including work in the area) source terms from **NUREG-0737**, Item II.B.2, or **Regulatory Guide 1.183**.	Y	12.3.5.2
	Coolant and corrosion activation products source terms should be based	Y	11.1.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	on applicable reactor operating experience.		Tables 11.1-2 & 11.1-4
	The buildup of activated corrosion products in various components and systems should be addressed.	Y	12.2.1.2 Table 12.2-5
	Any allowances made in design source terms for the buildup of activated corrosion products should be explained.	N/A-OTHER (No specific allowances)	N/A
	Neutron and prompt gamma source terms should be based on reactor core physics calculations and applicable reactor operating experience.	Y	12.2.1.1 Tables12.2.-1 & 12.2-2
	The tables of source parameters, which can be placed in SAR Chapter 12 or referenced to SAR Chapter 11, will be acceptable if the accompanying text either in this section or other referenced sections makes it clear how the values are used in a shield design calculation or in a ventilation system design.	Y	
	In addition, the quantities will be acceptable if the specific values given in the tables are consistent with **ANSI/ANS Standard 18.1** and **Regulatory Guide 1.112** for coolant and corrosion activation products source terms.	Y	11.1.2.1
	For PWRs designed for the recycling of tritiated water, tritium concentrations in contained sources and airborne concentrations in the regions specified in item I.2 above should be based on a primary coolant concentration of 1.3x10 Bq/gm (3.5 µCi/gm)	N/A-OTHER (Not designed for Tritium recycling)	N/A
SRP 12.3 -12.4	**Radiation Protection Design Features (R3, 03/2007)**		
12.3-12.4-AC-01	**10 CFR 20.1101(b)** and the definition of ALARA in **10 CFR 20.1003**, as	N/A-COL	12.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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Radiation Protection

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	they relate to persons involved in licensed activities making every reasonable effort to maintain radiation exposures ALARA		
12.3-12.4-AC-02	**10 CFR 20.1201**, as it relates to occupational dose limits for adults.	Y	12.1.2
		N/A-COL	12.1.3 12.5
12.3-12.4-AC-03	**10 CFR 20.1201**, **10 CFR 20.1202**, **10 CFR 20.1203**, **10 CFR 20.1204**, **10 CFR 20.1701**, and **10 CFR 20.1702**, as they relate to design features, ventilation, monitoring, and dose assessment for controlling the intake of radioactive materials	Y	12.3.3 12.3.4
		N/A-COL	12.5
12.3-12.4-AC-04	**10 CFR 20.1301** and **10 CFR 20.1302**, as they relate to the facility design features that impact the radiation exposure to a member of the public from noneffluent sources associated with normal operations and anticipated operational occurrences	Y	12.3.5.3
		N/A-COL	12.5
12.3-12.4-AC-05	**10 CFR 20.1406**, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the generation of radioactive waste	Y	12.3.6
12.3-12.4-AC-06	**10 CFR 20.1601**, **10 CFR 20.1602**, **10 CFR 20.1901**, **10 CFR 20.1902**, **10 CFR 20.1903**, and **10 CFR 20.1904**, as they relate to the identification of potential sources of radiation exposure and the controls of access to and work within areas of the facility with a high potential for radiation exposure	Y	12.3.1.8
		N/A-COL	12.5
12.3-12.4-AC-07	**10 CFR 20.1801**, as it relates to securing licensed materials against unauthorized removal from the place of storage	N/A-COL	12.5
12.3-12.4-AC-08	**General Design Criterion (GDC)19** found in Appendix A to 10 CFR Part	Y	12.3.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	50, as it relates to the provision of adequate radiation protection to permit access to areas necessary for occupancy after an accident, without personnel receiving radiation exposures in excess of 50 millisievert (mSv) (5 rem) to the whole body or the equivalent to any part of the whole body for the duration of the accident in accordance with 10 CFR 50.34(f)(vii)1		
12.3-12.4-AC-09	**GDC 61**, as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems that may contain radioactivity, designed to ensure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems	Y	5.4.7 9.1.3 12.1.2.1.2 12.3.1 12.3.2.3 12.3.3 12.3.5.2
12.3-12.4-AC-10	**GDC 63**, as it relates to detecting excessive radiation levels in the facility	Y	12.3.4.1 12.3.4.2 Tables 12.3-2 & 12.3-3
12.3-12.4-AC-11	**10 CFR 50.68**, as it relates to procedures and criteria for radiation monitoring in areas where special nuclear material is stored and handled	Y	12.3.4.4
12.3-12.4-AC-12	**10 CFR 52.47(b)(1)**, which requires that a DC FSAR contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
12.3-12.4-AC-13	**10 CFR 52.80(a)**, which requires that a COL FSAR contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations	N/A-COL	N/A
12.3-12.4-SAC-01	Facility Design Features The acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of **10 CFR 20.1201**, **10 CFR 20.1202**, **10 CFR 20.1203**, **10 CFR 20.1204**, and **10 CFR 20.1207**, as well as the radiation protection aspects of **GDC 19 and 61**, and **10 CFR 50.34**. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling and processing, inservice inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design and that radiation protection features incorporated into the design will keep potential radiation exposure from these activities ALARA in accordance with **10 CFR 20.1101(b)**, the definition of ALARA in **10 CFR 20.1003**, and **Regulatory Guides 8.8 and 8.10**. Such features may include (1) the ease of accessibility to work, inspection, and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products, (4) the ability to reduce time required in radiation fields, and (5) a provision for portable shielding and remote handling tools. Access control will be judged for acceptability in	Y N/A-COL	12.3.3 12.3.4 12.3.5.2 12.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	accordance with the requirements of **10 CFR 20.1601**, **10 CFR 20.1602**, **10 CFR 20.1901**, **10 CFR 20.1902**, and **10 CFR 20.1903** or access control alternatives in Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, NUREG-1433, and NUREG-1434).		
	Facility design, to the extent practicable, should minimize the potential for creating a very high radiation area during normal operations, including abnormal operational occurrences (such as dropping a fuel bundle during fuel handling operations).	Y	12.3.1 12.3.2
	High and very high radiation areas should be remote from normally occupied rooms and corridors such that personnel access to these areas can be controlled in accordance with **10 CFR 20.1601** and **10 CFR 20.1602** and the guidance in **Regulatory Guide 8.38**.	Y	12.3.1.8
		N/A-COL	12.5
	All accessible portions of the spent fuel transfer tube or canal that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) should be shielded during fuel transfer. This shielding should be such that the resultant contact radiation levels are no greater than 1 Gy per hour (100 rads per hour).	Y	12.3.1.8.2 Figures 12.3-34 & 12-3-35
	All accessible portions of the spent fuel transfer tube are clearly marked	Y	Figures 12.3-13

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	with a sign stating that potentially lethal radiation fields are possible during fuel transfer. If removable shielding is used for the fuel transfer tubes, it must also be explicitly marked as above. If other than permanent shielding is used, local audible and visible alarming radiation monitors must be installed to alert personnel if temporary fuel transfer tube shielding is removed during fuel transfer operations. Similar precautions should also apply to any other plant radiation source having radiation levels greater than 1 Gy per hour (100 rads per hour).		through 12.2-57
		N/A-COL	12.5
	The areas inside the plant structures, as well as in the general plant yard, should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified. Maximum zone dose rates should be defined for each zone, depending on anticipated occupancy and access control. The areas that must be occupied on a predictable basis (based on the number of people and stay or transit times) during normal operations and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling; processing, use, storage, and disposal; normal maintenance; routine operational surveillance; inservice inspection; and calibration) should be zoned such that this occupancy results in an annual dose to each of the involved individuals that is as far below the limits of **10 CFR Part 20** as is reasonably achievable, and a total person-sievert (person-rem) dose that is ALARA. Based on current operating experience and on predictions being made for new plant designs, it is expected that the plant shielding can be designed, the plant can be zoned, and sufficient radiation protection design features can be incorporated, such that individuals in shielded areas would receive a small fraction of the **10 CFR Part 20** limits.	Y	12.3.2.3 Figures 12.3-13 through 12.2-57

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	All vital areas, in which radiation may unduly limit personnel occupancy during operations following an accident resulting in a degraded core, should be identified. Personnel access to these areas under accident conditions should be demonstrated in accordance with **10 CFR 50.34(f)(2)(vii)**, using the methods listed in Section II.B.2 of **NUREG-0737**. The analysis should consider access to, stay time in, and egress from these vital areas.	Y	12.3.5.2
12.3-12.4-SAC-02	Shielding The staff will evaluate the shielding design in terms of the assumptions used to calculate shield thickness, the calculational methods used, and the parameters chosen. A number of acceptable shielding calculational codes are available that are effective for determining the necessary shield thickness for gamma ray and combination neutron-gamma sources. The code description file of the Radiation Safety Information Computational Center (formerly the Radiation Shielding Information Center) at Oak Ridge National Laboratory includes most of the codes used by shield designers, which means that the codes have been tested and authenticated for operation but not for reliability and accuracy. Radiation shielding codes vary in complexity and accuracy from the relatively simple point-kernel methods, to the more complex discrete ordinates methods, to the still more rigorous Monte Carlo methods. The staff may use these codes, as necessary, to calculate dose rates for given shield designs and source strengths as a confirmation of the applicant's method.	Y	12.3.2.3 Figures 12.3-13 through 12.2-57
	The applicant's shielding design is acceptable if the methods are comparable to commonly accepted shielding calculations and if	Y	12.3.2.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	assumptions regarding source terms, cross sections, shield and source geometries, and transport methods are realistic.		
	Labyrinth shielded access ways and penetrations should be used to minimize radiation steaming and scatter around shields.	Y	12.3.1.7.4 12.3.2.1
	Composition of the shielding material should be selected to minimize, to the extent practicable, the potential for the shield itself to become a radiation source (either from activation of the shield material or production of secondary radiation resulting from interactions with the primary radiation). Effective shield design is essential to meeting the criteria that ORE will be ALARA. In addition, **Regulatory Guide 1.69** and **ANSI/ANS-6.4-1997** provide guidance on the fabrication and installation of concrete shields for occupational radiation protection at nuclear power plants. Acceptability of the shield construction will be based on an indication that the guidance of these documents have been implemented in facility construction, or that acceptable alternatives have been proposed.	Y	12.3.2.1
	Regulatory Guide 8.8 provides additional acceptance criteria regarding shielding and isolation in radiation protection design.	Y	12.1.1.2 12.3.1
12.3-12.4-SAC-03	<u>Ventilation</u> The ventilation system will be acceptable for radiation protection purposes if the criteria and bases for ventilation rates within the areas covered in SAR Section 12.2.2 will ensure that air will flow from areas of low potential airborne radioactivity to areas of higher airborne radioactivity and then to filters or vents, that the concentrations of radioactive material in areas normally occupied can be maintained in	Y	12.3.3.3 Table 12.2-18

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	accordance with the requirements **10 CFR 20.1701**, and that the dose limits of **10 CFR 20.1201** are met consistent with the requirements of **10 CFR 20.1202**, **10 CFR 20.1203**, and **10 CFR 20.1204**.		
	The system has adequate capability to reduce concentrations of airborne radioactivity to 1.0 derived air concentration (DAC), as specified in **Appendix B to 10 CFR Part 20**, in areas not normally occupied where maintenance or inservice inspection must be performed.	Y	Table 12.2-18
	The system is designed so that filters containing radioactivity can be easily maintained and will not create an additional radiation hazard to personnel maintaining them, or those in adjacent occupied areas.	Y	12.3.3.4
	Acceptability of the ventilation system, relative to radioactive gases and particulates, will also be based on evidence that the applicant has applied the guidance of **Regulatory Guide 8.8** or proposed acceptable alternatives.	Y (Other RGs used bound guidance in RG 8.8)	12.3.3
	Regulatory Guide 1.52, particularly Sections C.3.10 and 4.10, provides guidance that can be used in this review, although the guide relates to mitigating accidents involving airborne radioactivity. Good practices in that regard apply to normal operation as well, since the release of radioactivity in normal operational occurrences is usually different only in quantity from some of the accident cases.	Y	6.5.1.1 12.3.3.3
12.3-12.4-SAC-04	<u>Area Radiation and Airborne Radioactivity Monitoring Systems</u>		
	A. The area radiation monitoring systems will be acceptable if they meet the provisions of **10 CFR 20.1501**, **10 CFR 50.34(f)(2)(xvii)**; the guidance in **NUREG-0737**, **Regulatory Guide 8.25**, and **Regulatory Guide 1.97**, Revision 3; and the following criteria:	Y	12.3.4 12.3.4.1.3 12.3.4.1.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12
Radiation Protection

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	i. The detectors are located in areas that normally may be occupied without restricted access and that may have a potential for radiation fields in excess of the radiation zone designations discussed in the third paragraph under item 1, above, in accordance with **ANSI/ANS-HPSSC-6.8.1**.	Y	12.3.4.1.1 Tables 12.3-2 & 12.3-3
	ii. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located as well as the maximum dose rate for anticipated operational occurrences and accidents.	Y	12.3.4.1.1 12.3.2.3
	iii. The detectors are calibrated during fuel outages and after the performance of any maintenance work on the detector..	N/A-COL	12.5
	iv. Each monitor has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms	Y	12.3.4
	v. Readout and annunciation are provided in the control room.	Y	12.3.4
	vi. The in-containment high-range radiation monitors meet the criteria of **10 CFR 50.34(f)(2)(xvii)**.	Y	12.3.4.1.3
	vii. Emergency power is initiated after a loss of offsite power	Y	12.3.4.1.2 12.3.4.1.3
	B. The airborne radioactivity monitoring system will be acceptable if it is consistent with the guidance on continuous air sampling in **Regulatory Guide 8.25** and meets the following criteria:.	Y N/A-COL	12.3.4 12.5
	i. Engineering controls provide the principal protection against the intake of radioactive materials	Y	12.3.3.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	ii. Air should be sampled at normally occupied locations where airborne radioactivity may exist, such as solid waste handling areas, spent fuel pools, reactor operating floors, and BWR turbine buildings. The monitoring system should be capable of detecting 10 DAC-hours of particulate and iodine radioactivity from any compartment that has a possibility of containing airborne radioactivity and that normally may be occupied by personnel, taking into account dilution in the ventilation system. Continuous monitoring of air being exhausted from locations within the facility during normal operation is an acceptable method.	Y	12.3.4.2.1 Table 12.3-3
	Noble gas monitors should be calibrated such that, when monitoring for 133Xe, the instrument response will determine concentrations accurately.	N/A-COL	12.5
	iii. Representative air concentrations are measured at the detectors, which are located as close to the sampler intakes as possible.	Y	12.3.4.2.1
	iv. Ventilation monitors are upstream of high-efficiency particulate air filters.	Y	12.3.4.2.1
	v. The detectors are calibrated routinely and after any maintenance work is performed on the detector.	N/A-COL	12.5
	vi. Each location has a local audible alarm and variable alarm set points. Monitors located in high noise areas should also have visual alarms.	Y	12.3.4
	vii. Readout and annunciation are provided in the control room.	Y	12.3.4
	viii. Emergency power is initiated after a loss of offsite power.	Y	12.3.4.2.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
			12.3.4.2.2 12.3.4.2.3
	C. The in-plant accident radiation monitoring systems will be acceptable if they meet the following criteria:		
	i. Personnel have the capability to assess the radiation hazard in areas that may be accessed during the course of an accident, in accordance with the criteria of **10 CFR 50.34(f)(2)(xvii)**; **NUREG-0737**, item II.F.1; and **Regulatory Guide 1.97**, Revision 3.	Y	12.3.4
	ii. Portable instruments to be used in the event of an accident should be placed so as to be readily available to personnel responding to an emergency.	N/A-COL	12.5
	iii. Emergency power should be provided for installed accident monitoring systems.	Y	12.3.4.1.2 12.3.4.1.3 12.3.4.2.2
	iv. The accident monitoring systems should have usable ranges that include the maximum calculated accident levels and should be designed to operate properly in the environment caused by the accident.	Y	Tables 12.3-2 & 12.3-3
	v. Two high-range radiation monitors are provided in containment in accordance with the requirements of **10 CFR 50.34(f)(2)(xvii)** and item II.F.1 of **NUREG-0737**.	Y	12.3.4.1.3
	D. **Appendix A to Regulatory Guide 1.21** provides useful guidance about effluent monitoring that applies to the acceptability of in-plant airborne radioactivity monitoring. **Regulatory Guide 8.2** includes	N/A-COL	12.3.4.5

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CHAPTER 12 Radiation Protection			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	guidance on surveys to evaluate radiation hazards. The detailed guidance in **ANSI N13.1-1999** covers the sampling of airborne radioactive materials in ventilation ducts and stacks of nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. **Regulatory Guide 8.8** provides further guidance on monitoring systems.		
	E. Instrumentation for monitoring areas where reactor fuel is stored or handled will be acceptable if it meets the criteria of **10 CFR 50.68**.	Y	12.3.4.4
12.3-12.4-SAC-05	<u>Dose Assessment</u> The dose assessment will be acceptable if it documents in appropriate detail the assumptions made, calculations used, results for each radiation zone (including numbers and types of workers involved in each), expected and design dose rates, and projected person-Sievert (person-rem) doses, in accordance with **Regulatory Guide 8.19**.	Y	12.3.5.1 Tables 12.3-4 through 12.3-10
SRP 12.5	**Operational Radiation Protection Program (R3, 03/2007)**	N/A-COL	12.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 13
Conduct of Operations

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 13.1.1	**Management and Technical Support Organization (R5, 03/2007)**	N/A-COL	13.1
SRP 13.1.2 - 13.1.3	**Operating Organization (R6, 03/2007)**	N/A-COL	13.1
SRP 13.2.1	**Reactor Operator Requalification Program; Reactor Operator Training (R3, 03/2007)**	N/A-COL	13.2
SRP 13.2.2	**Non-Licensed Plant Staff Training (R3, 03/2007)**	N/A-COL	13.2
SRP 13.3	**Emergency Planning (R3, 03/2007)**		
13.3-AC-A	**10 CFR 50.33**, **10 CFR 50.34**, **10 CFR 50.47**, **10 CFR 100.1**, **10 CFR 100.3**, **10 CFR 100.20**, and **10 CFR 100.21(g)**, as they relate to emergency planning and preparedness.	N/A-COL	13.3
13.3-AC-B	**10 CFR Part 50, Appendix E**, as it relates to emergency planning and preparedness, and the emergency response data system (ERDS) [or successor system to ERDS].	Y	13.3
13.3-AC-C	**10 CFR 52.17** and **10 CFR 52.18**, as they relate to emergency planning information submitted in an ESP application. **10 CFR 52.17(b)(3)** provides the requirement for ITAAC in an ESP application submitted under **10 CFR 52.17(b)(2)**.	N/A-ESP	N/A
13.3-AC-D	**10 CFR 52.47** and **10 CFR 52.48**, as they relate to emergency planning information submitted in a standard design certification application. **10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	N/A-COL (Deferred to applicant per SRP 13.3, 13.3-SAC-23 as permitted under SRP 13.3, 13.3-SAC-22)	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 13
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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	the Atomic Energy Act, and the NRC's regulations.		
13.3-AC-E	**10 CFR 52.77**, **10 CFR 52.79**, **10 CFR 52.80**, **10 CFR 52.81**, and **10 CFR 52.83**, as they relate to emergency planning and preparedness associated with a COL application. **10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	13.3
13.3-AC-F	**10 CFR 50.72(a)(3)**, **10 CFR 50.72(a)(4)**, **10 CFR 50.72(c)(3)**, and **10 CFR 73.71(a),** as they relate to notification of NRC for an emergency class declaration, ERDS notification, and requirements for reporting safeguards events and maintaining an open emergency notification system (ENS) line.4	N/A-COL	13.3
13.3-AC-G	**Interim Compensatory Measures (ICMs) B.5.c, B.5.d, and B.5.e of Commission Orders of February 25, 2002**, to all operating commercial nuclear power plants, relating to security-based emergency plans and preparedness.	N/A-COL	13.3
13.3-AC-H	**44 CFR Parts 350, 351, and 352**, including applicable FEMA policies, REP-series guidance documents and associated memoranda, as they relate to offsite radiological emergency planning and preparedness.	N/A-COL	13.3
13.3-SAC-01	All of the standards of **10 CFR 50.47(b)**, as supported by the guidance in the corresponding planning standards and evaluation criteria of **NUREG-0654/FEMA-REP-1**, Rev. 1, (including the March 2002 addenda) must be	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	met before an OL is issued pursuant to 10 CFR 50.57 or a COL is issued pursuant to 10 CFR 52.97. In addition, for the first reactor at a site. Appendix E to 10 CFR Part 50 requires that a full participation exercise be conducted within 2 years before NRC issuance of an operating license for full power (i.e., one authorizing operation above 5 percent of rated power). Because this exercise would be included in the ITAAC required for a COL, it's acceptance criteria would have to be satisfied before fuel loading pursuant to a COL (see Table 14.3.10-1).		
13.3-SAC-02	The onsite and, except as provided in **10 CFR 50.47(d)**, offsite emergency response plans for nuclear power reactors must meet the standards established in **10 CFR 50.47(b)** and applicable requirements of **Appendix E to 10 CFR Part 50**. Compliance with these regulations is determined by using the guidance in **Regulatory Guide (RG) 1.101**, Rev. 2, which endorses **NUREG-0654/FEMA-REP-1**, Rev. 1, and through it **NUREG-0396**, and **NUREG-0696**. **NUREG-0654/FEMA-REP-1**, Rev. 1, establishes an acceptable basis for NRC licensees and State, tribal and local governments to develop radiological emergency plans and procedures, and improve their overall state of emergency preparedness. **NUREG-0696** discusses the facilities and systems to be provided by nuclear power plant licensees to aid the licensee's response to emergency situations. Additional guidance is provided in **NUREG-0718**, **NUREG-0737**, **Supplement 1 to NUREG-0737**, **NUREG-0814**, and **Supplement 3 to NUREG-0654/ FEMA-REP-1**, Rev. 1.	N/A-COL	13.3
13.3-SAC-03	**10 CFR 50.47(b)(4)** requires a standard emergency classification and action level scheme. **Section IV.C, "Activation of Emergency Organization," of Appendix E identifies the four emergency classes. Section IV.B, "Assessment Actions," of Appendix E to 10 CFR Part**	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 13 Conduct of Operations			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	50 also requires emergency action levels. The emergency plan should include the emergency classification level scheme described in **Appendix 1 and Supplement 3 to NUREG-0654**. The staff anticipates that any new application will use an emergency action level scheme similar to that described in Revision 4 of **NEI 99-01**, “Methodology for Development of Emergency Action Levels,” dated January 2003, which was endorsed in Revision 4 **Regulatory Guide (RG) 1.101**, “Emergency Planning and Preparedness for Nuclear Power Reactors,” dated October 2003. However, Revision 4 of NEI 99-01, “Methodology for Development of Emergency Action Levels,” dated January 2003, is not considered to be entirely applicable to advanced light water reactor designs. Even though the majority of Revision 4 of NEI 99-01 may be applicable to any reactor design and should be used, the unique characteristics of the new reactor should be addressed in the development of emergency action levels specific to the new plant and the site. The format of the emergency action level scheme should follow the convention established in **Regulatory Information Summary 2003-18**, “Use of Nuclear Energy Institute (NEI) 99-01, Methodology for Development of Emergency Action Levels,” Revision 4, dated January 2003, and its supplements. **Section IV.B. “Assessment Actions,” of Appendix E to 10 CFR Part 50** also requires that the initial emergency actions be discussed and agreed on by the State and local governmental authorities. The applicant should provide some form of confirmation of the agreement, such as a letter signed by State and local governmental authorities, in the emergency plan, if the applicant provides emergency action levels different from those for the existing reactor(s) on the site.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
13.3-SAC-04	**Appendix 2, "Meteorological Criteria for Emergency Preparedness at Operating Nuclear Power Plants," to NUREG-0654/FEMA-REP-1**, Rev. 1, provides guidance related to the planning standards codified in **10 CFR 50.47(b)(8) and (9)** and the requirements of **Section IV.E.2 of Appendix E to 10 CFR Part 50**. Proposed revision 1 to **Regulatory Guide 1.23**, "Meteorological Programs in Support of Nuclear Power Plants," is referenced in Appendix 2 to NUREG-0654 as a source of acceptance criteria for meteorological measurements. Since Appendix 2 was issued, additional guidance related to meteorological systems has been developed. **NUREG-0696**, "Functional Criteria for Emergency Response Facilities," refers to the guidance in proposed Revision 1 to **Regulatory Guide 1.23**, Revision 2 to **Regulatory Guide 1.97**, and **Appendix 2 to NUREG-0654/FEMA-REP-1**, Rev. 1. **Supplement 1 to NUREG-0737**, "Clarification of TMI Action Plan Requirements," (**Generic Letter 82-33**) clarifies the guidance in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," and contains guidance related to the need to provide reliable indication of meteorological variables in the control room, Technical Support Center, and Emergency Operations Facility in the vicinity (up to about 10 miles) of the plant site. Revision 3 of **Regulatory Guide 1.97** was issued in May 1983 and Revision 4 was issued in June 2006. Revision 1 to **Regulatory Guide 1.23** was issued in March 2007.	N/A-COL	13.3
13.3-SAC-05	**Supplement 1 to NUREG-0737**, "Clarification of TMI Action Plan Requirements," (**Generic Letter 82-33**) clarifies the guidance in Revision 2 of **Regulatory Guide 1.97**, "Instrumentation for Light-water-cooled Nuclear Power Plants to Assess Plant and Environs Conditions During	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	and Following an Accident," and contains guidance related to upgrading emergency response facilities and meeting the requirements of **10 CFR 50.47(b)(6), (8), (9)** and **Section IV.E of 10 CFR Part 50**.		
13.3-SAC-06	**Appendix 3, "Means for Providing Prompt Alerting and Notification of Response Organizations and the Population," to NUREG-0654/FEMA-REP-1**, Rev. 1, provides guidance related to **10 CFR 50.47(b)(5) and (6)**.	N/A-COL	13.3
13.3-SAC-07	**Supplement 3, "Criteria for Protective Action Recommendations for Severe Accidents," to NUREG-0654/FEMA-REP-1**, Rev.1, provides guidance for the development of protective action recommendations for the public for severe reactor accidents. The guidance updates and simplifies the decision-making process for protective actions for severe reactor accidents given in Appendix 1 to NUREG-0654/FEMA-REP-1, Rev.1.	N/A-COL	13.3
13.3-SAC-08	**RG 1.101**, Rev. 2, states that the criteria and recommendations in **NUREG-654/FEMAREP- 1**, Rev. 1, are considered by the NRC staff to be acceptable methods for complying with the standards in **10 CFR 50.47**. Except in those cases in which the applicant or licensee proposes acceptable alternative methods for complying with specific portions of the regulations, the methods described in NUREG-0654/FEMA-REP-1, Rev. 1, will be used as a basis for evaluating the adequacy of the emergency plans. If an applicant chooses to propose an alternative practice or method for complying with the regulations, the application should provide an appropriate justification.	N/A-COL	13.3
13.3-SAC-09	In addition to **NUREG-0654/FEMA-REP-1**, Rev. 1, FEMA will evaluate State, tribal, and local government planning and preparedness on the	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	basis of applicable policies and guidance, including approved alternative approaches and methods. FEMA will base its findings and determinations, relating to the adequacy of offsite radiological emergency planning and preparedness, on these evaluations.		
13.3-SAC-10	**10 CFR 50.33(g)**, **10 CFR 50.47(c)(2)**, and **Section I of Appendix E to 10 CFR Part 50** require that the size of the EPZ for a nuclear power plant shall be determined in relation to local emergency response needs and capabilities, as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. **10 CFR 52.77** requires that the COL application must contain all of the information required by **10 CFR 50.33**. **10 CFR 50.33(g)** requires that an applicant for an operating license submit radiological emergency response plans of State and local government entities that are wholly or partially within the 10-mile plume exposure EPZ, as well as the plans of State governments wholly or partially within the 50- mile ingestion pathway EPZ. An applicant should also submit plans for tribal governmental entities affected by the 10-mile EPZ. NUREG-0396 provides additional guidance relating to the definition of the EPZs.	N/A-COL	13.3
13.3-SAC-11	**Section IV of Appendix E to 10 CFR Part 50**, through **10 CFR 52.79(a)(21)** and **10 CFR 50.34**, requires that an application for an OL or COL provide an analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ; i.e., an ETE. The NRC regulations do not specify a limit for such estimated evacuation times. An ETE can identify physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. An ETE provides an analysis of the time required to evacuate and for taking other protective actions for	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	various sectors and distances within the plume exposure EPZ. This information can be used by decision makers in responding to an actual emergency to aid in deciding what protective actions to implement. **Appendix 4 to NUREG-0654/FEMAREP -1**, Rev. 1, and **Supplement 2 to NUREG-0654/FEMA-REP-1**, Rev. 1, provide guidance relating to performing an ETE analysis. **NUREG/CR-6863** provides additional information on ETEs.		
13.3-SAC-12	**Section VI of Appendix E to 10 CFR Part 50** requires an emergency response data system (ERDS). The ERDS is a direct near real-time electronic data link between a licensee's onsite computer system and the NRC Operations Center, and provides for the automated transmission of a limited data set of selected parameters from a licensee's installed onsite computer system in the event of an emergency. **NUREG-1394** provides the minimum standards and acceptable methods that may be used to implement and comply with the ERDS requirements.	N/A-COL	13.3
13.3-SAC-13	Insofar as emergency planning and preparedness requirements are concerned, **10 CFR 50.47(d)** provides that a license authorizing fuel loading and/or low-power testing and training (up to 5 percent of the rated power) may be issued after a finding is made by the NRC that the state of onsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. The assessment of the applicant's onsite emergency plan will be based on the pertinent standards in **10 CFR 50.47(b)** and the requirements of **Appendix E to 10 CFR Part 50**. However, the acceptability of an applicant's emergency plans will be reviewed against the standards with offsite aspects presented in **10 CFR 50.47(d)(1)-(7)**.	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
13.3-SAC-14	Where an applicant for an OL or COL asserts that its inability to demonstrate compliance with the offsite emergency planning requirements of **10 CFR 50.47(b)** is wholly or substantially the result of the non-participation of State and/or local governments, an operating license may be issued if the applicant demonstrates to the Commission's satisfaction those elements listed in **10 CFR 50.47(c)(1)(i)-(iii)**. (See **10 CFR 50.47(c)(1)** and **10 CFR 52.79(a)(22)(ii)**.) **Supplement 1 to NUREG-0654/FEMA-REP-1**, Rev. 1, provides guidance for the development, review, and evaluation of utility offsite radiological emergency response planning and preparedness, for those situations in which State and/or local governments decline to participate in emergency planning.	N/A-COL	13.3
13.3-SAC-15	The minimum acceptance criteria for all ESP applications, located in **10 CFR 52.17(b)(1)**, require that ESP applications identify physical characteristics unique to the proposed site that could pose a significant impediment to the development of emergency plans. If such physical characteristics are identified, the applicant must also identify measures that would, when implemented, mitigate or eliminate the significant impediment. Applications providing only the information required by 10 CFR 52.17(b)(1) must also include a description of contacts and arrangements (preferably letters of agreement) made with local, State, and Federal governmental agencies with emergency planning responsibilities, in accordance with 10 CFR 52.17(b)(4). The applicant may choose to submit additional emergency planning information in the ESP application to address the two options in 10 CFR 52.17(b)(2). The two options allow an ESP applicant to propose either major features of the emergency plans, or to provide complete and integrated emergency	N/A-ESP	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	plans. While neither option is required, each would provide for a more definitive finding concerning emergency plans and preparedness at the ESP stage than would be the case for submittal of only the minimum required information. Complete and integrated emergency plans in an ESP application will be reviewed in accordance with the applicable requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50. Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, provides guidance relating to emergency planning information in an ESP application.		
13.3-SAC-16	For an ESP application, a preliminary analysis of evacuation times is one example of how some significant impediments to the development of emergency plans may be identified. Other factors, such as the availability of adequate shelter facilities, in consideration of local building practices and land use (e.g., outdoor recreation facilities, including camps, beaches, hunting or fishing areas), and the presence of large institutional or other special needs populations (e.g., schools, hospitals, nursing homes, prisons) should also be addressed when identifying significant impediments to the development of emergency plans. Any ETE analysis or other identification of physical impediments should include the latest population census numbers and reflect the most recent local conditions. Appendix 4 to NUREG-0654/FEMA-REP-1, Rev. 1, and Supplement 2 to NUREG- 0654/FEMA-REP-1, Rev. 1, provide guidance relating to performing an ETE analysis. NUREG/CR-6863 provides additional information on ETEs.	N/A-ESP	N/A
13.3-SAC-17	For applications that require site approval for a stationary power reactor subject to 10 CFR Part 50 or **10 CFR Part 52** (e.g., CP, OL, ESP and COL), **10 CFR 100.1** and **10 CFR 100.21(g)** require the identification of physical characteristics unique to the proposed site that could pose a	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	significant impediment to the development of emergency plans. This siting requirement is similar to that in 10 CFR 52.17(b)(1) for an ESP application, and the means for identifying significant impediments (e.g., an analysis of evacuation times or ETE) could apply to non-ESP applications. Further, if such physical characteristics are identified, the application must also identify measures that would, when implemented, mitigate or eliminate the significant impediment. Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards (see 10 CFR 100.10(d), which applies to applications submitted before January 10, 1997). The application should provide a projection of the population within the 10-mile EPZ throughout the requested duration of the application; including a discussion of the sources of information and methodology that supports the population projection. The application should specifically address whether the projected population creates a significant impediment to the development of emergency plans over the requested duration of the ESP or COL application, including how it would affect the ETE. If a significant impediment is created, then the applicant should identify measures that would, when implemented, mitigate or eliminate the significant impediment. Additional site-related guidance is provided in **RG 4.7**, and in ESP-related guidance documents (e.g., Supplement 2 to NUREG-654/FEMA-REP-1, Rev. 1).		
13.3-SAC-18	Copies of letters of agreement or other certifications, reflecting contacts and arrangements made with local, State, and Federal agencies with supporting emergency responsibilities, should be included in a CP, OL,	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	ESP or COL application, as required by **10 CFR 52.17(b)(4)**, **10 CFR 52.79(a)(22)**, or **Section II.B of Appendix E to 10 CFR Part 50.9** The agreement information should be up-to-date when the application is submitted, and should reflect use of the proposed site for possible construction of a new reactor (or reactors). In addition, a discussion of the details associated with any ambiguous or incomplete language in the letters of agreement should be provided in the application. For an existing reactor site, the letters of agreement or other certifications should clearly address the presence of an additional reactor (or reactors) at the site, and any impact that would have on governmental agency or private organization emergency planning responsibilities, including acknowledgment by the agencies or organization of the proposed expanded responsibilities. If the applicant is unable to make arrangements with local, tribal, State, and Federal governmental agencies with emergency planning responsibilities, for whatever reason, the applicant should discuss its efforts to make such arrangements and describe any compensatory measures the applicant has taken or plans to take because of the lack of such arrangements. **Supplement 1 to NUREG-654/FEMA-REP-1**, Rev. 1, provides guidance for the development, review, and evaluation of utility offsite radiological emergency response planning and preparedness (i.e., a utility plan), for those situations in which State and/or local governments decline to participate in emergency planning. (See also **10 CFR 50.47(c)(1)**.)		
13.3-SAC-19	**Supplement 2 to NUREG-0654/FEMA-REP-1**, Rev. 1, will be used as the primary guidance for the review of emergency preparedness information and plans submitted with an ESP application pursuant to **Subpart A of 10 CFR Part 52**. For a pre-existing nuclear facility, all	N/A-ESP	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	major features of the emergency plan (i.e., all 14 planning standards) identified in Supplement 2 to NUREG-0654/FEMA-REP-1, Rev. 1, should be addressed in the ESP application. The detailed, specific evaluation criteria for each of the major features in Supplement 2 should be addressed for both a pre-existing nuclear facility, as well as for applicable major features associated with a site without a pre-existing nuclear facility. If emergency planning information is not provided on all 14 major features (including the detailed, specific evaluation criteria) in Section V of Supplement 2, the ESP application will not be rejected. The review and evaluation will, however, only be based on, and specifically limited to, the submitted information that relates to the guidance in Supplement 2 of NUREG-0654/FEMA-REP-1, Rev. 1.		
13.3-SAC-20	The planning standards and evaluation criteria for preparing and evaluating an ESP application containing complete and integrated emergency plans are provided in **NUREG-0654/FEMA-REP-1**, Rev. 1. Under this ESP option, the applicant should make a good-faith effort to obtain from the government agencies certifications that (1) the proposed emergency plans are practicable; (2) these agencies are committed to participating in any further development of the plans, including any required field demonstrations; and (3) these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility plan pursuant to 10 CFR 50.47(c)(1), sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site. The	N/A-ESP	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	utility-prepared emergency plans and preparedness will be reviewed and evaluated using the guidance in Supplement 1 to NUREG-0654/FEMA-REP-1, Rev. 1.		
13.3-SAC-21	**10 CFR 52.17(b)(3)** allows an applicant for an ESP, that proposes major features of the emergency plans or complete and integrated emergency plans, to include proposed ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-ESP	N/A
13.3-SAC-22	**10 CFR 52.47(b)(1)** allows an applicant for a design certification to include proposed ITAAC, including those applicable to emergency planning, which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	13.3
13.3-SAC-23	**10 CFR 52.80(a)** requires that an application for a combined license includes proposed emergency planning ITAAC which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL (Deferred to applicant as permitted under SRP 13.3, 13.3-SAC-22)	Tier 1
13.3-SAC-24	Table 14.3.10-1 provides an acceptable set of generic emergency	N/A-COL	13.3

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	planning ITAAC that an applicant may use to develop application-specific ITAAC, tailored to the specific reactor design and emergency planning program requirements. A smaller set of ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC in Table 14.3.10-1 that are not used. Table 14.3.10-1 is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific emergency planning ITAAC (i.e., beyond those listed in Table 14.3.10-1) may be proposed, and they will be examined to determine their acceptability on a case-by-case basis. In general, ITAAC are inappropriate for procedure-level details associated with the emergency plans, in that procedure adequacy and implementation can be evaluated under the exercise ITAAC, and should be limited to those aspects of emergency planning and preparedness that can not reasonably be addressed prior to construction of the plant. Each EP-ITAAC must have an objective acceptance criteria stated.		
13.3-SAC-25	For those licensees subject to 10 CFR 50.34(f), **10 CFR 50.34(f)(2)(xxv)** requires that an applicant provide a TSC, OSC, and, for a CP application only, a near-site emergency operations facility (EOF) (TMI Item III.A.1.2). **NUREG-0696**, **Appendix B to NUREG- 0718**, **NUREG-0737**, and **Supplement 1 to NUREG-0737** provide guidance relating to the design and implementation of emergency response facilities (e.g., TSC, OSC, EOF). In addition, **10 CFR 50.47(b)(8)** and **Subsection IV.E.8 of Appendix E to 10 CFR Part 50** requires that the design should include adequate emergency facilities and equipment to support emergency response. **NUREG-0696**, **NUREG-0737**, and **Supplement 1 to NUREG-0737** provide guidance relating to occupancy and radiological habitability	N/A-COL	13.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	of vital areas (including the TSC), which aid in the mitigation of or recovery from an accident.		
13.3-SAC-26	For those licensees subject to 10 CFR 50.34(f), **10 CFR 50.34(f)(2)(iv)** requires that an applicant seeking an operating license shall provide an SPDS in both the TSC and EOF (TMI Item I.D.2). The SPDS includes the minimum set of plant parameters needed to assess the safety status of the plant in a timely manner, and is capable of indicating when process limits are being approached or exceeded. **Supplement 1 to NUREG-0737**, **NUREG-0696**, and **NUREG-0814** provide guidance regarding the SPDS. (The SPDS is reviewed under **SRP Sections 7.5 and 18.2**.)	N/A-COL	13.3
13.3-SAC-27	For those licensees subject to 10 CFR 50.34(f), **10 CFR 50.34(f)(2)(viii)** requires that an applicant provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials, while ensuring that no individual receives radiation exposure in excess of 0.05 Sv (5 rem) to the whole body or 0.5 Sv (50 rem) to the extremities (TMI Item II.B.3). In addition, **10 CFR 50.47(b)(9)** requires adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition. To address this regulation, the NRC has concluded that source term information should be obtained and analyzed, to continuously assess and refine dose assessments and confirm or modify initial protective action recommendations. Finally, **10 CFR 50.47(b)(11)** requires the establishment of the means for controlling radiological exposure to emergency workers. Post-accident sampling systems are discussed in the October 31, 2000, Model Safety Evaluation, as it relates to the development of contingency plans for sampling and analysis of highly	N/A-COL	13.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 13 Conduct of Operations			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	radioactive samples from the reactor coolant system, containment sump, and containment atmosphere.		
13.3-SAC-28	For those licensees subject to 10 CFR 50.34(f), **10 CFR 50.34(f)(2)(xvii)** requires instrumentation to measure, record and readout of various containment parameters, including noble gas effluents at all potential, accident release points. In addition, an applicant must 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 (TMI Item II.F.1). **RG 1.97** provides guidance relating to instrumentation to assess plant and environmental conditions during and following an accident.	N/A-COL	13.3
13.3-SAC-29	**10 CFR 50.72(a)(3) and (c)(3)** require the notification of the NRC Operations Center following the declaration of an emergency in accordance with the licensee's approved emergency plans, and the establishment of an open and continuous communications channel when requested by the NRC. **10 CFR 50.72(a)(4)** establishes requirements for the activation of the ERDS following the licensee's declaration of an alert, site area emergency, or general emergency. **NUREG-1022** provides the minimum standards and acceptance methods that may be used to comply with these NRC reporting requirements. **10 CFR 73.71(a)** requires the notification of the NRC Operations Center, after the discovery of an imminent or actual safeguards threat against the facility or other safeguards events. **Regulatory Guide 5.62** provides the minimum standards and acceptance methods that may be used to comply with these NRC reporting requirements.	N/A-COL	13.3
13.3-SAC-30	The emergency planning and preparedness standards and requirements	N/A-COL	13.3

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CHAPTER 13 Conduct of Operations			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	in **10 CFR Part 50**, **10 CFR Part 52**, and **10 CFR Part 100** are supplemented by various generic communications and Commission Orders. Those generic communications that relate to emergency planning and are currently in effect are identified in Subsection VI (below). They provide additional guidance and criteria for meeting the relevant emergency planning standards and requirements. Any subsequently issued generic communications or Commission Orders that pertain to emergency planning and preparedness and are relevant to the application should also be addressed by the applicant.		
13.3-SAC-31	Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Emergency Planning program are reviewed in accordance with **10 CFR 50.47**, **Part 50 Appendix E**. The implementation milestones are as follows: full participation exercise conducted within 2 years of scheduled date for initial loading of fuel per **10 CFR 50, Appendix E.IV.F.2a(ii)**; onsite exercise conducted within 1 year before the schedule date for initial loading of fuel per 10 CFR Part 50, Appendix E.IV.F.2a(ii); and applicant's detailed implementing procedures for its emergency plan submitted no less than within 180 days prior to scheduled date for initial loading of fuel per **10 CFR Part 50, Appendix E.V**.	N/A-COL	13.3
SRP 13.4	**Operational Programs (R3, 03/2007)**	N/A-COL	13.4
SRP 13.5.1.1	**Administrative Procedures – General (03/2007)**	N/A-COL	13.5.1
SRP 13.5.2.1	**Operating and Emergency Operating Procedures (R1, 03/2007)**	N/A-COL	13.5.2
SRP 13.6	**Physical Security (03/2007)**	N/A-INFO	N/A

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CHAPTER 13 Conduct of Operations			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 13.6.1	**Physical Security – Combined License (03/2007)**	N/A-COL	13.6
SRP 13.6.2	**Physical Security – Design Certification (03/2007)**		
13.6.2-AC-01	**10 CFR Part 50**, "Domestic Licensing of Production and Utilization Facilities."	Y	13.6
		N/A-COL	13.6
13.6.2-AC-02	**10 CFR Part 52**, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."	Y	13.6
		N/A-COL	13.6
13.6.2-AC-03	**10 CFR 73.1(a)(1)** , "Radiological Sabotage."	Y	13.6
		N/A-COL	13.6
13.6.2-AC-04	**10 CFR 73.55**, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage," **and Appendices B, C, G and H**.	Y	13.6
		N/A-COL	13.6
13.6.2-AC-05	**10 CFR Part 74**, "Material Control and Accounting of Special Nuclear Material."	Y	13.6
		N/A-COL	13.6
13.6.2-AC-06	**10 CFR Part 100,** "Reactor Site Criteria."	Y	13.6
		N/A-COL	13.6
13.6.2-AC-07	**Regulatory Guide 1.70**, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, November 1978.	N/A-SUP (See RG1.206)	N/A
13.6.2-AC-08	**Regulatory Guide 1.91**, Evaluations of Explosions Postulated to Occur at Transportation Routes Near Nuclear Power Plants, February 1978.	N/A-COL	13.6
13.6.2-AC-09	**Regulatory Guide 4.7**, General Site Suitability Criteria for Nuclear Power Stations, April 1998.	N/A-COL	13.6

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CHAPTER 13 Conduct of Operations			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
13.6.2-AC-10	**Regulatory Guide 5.12**, General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials, November 1973.	Y	13.6
		N/A-COL	13.6
13.6.2-AC-11	**Regulatory Guide 5.65**, Vital Area Access Controls, Protection of Physical Security Equipment and Key and Lock Controls, September 1986.	Y	13.6
		N/A-COL	13.6
13.6.2-AC-12	**Regulatory Guide 5.7**, Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas, Revision 1, May 1980.	N/A-COL	13.6
13.6.2-AC-13	**Regulatory Guide 5.44**, Perimeter Intrusion Alarm Systems, Revision 3, October 1997.	N/A-COL	13.6
13.6.2-AC-14	**Information Notice No. 86-83**: Underground Pathways into Protected Areas, Vital Areas, and Controlled Access Areas, September 19, 1986.	N/A-COL	13.6
13.6.2-AC-15	**Regulatory Information Summary 2005-04**, Guidance on the Protection of Unattended Openings that Intersect a Security Boundary or Area, April 14, 2005.	N/A-COL	13.6
13.6.2-AC-16	**Regulatory Guide 5.29**, Material Control and Accounting for Nuclear Power Reactors.	Y	13.6
		N/A-COL	13.6
13.6.2-AC-17	**American National Standards Institute (ANSI) N15.8**, Nuclear Material Control Systems for Nuclear Power Plants, 1974.	Y	13.6
		N/A-COL	13.6
13.6.2-SAC-01	**Section (c) of 10 CFR 73.55** – Physical Barriers. The licensee shall locate vital equipment only within a vital area, which in turn, shall be located within a protected area such that access to vital equipment requires passage through at least two physical barriers as defined in **10 CFR 73.2**. The physical barriers at the perimeter shall be separated from	Y	13.6
		N/A-COL	13.6

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	any other barrier designated as a physical barrier for a vital area within the protected area. Isolation zones in outdoor areas adjacent to the physical barrier at the perimeter of the protected area permit observation. Intrusion detection system detects penetration or attempted penetration of the protected area (PA) barrier. All exterior areas within the protected area are illuminated. The external walls, doors, ceiling and floors in the main control room are bullet resistant. Vehicle control measures which include vehicle barrier systems protect against the use of land vehicle.		
13.6.2-SAC-02	**Section (d) of 10 CFR 73.55 -** Access Requirements. The licensee shall control all points of personnel and vehicle access into a protected area, to include detection equipment capable of detecting firearms, explosives and incendiary devices. Unoccupied vital areas are locked and alarmed with activated intrusion detection systems that annunciate in both the central and secondary alarm stations upon intrusion into a vital area. The individual responsible for the last access control function (controlling admission to the protected area) must be isolated within a bullet-resisting structure.	N/A-COL	13.6
13.6.2-SAC-03	**Section (e) of 10 CFR 73.55 –** Detection Aids. All alarms required pursuant to this part shall annunciate in a continuously manned central alarm station located within the protected area and in at least one other continuously manned station, not necessarily onsite, such that a single act cannot remove the capabilities of calling for assistance or otherwise responding to an alarm. The central alarm station shall be considered a vital area, shall be bullet-resisting, the interior will not be visible from the protected area perimeter, and associated onsite secondary power supplies for alarm annunciators and non-portable communication equipment must be located within vital areas. Alarm devices and	Y N/A-COL	13.6 13.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	transmission lines must be tamper indicating and self checking. Alarm annunciation shall indicate type of alarm and location. All emergency exits from protected and vital areas shall be alarmed.		
13.6.2-SAC-04	**Section (f) of 10 CFR 73.55 -** Communication Requirements. Each security officer, watchman or armed response individual shall be capable of maintaining continuous communications with an individual in each continuously manned alarm stations. Conventional telephone and radio or microwave transmitted two-way voice communications shall be established with local law enforcement authorities.	N/A-COL	13.6
13.6.2-SAC-05	**Section (g) of 10 CFR 73.55 -** Testing and Maintenance. Each applicant shall develop test and maintenance provisions for intrusion alarms, emergency alarms, communication equipment, access control equipment, physical barriers, and other security-related devices or equipment.	N/A-COL	13.6
SRP 13.6.3	**Physical Security – Early Site Permit (03/2007)**	N/A-ESP	N/A

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CHAPTER 14
Initial Test Program and ITAAC-Design Certification

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 14.2	**Initial Plant Test Program – Design Certification and New License Applicants (R3, 03/2007)**		
14.2-AC-01	**10 CFR 50.34(b)(6)(iii)**, which requires the applicant to provide plans for preoperational testing and initial operations.	Y	14.2.1 through 14.2.11
14.2-AC-02	**10 CFR 30.53(c)**, as it relates to testing radiation detection and monitoring instruments.	Y	14.2.12.12 Through 14.2.12.19
14.2-AC-03	**Section XI of Appendix B to 10 CFR Part 50**, as it relates to test programs established to assure that SSCs will perform satisfactorily in service.	Y	14.2.12.1 through 14.2.12.21
14.2-AC-04	**Section III.A.4 of Appendix J to 10 CFR Part 50**, as it relates to the preoperational leakage rate testing of the primary reactor containment and related systems and components penetrating the primary containment pressure boundary.	Y	14.2.12.3
14.2-AC-05	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	ITAAC	Tier 1
14.2-AC-06	**10 CFR 52.79(a)(28)**, which requires COL applicants to provide plans for preoperational testing and initial operations.	Y	14.2.4 through

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
			14.2.11
		N/A-COL	14.2.4 through 14.2.11
14.2-AC-07	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.2-SAC-01	Summary of Test Program and Objectives This SRP section lists the general criteria of RG 1.68 that a DC, COL, or OL applicant or holder should address in its safety analysis report (SAR). DC/COL/OL Applicants A. The ITP should describe its objectives, including a description of the objectives for each of the major phases of the test program. B. The ITP should describe the criteria for selection of plant features to be tested by the applicant. C. Objectives and testing selection criteria should be consistent with the general guidelines and applicable regulatory positions in **RG 1.68**. Applicants should appropriately justify exceptions.	Y	14.2.1 through 14.2.11
		N/A-COL	14.2.1 through 14.2.11
14.2-SAC-02	Test Program's Conformance with Regulatory Guides	Y	14.2.7

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	DC/COL/OL Applicants A. The applicant should commit to the revision of **RG 1.68** and the RGs listed in RG 1.68, that are referenced in this SRP and are in effect six months prior to submittal. The applicant may propose exceptions or alternatives to the specific criteria in any of these RGs, and the staff may find them acceptable if the applicant provides adequate justification. The reviewer responsible for the RG evaluates any exceptions or alternatives. The safety evaluation report (SER) should also list such exceptions or alternatives.	N/A-COL	14.2.7
14.2-SAC-03	Initial Test Program Administrative Procedures DC Applicant The applicant should provide a summary description of the following areas: A. The applicant should provide general guidance to control ITP activities, including administrative controls that will be used to develop, review, and approve individual test procedures, coordination with organizations involved in the test program, participation of plant operating and technical staff, and review, evaluation, and approval of test results. B. The applicant should include general guidance for the review of relevant operating and testing experiences at other facilities. This guidance should recognize reportable occurrences of repeatedly experienced safety concerns and other operating experiences that could potentially impact the performance of the test program. C. The applicant should include general guidance about how, and to what extent, the test program will use and/or test plant operating,	Y	14.2.1 through 14.2.11 (Programmatic) 14.2.12.1 (Test abstracts)
		N/A-COL	14.2.1 through 14.2.11 (Programmatic) 14.2.12.1 (Test abstracts)

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CHAPTER 14 Initial Test Program and ITAAC-Design Certification			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	emergency, and surveillance procedures. D. The applicant should provide test abstracts of SSCs and unique design features that will be tested to verify that system and component performance is in accordance with the design. These test abstracts should include the objectives, tests, and acceptance criteria that will be included in the test procedures. COL/OL Applicants The applicant should provide a detailed description of the following areas: A. Management Organizations i. The applicant should provide organizational descriptions for principal management positions responsible for the planning, execution, and documentation of preoperational and startup testing activities. ii. The applicant should provide (1) the organizational descriptions for any augmenting organizations or other personnel who will manage or execute any phase of the test program, and (2) the responsibilities, interfaces, and authorities of the principal participants. B. Conduct of the Initial Test Program i. The applicant should conduct the ITP using detailed procedures approved by designated managers in the applicant's organization. ii. Administrative controls should be established to ensure that the designated construction-related inspections and tests are completed before preoperational testing begins. The applicant should also include in the ITP adequate controls for the		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 14 Initial Test Program and ITAAC-Design Certification			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	evaluation and approval of preoperational test results before initial startup tests begin. iii. Administrative controls should address adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures. iv. The controls that the applicant uses to ensure that the test prerequisites are met should include requirements for (1) inspections, checks, and similar controls, (2) identification of test personnel completing data forms or checksheets, and (3) identification of dates of completion. Each major phase of the test program as well as individual tests should satisfy these requirements. v. The staff will find that the controls provided for plant modification and repairs, identified as a result of plant testing, are acceptable if the controls (1) are sufficient to ensure that the required repairs or modifications will be made, (2) will ensure retesting is conducted following such modifications or repairs, and (3) will ensure a review of any proposed facility modifications by the original design organization or other designated design organizations. The applicant's requirements for documentation associated with such controls should permit audits to be conducted to ensure its proper implementation. C. Test Program Schedule and Sequence i. The applicant should develop a schedule for conducting each major phase of the ITP. ii. The schedule should establish that the safety of the plant will not		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 14 **Initial Test Program and ITAAC-Design Certification**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	depend on the performance of untested SSCs. iii. Overlapping test program schedules (for multiunit sites) should not result in significant divisions of responsibilities or dilutions of the staff implementing the test program. iv. The sequential schedule for individual startup tests should establish that test requirements will be completed in accordance with plant technical specification requirements for SSC operability before changing plant modes. D. Staff Responsibilities, Authorities, and Qualifications i. The applicant should describe the education, training, and experience requirements established for each management and operating staff member—including the NSSS vendor, architect-engineer, and other major contractors, subcontractors, and vendors, as appropriate—who will conduct the preoperational and startup tests and will develop testing, operating, and emergency procedures. ii. The applicant should develop a training program for each functional group of employees in the organization relative to the schedule for preoperational testing and initial startup testing to ensure that the necessary plant staff are ready to begin the test program. E. Development, Review, and Approval of Test Procedures i. The applicant is responsible for the preparation of preoperational and startup test procedures. This includes the methodology used for the generation, review, and approval of test procedures. ii. The applicant should use the NSSS vendor, architect-engineer,		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 14 Initial Test Program and ITAAC-Design Certification			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and other major contractors, as appropriate, to provide the test objectives and acceptance criteria used in developing detailed test procedures. iii. The applicant's administrative system for use in reviewing and approving individual test procedures should provide for appropriate levels of review before approval. iv. Controls should be in place to ensure that test procedures include appropriate prerequisites, test objectives, safety precautions, testing of initial conditions, methods to direct and control test performance, and acceptance criteria for evaluating the test. v. The applicant should include provisions to ensure that retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments. vi. The format for the test procedures should be similar to that in **RG 1.68**, or the reviewer should consider whether the justification provided by the applicant for exception is acceptable. The format should include checklists and signature blocks to control the sequencing of testing. vii. Approved test procedures should be in a form suitable for review by regulatory inspectors at least 60 days before their intended use. Licensees should provide timely notification to NRC of changes in approved test procedures that have been made available for NRC review. F. Review, Evaluation, and Approval of Test Results i. The applicant should develop the procedures that will govern the		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	review, evaluation, and approval of test results for each phase of the test program. Specific procedures should be implemented to ensure notification of responsible organizations, such as design organizations, when test acceptance criteria are not met and specific controls have been established to resolve such problems. ii. Before proceeding with testing, the applicant should provide controls relating to (1) the methods and schedules for approval of test data for each major phase, and (2) the methods used for initial review of individual parts of multiple tests (e.g., hot functional testing). iii. The controls that will govern the review, evaluation, and approval of test results should provide a technical evaluation of test results by qualified personnel and approval of such results by personnel in designated management positions in the applicant's organization. iv. The applicant should include provisions to allow design organizations to participate in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria. v. Provisions should be in place to retain test reports, including test procedures and results, as part of the plant historical records. Startup test reports should be prepared in accordance with **RG 1.16**, or the reviewer should consider whether the justification provided by the applicant for exception is acceptable. G. Utilization of Reactor Operating and Testing Experiences in the		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 14 Initial Test Program and ITAAC-Design Certification			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Development of the Test Program i. The applicant should provide a summary of the principal conclusions or findings from the review of operating and testing experiences at other reactor facilities and their effect on the test program. This review should recognize categories of reportable, repeatedly experienced occurrences and other operating experiences that could potentially impact the performance of the test program. H. Trial Use of Plant Operating and Emergency Procedures i. The applicant should incorporate, to the extent practicable, the plant operating, emergency, and surveillance procedures into the test program or otherwise verify these procedures through use during the test program. ii. The applicant should provide additional operator training and participation based on the performance and evaluation of the test results of certain initial tests. An acceptable program will satisfy the criteria described in **Three Mile Island (TMI) Action Plan Item I.G.1 of NUREG-0660** and **NUREG-0737**.		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
14.2-SAC-04	Initial Startup Tests <u>DC Applicants</u> The applicant should provide a summary description of the following areas: A. Initial Fuel Loading/Initial Criticality/Low-Power/Power Ascension Testing i. The applicant should include in the ITP a description of the general provisions and precautions for fuel loading, initial fuel loading, initial criticality, low-power testing, and power ascension phases. Precautions, prerequisites, and measures should be consistent with the guidelines and regulatory positions in **RG 1.68**. This includes guidance for (1) the completion of all ITAAC associated with preoperational tests before fuel load, (2) measures to review and evaluate the results of the completed preoperational tests, (3) appropriate remedial actions to take if acceptance criteria are not satisfied, (4) applicable technical specification requirements, and (5) actions to take if unanticipated errors or malfunctions occur. <u>COL/OL Applicants</u> The applicant should provide a detailed description of the following areas:	Y	14.2.1.1.3 14.2.1.1.4 14.2.1.1.5
	A. Initial Fuel Loading and Initial Criticality	Y	14.2.10
	i. The applicant should provide measures to ensure that preoperational tests are evaluated and approved before fuel loading begins. ii. The procedures that will guide initial fuel loading and initial criticality should include precautions, prerequisites, and	N/A-COL	14.2.10

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	measures consistent with the guidelines and regulatory positions in **RG 1.68**. The staff will review exceptions to regulatory positions and their associated justification on a case-by-case basis. iii. Technical specifications should be instituted to ensure the operability of systems required for fuel loading. iv. The applicant should describe the minimum conditions for initial core loading, which may include, but are not limited to: (1) The reactor containment structure should be complete, and containment integrity should be demonstrated according to technical specifications. (2) Fuel handling tools and equipment should be available, and operators should be familiar with the use and operation of equipment. (3) The reactor vessel and associated components should be ready to receive fuel. (4) Nuclear instrumentation should be tested and verified to be operable. v. The applicant should include provisions to verify that core flux levels are within predicted or acceptable values. vi. The applicant should provide measures to stop core loading operations if an unexpected or unanalyzed condition occurs. vii. At the completion of fuel loading, the applicant should perform sufficient tests, as necessary, to ensure that the facility is in a final state of readiness to achieve initial criticality and to perform low-power tests.		

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B. Low-Power/Power Ascension Testing i. The applicant should include procedures that will control low-power and power ascension testing. These procedures should include precautions, prerequisites, and measures consistent with the guidelines and regulatory positions in **RG 1.68**. The staff will review exceptions to regulatory positions and their associated justifications for acceptability on a case-by-case basis.	Y	14.2.12.14 through 14.2.12.21
		N/A-COL	14.2.12.14 through 14.2.12.21
14.2-SAC-05	Individual Test Descriptions/Abstracts <u>DC/COL/OL Applicants</u> A. The applicant should provide abstracts of planned tests to demonstrate and verify the performance capabilities of SSCs and design features that serve the following functions: i. Used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintenance of the reactor in a safe condition for an extended shutdown period ii. Used for safe shutdown and cooldown of the reactor under transient conditions (infrequently or moderately frequent events) and postulated accident conditions and for maintenance of the reactor in a safe condition for an extended shutdown period following such condition iii. Used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications iv. Classified as engineered safety features or used to support or ensure the operations of engineered safety features within design limits	Y	14.2.8.1 14.2.12
		N/A-COL	14.2.8.1 14.2.12

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<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>v. Assumed to function, or for which credit is taken, in the accident analysis for the facility, as described in the DCD or SAR (as applicable)
vi. Used to process, store, control, measure, or limit the release of radioactive materials
vii. Used in a special low-power testing program to be conducted at power levels no greater than 5 percent for the purpose of providing meaningful technical information beyond that obtained in the normal startup test program, as required for the resolution of TMI Action Item I.G.1
viii. Identified as risk significant in the design-specific probabilistic risk assessment
B. The abstracts should include test objectives, prerequisites, test methods, significant parameters and plant performance characteristics to be monitored, and acceptance criteria in sufficient detail to establish the functional adequacy of the SSCs and design features tested.
C. For new, unique, or first-of-a-kind design features used in the facility, the functional testing requirements and acceptance criteria necessary to verify their performance should be submitted for review and approval.
D. If the testing method will not subject the SSC to representative design operating conditions, the test abstract should contain sufficient information to justify the proposed test method.</td><td></td><td></td></tr>
</table>

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 14 Initial Test Program and ITAAC-Design Certification			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
14.2-SAC-06	Initial Test Program Acceptance Criteria DC Applicants		
	A. The applicant should provide in Tier 1 a general description of the preoperational and power ascension test programs and the major program documents that define how the ITP will be conducted and controlled (i.e., a site-specific startup administrative manual, test specifications, and test procedures). Tier 2, Chapter 14.2, should contain a complete description of the ITP.	ITAAC	Tier 1
	B. The applicant should describe the key elements of the ITP in Tier 1 to ensure that the COL applicant cannot unilaterally initiate subsequent changes in the conduct of the ITP.	ITAAC	Tier 1
	C. The applicant should include provisions to ensure that test procedures and test specifications are made available to the NRC. COL/OL Applicants	Y	14.2.11
	A. Applicants referencing a certified design should provide a clearly and sufficiently described ITP in terms of scope and level of detail in accordance with the rule certifying the design and the design control document	N/A-COL	N/A
	B. An applicant which does not reference a certified design should provide a clearly and sufficiently described ITP in terms of scope and level of detail in accordance with **RG 1.68**.	N/A-COL	N/A
	C. Refer to **SRP Section 14.3.10** for additional guidance.	N/A-COL	N/A
SRP 14.2.1	**Generic Guidelines for Extended Power Uprate Testing Programs (08/2006)**	N/A-COL	N/A

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 14.3	**Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification (03/2007)**		
14.3-AC-01	**10 CFR 52.17(b)(3)**, requires that an ESP application proposing complete and integrated emergency plans contain ITAAC and that an ESP application proposing major features of the emergency plans may contain ITAAC.	N/A-ESP	N/A
14.3-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and should operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3-AC-03	**10 CFR 52.47(a)(26)**, which requires that a DC application contain justification that compliance with the interface requirements of paragraph (a)(25) of this section is verifiable through inspections, tests, or analyses. The method to be used for verification of interface requirements should be included as part of the proposed ITAAC required by paragraph (b)(1) of this section.	Y	14.3 Tier 1, Section 4
14.3-AC-04	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee should perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses should will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
14.3-SAC-01	Acceptance on the scope of the ITAAC is based on the complete facility or for a DC application, limited to the SSCs covered by the DC.	Y ITAAC	14.3 Tier 1
14.3-SAC-02	Acceptance criteria on the sufficiency of the ITAAC for the areas of review are specified in SRP Section 14.3 subsections.	See SRP 14,3,2 through SRP 14.3.12	
SRP 14.3.2	**Structural and Systems Engineering – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.2-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.2-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.2-SAC-01	The reviewer should primarily utilize the NRC rules and regulations to review the top level commitments in Tier 1. Other sources of review guidelines include RGs, SRP guidelines, and PRA insights from the standard design safety and severe accident analyses and operating	Y	14.3 Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	experience. If applicable, the staff also must adhere to policy decisions by the Commission. Examples of these are contained in the SRM related to **SECY-90-016**, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance in the SRM related to **SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The SRM related to SECY-93-087 is dated July 21, 1993.		
14.3.2-SAC-02	Design descriptions, figures (including key dimensions) and ITAAC should be developed and grouped by systems and building structures. For building structures, the structural capability is typically verified by performing an analysis to reconcile the as-built data with the structural design bases for each safety-related building. System-specific performance tests are typically conducted to demonstrate that the system can perform its intended function. For major components, the verification of design, fabrication, testing, and performance requirements should be partially addressed in conjunction with the specific system ITAAC. The review checklists for fluid systems, electrical systems, and building structures in Appendix C of SRP Section 14.3 should be used as aids for establishing consistency and completeness for the Tier 1 information.	Y ITAAC	14.3 Tier 1
14.3.2-SAC-03	<u>Review of the Standard Design Structural Integrity</u>. The scope of structural design covers the major structural systems in the standard design plant, including the RPV, ASME Code Class 1, 2, and 3 piping systems, and major building structures (primary containment, reactor building, control building, turbine building, service building, and radwaste building). For PWRs, this includes the reactor vessel (RV), ASME Code Class 1, 2, and 3 piping systems, and major building	Y	14.3 Tier 1, Section 2.01

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	structures (primary containment, nuclear island structures, turbine building, component cooling water (CCW) heat exchanger structures, diesel fuel storage structures (DFSSs), and radwaste building). The RPV, piping systems, and primary containment (For PWRS, RV, piping systems, and primary containment) are included because they provide the defense-in-depth principle for nuclear plants. The major building structures house those systems and components that are important to safety. In establishing the top level requirements for structural design, the staff used the General Design Criteria (GDC) of 10 CFR Part 50, Appendix A, as its basis. The primary general design criteria pertaining to the major structural system design are GDC 1, "Quality Standards and Records," GDC 2, "Design Bases for the Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Basis," GDC 14, "Reactor Coolant Pressure Boundary," GDC 16, "Containment Design," and GDC 50, "Containment Design Basis." **GDC 1** requires, in part, the need for structures, systems and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. **GDC 2** requires, in part, the need to design structures, systems, and components important to safety to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods without loss of capability to perform their safety functions, including the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena. **GDC 4** requires, in part, the need to protect structures, systems, and		

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<table>
<tr><th colspan="4">CHAPTER 14
Initial Test Program and ITAAC-Design Certification</th></tr>
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<tr><td></td><td>components important to safety from dynamic effects including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.
GDC 14 requires, in part, the need for the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
GDC 16 requires, in part, the need for the reactor containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment.
GDC 50 requires, in part, the need for the reactor containment structure including access openings and penetrations to be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.
Using the above GDC as its basis, the following top-level attributes should be verified by ITAAC:
A. pressure boundary integrity (GDC 14, 16 and 50)
B. normal loads (GDC 2)
C. seismic loads (GDC 2)
D. suppression pool hydrodynamic loads (GDC 4)
E. flood, wind, and tornado (GDC 2)
F. rain and snow (GDC 2)
G. pipe rupture (GDC 4)</td><td></td><td></td></tr>
</table>

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	H. codes and standards (**GDC 1**) I. **10 CFR 50, Appendix J** (**GDC 16**)		
	In addition, to ensure that the final as-built plant conforms to the certified design, applicants should provide ITAAC to reconcile the as-built plant with the structural design basis. A summary of the top-level structural design requirements for the major structural systems that are verified by the structures and systems in Tier 1 and the piping design information in Tier 1.	Y ITAAC	14.3 Tier 1
14.3.2-SAC-04	Pressure Boundary Integrity. To ensure that the applicable requirements of **GDC 14, 16, and 50** have been adequately addressed, ITAAC should be established to verify the pressure boundary integrity of the RPV, piping, and primary containment (For PWRs, RV, piping, and primary containment) for the standard design. **GDC 16**, **GDC 50**, and **10 CFR 50, Appendix J** apply to the primary containment and **GDC 14** applies to the RPV (RV for PWRs) and the reactor coolant pressure boundary piping systems. The pressure integrity for these major structural systems are needed to ensure the defense-in-depth principle. For the RPV and piping, hydrostatic tests performed in conjunction with the ASME Boiler and Pressure Vessel Code, Section III, should be required by ITAAC. See the standard ITAAC for hydrostatic tests in Appendix D to SRP Section 14.3. For the primary containment, a structural integrity test and containment integrated leakage rate test should be required by ITAAC to be performed on the pressure boundary components of the primary containment in accordance with the **ASME Boiler and Pressure Vessel Code, Section III**, and **10 CFR 50,**	ITAAC	Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Appendix J. Because the requirements of **GDC 14, 16, and 50** do not apply to the reactor, control, turbine, service, and radwaste buildings (nuclear island structures, turbine building, CCW heat exchanger structures, DFSSs, and radwaste building for PWRs), ITAAC are not required to verify the pressure integrity for these other buildings.		
14.3.2-SAC-05	<u>Normal Loads</u>. To ensure that the applicable requirements of **GDC 2** have been adequately addressed, ITAAC should be established to verify that the normal and accident loads have been appropriately combined with the effects of natural phenomena.	ITAAC	Tier 1
	For piping systems, ITAAC should require an analysis to reconcile the as-built piping design with the design-basis loads (which include the appropriate combination of normal and accident loads). See **SRP Section 14.3.3** for additional information. For the RPV, the fabrication may be performed primarily in the vendor's shop where adherence to design drawings is tightly controlled. Therefore, ITAAC for the as-built reconciliation of normal loads with accident loads for the RPV are inappropriate. Instead, ITAAC should verify that the ASME Code-required reports exist to document that the RPV has been designed, fabricated, inspected, and tested to Code requirements to ensure adequate safety margin. Similarly, for safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the combination of normal and accident loads with the effects of natural phenomena). The analysis results should be documented in a structural analysis report, the scope and contents of	Y ITAAC	14.3 Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	which must be described in Tier 2. The staff may determine that the design of certain structures does not require verification by ITAAC, based on their safety significance. In particular, these ITAAC should apply only to safety-related structures and are not applicable to the service and turbine buildings (radwaste and turbine building for PWRs). However, ITAAC for other design aspects of structures may be appropriate.		
14.3.2-SAC-06	<u>Seismic Loads</u>. To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC are established to verify that the safety-related systems and structures have been designed to seismic loadings. Component qualification for seismic loads should be addressed by ITAAC for verifying the basic configuration of systems. See the standard ITAAC for basic configuration in Appendix D to SRP Section 14.3 for additional information, and the discussion in **SRP Section 14.3.3**. As discussed above for normal loads on piping systems and the RPV, ITAAC should require an analysis to reconcile the as-built piping design with the design basis loads (which include seismic loads). See also the discussion in **SRP Section 14.3.3**. For the RPV, ITAAC for the as-built reconciliation of seismic loads for the RPV are deemed to be inappropriate as previously discussed. Instead, ITAAC verify that the ASME Code required reports exist for the RPV ensuring that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements. For safety-related buildings, ITAAC require an analysis for reconciling the as-built plant with the structural design-basis loads (which include seismic loads). The analysis results are to be documented in a structural analysis report, as discussed above. These ITAAC apply only to safety-	Y ITAAC	14.3 Tier 1

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	related structures and are not applicable to the service and turbine buildings (radwaste and turbine building for PWRs). However, because the leakage path for fission products includes components within the turbine building, the turbine building should be able to withstand the effects of a safe-shutdown earthquake, if not, ITAAC should be established to verify that, under seismic loads, the collapse of the turbine building will not impair the safety-related functions of any safety-related SSCs located adjacent to or within the turbine building.		
	For non-seismic Category I SSCs, the need for ITAAC to verify that their failure will not impair the ability of near-by safety-related SSCs to perform their safety-related functions should be assessed based on the specific design. If the design detail and as-built and as-procured information for many non-safety-related systems (e.g., field-run piping and balance-of-plant systems) is not provided by the applicant for design certification and the spatial relationship between such systems and seismic Category I SSCs cannot be established until after the as-built design information is available, the non-seismic to seismic (II/I) interaction cannot be evaluated until the plant has been constructed. Accordingly, the design criteria for ensuring acceptable II/I interactions and a commitment for the COL applicant to describe the process for completion of the design of balance-of-plant and non-safety related systems to minimize II/I interactions and proposed procedures for an inspection of the as-built plant for II/I interactions should be specified as a COL action item in Tier 2.	Y	Chapter 3
14.3.2-SAC-07	Suppression Pool Hydrodynamic Loads (BWRs only). To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC should be established to verify that the safety-related systems and structures have been designed to withstand	N/A-BWR	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	suppression pool hydrodynamic loadings, which include safety relief valve discharge and loss-of-coolant accident (LOCA) loadings. Component qualification for suppression pool hydrodynamic loads may be addressed by ITAAC established for verifying the basic configuration of systems.		
	As discussed above for seismic loads on piping systems and the RPV, ITAAC should require an analysis to reconcile the as-built piping design with the design- basis loads (which include suppression pool hydrodynamic loads). For the RPV, ITAAC should verify that the ASME Code-required reports exist to ensure that the RPV has been designed, fabricated, inspected, and tested to ASME Code requirements. For the reactor building and primary containment including the internal structures, ITAAC should require an analysis for reconciling the building as-built configuration with the structural design basis loads (which include suppression pool hydrodynamic loads). The as-built analysis results should be documented in a structural analysis report as discussed above. This report may be able to be satisfied using the ASME Code required reports for the reconciliation analysis for the primary containment. The effects of suppression pool hydrodynamic loads do not extend beyond the reactor building, and, thus, ITAAC are not required to verify these loadings for the building structures outside the reactor building.	Y ITAAC	14.3 Tier 1
	ITAAC also should require the verification of the horizontal vent system, water volume, and the safety-relief valve discharge line quencher arrangement to ensure adequacy of the suppression pool hydrodynamic loads used for design.	N/A-BWR	N/A

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14.3.2-SAC-08	Flood, Wind, Tornado, Rain, and Snow. To ensure that the applicable requirements of GDC 2 have been adequately addressed, ITAAC should be established to verify that the safety-related systems and structures have been designed to withstand the effects of natural phenomena other than those associated with seismic loadings. The effects include those associated with flood, wind, tornado, rain, and snow.	Y	Tier 1, Section 2.01
	These loadings do not apply to the RPV, the ASME Code Class 1, 2, and 3 piping systems and components, nor the primary containment (except for the exposed portions of the concrete containments) because they are all housed within the safety-related buildings. For safety-related buildings, ITAAC should require an analysis for reconciling the as-built plant with the structural design basis loads (which include the flood, wind, tornado, rain, and snow loads). Based on their safety significance, these ITAAC need apply only to safety-related structures and need not be applicable to the service and turbine buildings (radwaste and turbine building for PWRs).	Y ITAAC	14.3 Tier 1
	For flooding, site parameters are specified that require the maximum flood level and ground water level be below the finished plant grade level. ITAACs also require inspections to verify that divisional flood barriers and water-tight doors exist, and penetrations (except for water-tight doors) in the divisional walls are sealed up to the internal and external flood levels. In addition, for safety-related buildings, flood barriers are established up to the finished plant grade level to protect against water seepage, and flood doors and flood barrier penetrations are provided with flood protection features.	Y	Tier 1, Section 2.01

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	ITAAC should also require inspections to verify that water-tight doors exist, penetrations (except for water-tight doors) in the divisional walls are at least 2.5 m above the floor, and safety-related electrical, instrumentation, and control equipment are located at least 20 cm above the floor surface. In addition, for safety-related buildings, ITAAC should require that external walls below flood level are equal to or greater than 0.6 m to protect against water seepage, and penetrations in the external walls below flood level are provided with flood protection features.	Y	3.4.1
14.3.2-SAC-09	Pipe Break. To ensure that the applicable requirements of GDC 4 have been adequately addressed, ITAAC should be established to verify that the safety-related SSCs have been designed to the dynamic effects of pipe breaks. Component qualification for the dynamic effects of pipe breaks should be addressed by ITAAC established for verifying the basic configuration of systems.	N/A-COL	3.6.1 3.6.2
	For the RPV, ITAAC that verify the basic configuration of the RPV system require an inspection of the critical locations that establish the bounding loads in the LOCA analyses for the RPV to ensure that the as-built areas not exceed the postulated break areas assumed in the LOCA analyses.	ITAAC	Tier 1
	In addition, ITAAC should be established to verify by inspections of as-built, high-energy pipe break mitigation features and of the pipe break analysis report that safety-related SSCs be protected against the dynamic and environmental effects associated with postulated high-energy pipe breaks. ITAAC to verify pipe break loads are not required for the turbine, service, and radwaste buildings (turbine and radwaste buildings for PWRs) either because they are not safety-related structures	N/A-COL	3.6.1 3.6.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	or there are no high-energy lines located within the structure.		
14.3.2-SAC-10	Codes and Standards. To ensure that the applicable requirements of **GDC 1** have been adequately addressed, ITAAC should be established to verify that appropriate codes and standards are used in the design and construction of safety-related systems and components. In general, the staff considers those codes and standards endorsed by the regulations under **10 CFR 50.55a** in determining which codes and standards were appropriate for Tier 1 verification. The ASME Boiler and Pressure Vessel Code, Section III for Code Class 1, 2, and 3 systems and components is established as the code for the design and construction of standard design piping systems and the RPV. For safety-related building designs, the staff should base its safety findings on audits of standard design calculations which relied on specific codes and standards. These codes and standards are contained in the appropriate sections of DCD Tier 2 Chapter 3. Inspections will be conducted as a part of ITAAC to verify that ASME Code-required documents exist that demonstrate that the RPV, piping systems and containment pressure boundaries have been designed and constructed to their appropriate Code requirements. For other ASME Code components and equipment, the verification of Code compliance will be performed in conjunction with the quality assurance programs and by the authorized inspection agency as required by the ASME Boiler and Pressure Vessel Code. This DCD Tier 2 material should be considered for designation as Tier 2* information. Tier 2* information is information that, if considered for a change by an applicant or licensee that references the certified standard design, would require NRC approval	Y ITAAC	14.3 Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	prior to implementation of the change. Tier 2* material is discussed further in SRP Section 14.3.		
14.3.2-SAC-11	As-built Reconciliation. As discussed in various sections above, to ensure that the final as-built plant structures are built in accordance with the certified design as required by **10 CFR Part 52**, structural analyses should be performed which reconcile the as-built configuration of the plant structures with the structural design bases of the certified design. The structural analyses should be documented in structural analysis reports. Structural analysis reports should be verified in conjunction with ITAAC for the primary containment and the reactor, control, radwaste, and turbine buildings (nuclear island structures, radwaste building, CCW heat exchangers, DFSSs, and turbine building for PWRs). The detailed supporting information on what is required for an acceptable analysis report should be contained in DCD Tier 2 Chapter 3. Similarly for piping systems, an as-built analysis should be performed using the as-designed and as-built information. ITAAC should verify the existence of acceptable final as-built piping stress reports that conclude the as-built piping systems are adequately designed. See SRP Section 14.3.3 for additional information.	Y ITAAC	14.3 Tier 1
	For the RPV, the key dimensions of the RPV system should be verified in conjunction with the basic configuration check of the system. The key dimensions of the RPV system and the acceptable variations of the key dimensions should be provided in the certified design description. Alternatively, acceptable variations and the bases for them should be provided in Tier 2.	Y	4.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	For component qualification, tests, analyses, or a combination of tests and analyses should be performed for seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) to demonstrate that the as-built equipment and associated anchorages are qualified to withstand design basis dynamic loads without loss of safety function. These test and analyses should be performed as a part of ITAAC to verify the basic configuration of the system in which the equipment is located. See **Section 14.3.3** for additional information.	ITAAC	Tier 1
SRP 14.3.3	**Piping Systems and Components – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.3-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.3-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
14.3.3-SAC-01	Generic Piping Design. DC applicants may provide less than the complete design information for piping design before DC because the design may depend upon as-built and as-procured information. Instead, applicants may provide the processes and design acceptance criteria (DAC) by which design details in this area would be developed and evaluated. Implementation of the processes is the responsibility of the COL applicant or licensee. The DAC are discussed further in to SRP Section 14.3, Appendix A.	ITAAC	Tier 1
	The reviewer should use the SRP guidelines to evaluate the piping design information in Tiers 1 and 2 and audit the piping design criteria in detail, including sample calculations. The staff should evaluate the adequacy of the structural integrity and functional capability of safety-related piping systems. The review is not limited to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Classes 1, 2, and 3 piping and supports, but includes buried piping, instrumentation lines, the interaction of non-seismic Category I piping with seismic Category I piping, and any safety-related piping designed to industry standards other than the ASME Code. The staff's evaluation should include the analysis methods, design procedures, acceptance criteria, and related ITAAC (and DAC if applicable) that are to be used for the completion and verification of the standard design piping design. The staff's evaluation should include both DCD Tier 1 and Tier 2 information on the applicable codes and standards, analysis methods to be used for completing the piping design, modeling techniques, pipe stress analyses criteria, pipe support design criteria, high-energy line break criteria, and leak-before-break (LBB) approach applicable to the standard design.	Y	3.12 14.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Design descriptions and the associated DAC should be specified in Tier 1. The scope of the standard design to which the piping design information applies should be stated in Tier 1. This may be done on a generic basis using a single ITAAC applicable to multiple systems of the design, or applied to individual system ITAAC. If done using a generic piping design ITAAC, the Tier 1 should address its application to piping systems classified as both nuclear safety-related and non-nuclear safety systems. The nuclear safety-related piping systems must remain functional during and following a safe-shutdown earthquake (SSE), and should be designated in Tier 1 as seismic Category I and further classified as ASME Code Class 1, 2, or 3 in the individual systems of the standard design. Tier 1 should ensure that the piping systems will be designed to perform their safety-related functions under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events. The material in Tier 1 should also address the consequential effects of pipe ruptures such as jet impingement, potential missile generation, and pressure and temperature effects. The scope of the piping to be verified by the generic Piping ITAAC includes all ASME Code Class 1, 2, or 3 piping systems and high-energy piping systems. Tier 1 includes ASME Code Class piping systems because the ASME Boiler and Pressure Vessel Code, Section III is referenced in **10 CFR 50.55a**. Nuclear power plant components classified as Quality Groups A, B, and C are required by 10 CFR 50.55a to meet the requirements for ASME Code Class 1, 2, or 3, respectively. In each system description, the functional drawing identifies the boundaries of the ASME Code classification for the piping systems. The piping pressure	ITAAC N/A-COL	Tier 1 3.6.1 3.6.2

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	boundary and structural integrity are required to be maintained because they are directly involved in preventing or mitigating an accident or event under the defense-in-depth principle.		
	An acceptable approach to Tier 1 information for piping design is to specify distinct ITAAC that ensure the design process for piping systems occurs as described in the design description. For example, the first ITAAC specified in Tier 1 should require that an ASME Code certified stress report exists to ensure that the ASME Code Class 1, 2, or 3 piping systems and components are designed to retain their pressure integrity and functional capability under internal design and operating pressures and design basis loads. The specific contents and requirements of the certified stress report are contained in the ASME Code. The particular certified stress report to be used to satisfy the ITAAC should be specified in Tier 2. An acceptable version of an ASME Code certified stress report is the design document required by **ASME Code, Section III, Subarticle NCA-3550**. A certified piping stress report provides assurance that requirements of the ASME Code, Section III for design, fabrication, installation, examination, and testing have been met and that the design complies with the design specifications. A second ITAAC should require that a pipe break analysis report exists that documents that SSCs that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features. The design description should discuss the criteria used to postulate pipe breaks, the analytical methods used to perform pipe breaks, and the method to confirm the adequacy of the results of the pipe break analyses. The design description should be verified in a Pipe Break Analysis Report that provides assurance that the high-energy line break	Y ITAAC	14.3 Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	analyses have been completed. For postulated pipe breaks, the report confirms whether (A) piping stresses in the containment penetration area are within allowable stress limits, (B) pipe whip restraints and jet shield designs can mitigate pipe break loads, (C) loads on safety-related SSCs are within design load limits, and (4) SSCs are protected or qualified to withstand the environmental effects of postulated failures. The Pipe Break Analysis Report shall conclude that, for each postulated piping failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC should be contained in DCD Tier 2 Chapter 3. If the design uses Leak-Before-Break (LBB) methods, a third ITAAC should require that a LBB evaluation report exists which documents that LBB acceptance criteria are complied with for the as-built piping and piping materials. Bounding limits should be specified in Tier 2 using preliminary piping analysis results to establish a window of acceptable piping stress values for selected piping materials. The ITAAC verifies that these values are complied with using actual material properties and final piping configurations, and reconciles the as-built piping data with the LBB assumptions. Detailed information that supports this ITAAC should be contained in DCD Tier 2 Chapter 3.		
	A fourth ITAAC should require that an as-built piping stress report exists that documents the results of an as-built reconciliation analysis confirming that the final piping system has been built in accordance with the ASME Code certified stress report. The report provides an overall verification by inspection that the as-constructed piping system, including supports, are consistent with the certified design commitments. Specific attributes to be inspected should be described in the DCD Tier 2.	Y ITAAC	14.3 Tier 1
		N/A-COL	3.6.1 3.6.2

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	Although similar to the first ITAAC, this verification also provides assurance that the as-built documentation used for construction has been reconciled with the documentation used for design analysis and with the certified stress report discussed above. The inspection will also involve a review of the as-built, high-energy pipe break mitigation features (e.g., pipe whip restraints and jet impingement shields) to ensure that the installed features are consistent with the pipe break analysis report. The methodology and specific attributes to be inspected are described in the DCD Tier 2. Alternatively, if an NRC-approved LBB report exists, then the dynamic effects from those postulated high-energy pipe breaks could be excluded. The documentation for this as-built reconciliation review may become part of the certified stress report.		
	Selected material in DCD Tier 2 Chapter 3 provides design information and defines design processes that are acceptable for use in meeting the piping DAC in Tier 1. However, Tier 2 information may be changed by a COL applicant 1 or licensee referencing the certified design in accordance with a "50.59-like" process specified in the rule certifying the design. The staff's evaluation of the standard design for piping systems is based on the design processes and acceptance criteria material in the DAC and Tier 2. Consequently, the staff should consider designating selected aspects of these piping design processes as Tier 2* information. Tier 2* information is Tier 2 information that, if considered for a change by a COL applicant or licensee, requires NRC approval prior to implementation of the change. Consideration should also be given to allowing the designation of Tier 2* to expire at the first full power when the detailed design is complete and performance characteristics of the facility are known. Although applicants for design certification should	N/A-INFO	N/A

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	propose designating similar Tier 2* information to that in the DCDs for the evolutionary designs, the NRC bears the final responsibility for designating which material is Tier 2*. The basis for the use of Tier 2* should be discussed in the staff's safety evaluation report. The Tier 2* information is discussed further in Appendix A to SRP Section 14.3.		
	<u>Regulations, Codes and Standards</u>. The use of codes and standards in the certified design material (CDM) should be minimized with exceptions granted case by case. Instead, the applicable requirements from the regulations, codes, or standards should be stated in the CDM, rather than reference them. This ensures that the requirement is clear, and allows flexibility if the reference changes. References to various parts of **ASME Sections III and XI** may verify issues like pressure boundaries or pre-service inspection requirements. Also, references to **10 CFR Part 20** may be required for radiation protection. The specific code edition, volume, version, date, etc., should be specified in the site safety analysis report rather than Tier 1. This provides for specific requirements that are acceptable, yet allows the code to be updated via the change process in the rule certifying the design. It is important to note that, due to the provisions of **10 CFR 52.63** and the rule certifying the design, changes to the codes and standards in **10 CFR 50.55a** would not necessarily be requirements for the certified design.	ITAAC	Tier 1
14.3.3-SAC-02	<u>Verifications of Components and Systems</u>. In addition to the generic approach to piping design in Tier 1, the verification of piping and component classification, fabrication, dynamic and seismic qualification, and selected testing and performance		

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	requirements is also addressed by specific ITAAC in the individual Tier 1 systems.		
	A. Piping and Component Safety Classification. **10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1**, requires that safety-related SSCs be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed. Nuclear power plant components classified as Quality Groups A, B, and C are required by **10 CFR 50.55a** to meet the requirements for ASME Code Class 1, 2, or 3, respectively; therefore, SSC safety classifications should be in each system's design description, and the functional drawings should identify the ASME Code classification boundaries applicable to the safety class. The ASME Code classes in ASME Code, Section III, allow a choice of rules that provide assurance of structural integrity and quality commensurate with the relative importance assigned to the individual items of the nuclear power plant. The ASME Boiler and Pressure Vessel Code class requirements may be verified by either a generic piping design ITAAC or by each system ITAAC. The use of other codes and standards (e.g., American Institute of Steel Construction manual for building structural steel) is within the Tier 2 scope, and the DCD Tier 2 describes the applicable codes and standards for these other safety-related SSCs not designed to the ASME Boiler and Pressure Vessel Code, Section III.	Y	3.2.2 14.3
	B. <u>Fabrication (Welding)</u>. **10 CFR Part 50, Appendix A, GDC 14**, requires that the reactor coolant pressure boundary be designed, fabricated, erected, and	Y	3.2.2 14.3

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	tested to have an extremely low probability of abnormal leakage. In addition, GDC 30 requires that component parts of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. The ASME Code class welds are included in Tier 1 because the ASME Boiler and Pressure Vessel Code, Section III is referenced in **10 CFR 50.55a**, which requires nuclear power plant components classified as Quality Groups A, B, and C to meet ASME Code Class 1, 2, or 3 requirements, respectively. In each system description, the functional drawing shows the boundaries of the ASME Code classification. The integrity of the pressure boundary is required to be maintained because it is directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. ASME Code Class 1, 2, or 3 structural welds (e.g., pipe support welds) are not within Tier 1 scope because they indirectly prevent or mitigate accidents or events (e.g., pipe supports protect the piping but the piping itself is needed for accident mitigation). Thus, ASME Code Class 1, 2, or 3 structural welds are in the Tier 2 scope.		
	The integrity of the pressure boundary in the plant will be ensured, in part, through a verification of the welding quality. This verification is performed as a part of the basic configuration ITAAC of each specific system. The basic configuration ITAAC, one of the standard ITAAC listed in SRP Section 14.3, Appendix D, is required for most systems in Tier 1. The provisions of the basic configuration check that must be specified in Tier 1 include non-destructive examination of the as-built pressure boundary welds for the ASME Code Class 1, 2, or 3 SSCs in the design description.	ITAAC	Tier 1

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	The acceptance criteria for the welds are the **ASME Code, Section III** weld examination requirements. The specific weld examination requirements for a particular ASME Code Class 1, 2, or 3 component and weld type are tabulated in Tier 2. The specific weld examination requirements are considered Tier 2 because they could change depending on future revisions to the ASME Code, Section III requirements.		
	Other welding activities (non-ASME Code) include: i. Pressure-boundary welds other than ASME Code, Section III welds, ii. Structural and building steel welds, iii. Electrical cable tray and conduit support welds, iv. Heating, ventilation, and air-conditioning support welds, and v. Refueling cavity and spent fuel pool liner welds. These other types of welding are included in the Tier 2 scope. Tier 2 describes the applicable codes and standards for the other types of welding and the weld acceptance criteria. Similar to the ASME Code Classes 1, 2, and 3 structural welds, these other welds are needed for protection of safety-related SSCs but do not directly (or are redundant) prevent accidents or events. Accordingly, these other types of welding were deemed inappropriate for Tier 1 scope.	Y	3.9.4 3.12 14.3
	C. Hydrostatic Test. The integrity of the pressure boundary is required to be maintained because it is directly involved in preventing or mitigating an accident or event under the defense-in-depth principle. The pressure boundary integrity is also ensured, in part, through a hydrostatic test verifying	ITAAC	Tier 1

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	the leak-tightness of the ASME Code piping systems. A hydrostatic test is generally specified by the **ASME Code, Section III**, for ASME Code Class 1, 2, and 3 SSCs to verify whether pressure integrity is maintained in the process of fabricating the overall piping system, including any welding and bolting requirements. However, the ASME piping stress report in the generic piping ITAAC does not include the results of hydrostatic tests; therefore, the standard hydrostatic test ITAAC in SRP Section 14.3, Appendix D, should be specified in each system ITAAC with ASME Code Class 1, 2, or 3 SSCs. The hydrostatic test ITAAC also may be specified in other appropriate Tier 1 systems.		
	D. Equipment Seismic and Dynamic Qualification. The basic configuration ITAAC listed in SRP Section 14.3, Appendix D, include verifications of the dynamic qualification (e.g., seismic, loss-of-coolant accident, and safety relief valve discharge loads) of seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) in the design descriptions and figures. This inspection verifies the capability of mechanical and electrical equipment in as-built condition, including anchorages, to perform safety functions during and following a SSE. Detailed supporting information for dynamic qualification requirements, including seismic qualification records, is in DCD Tier 2, Chapter 3. The Tier 2 information describing dynamic qualification of equipment should be considered for designation as Tier 2*. Tier 2* information is addressed further in SRP Section 14.3, Appendix A.	Y ITAAC	14.3 Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	E. <u>MOVs and Other Valves</u>. The verification of the design qualification of valves is performed in conjunction with the basic configuration check for mechanical equipment as discussed above. For MOVs in particular, a special inspection is part of the basic configuration check to verify the records of vendor tests that demonstrate MOV ability to function under design conditions. The list of MOVs in Tier 1 should include, but not be limited to, those with active safety-related functions. These may be listed in Tier 2 in the inservice testing plan or other locations. The DCD Tier 2, Chapter 3 material should have detailed supporting information for the CDM for the methods of the COL applicant or licensee for the design, qualification, and testing of MOVs to demonstrate their design-basis capability. This material should be considered for designation as Tier 2* information. Tier 2* information is addressed further in SRP Section 14.3, Appendix A. In-situ testing of installed MOVs, POVs, and check valves, to verify whether they can perform intended functions under various fluid flow, differential pressure, electrical, and temperature conditions, should be conducted as appropriate in the applicable system ITAAC. Standard ITAAC are provided in Appendix D to SRP Section 14.3 for verification of the performance of these valves. These may be performed as part of the pre-operational test program. Tier 2 information should be provided that defines that these tests will be conducted under maximum achievable pre-operational conditions and describes the analyses that will be performed to show how the test results demonstrate that the valves will function under design basis conditions (See Tier 2 **Section 3.9.6**). For significant operating	Y ITAAC	14.3 Tier 1

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	problems with other types of valves, or with pumps in general, the proper operation of these components may be implicitly tested, if applicable, as part of other functional tests in the system ITAAC. They also may be tested in the pre-operational or power ascension test program.		
SRP 14.3.4	**Reactor Systems – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.4-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.4-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.4-SAC-01	Appendix A of SRP 14.3 describes and provides guidance relative to the content of the DCD for a design certification application and defines Tier 1 and Tier 2 design-related information that is to be ultimately incorporated by reference into the design certification rules. The basis for	Y ITAAC	14.3 Tier 1

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	identifying Tier 1 information as derived from Tier 2 information, which is essentially the same information as is required for a design certification application, is that the top-level design features and performance standards (Tier 1) are those that are most important to safety, including safety-related and defense-in-depth features and functions, and non-safety-related systems that potentially impact safety. Tier 1 should be reviewed to verify that plant safety analyses, such as for core cooling, transients, overpressure protection, steam generator tube rupture, and anticipated transients without scram (ATWS), are adequately addressed. Applicants should provide tables in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. For intersystem LOCAs, the design pressure of the piping of the systems that interface with the reactor coolant pressure boundary should be specified in the design descriptions or figures.		
	The specific fuel, control rod, and core designs presented in Tier 2 constitute an approved design that may be used for the COL first-cycle core loading without further NRC staff review. If any other core design is requested for the first cycle, the COL applicant or licensee will be required to submit for staff review those specific fuel, control rod, and core design analyses as described in DCD Tier 2 Chapters 4, 6, and 15. Much of the detailed supporting information in Tier 2 for the nuclear fuel, fuel channel, and control rods, if considered for a change by a COL applicant or licensee that references the certified standard design, would require prior NRC approval. Therefore, for the evolutionary and passive designs, the staff concluded that this information should be designated as Tier 2* information (see Appendix A of SRP Section 14.3 for a definition).	N/A-INFO	N/A

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	However, staff will allow some of the Tier 2* designations to expire after the first full-power operation of the facility when the detailed design has been completed and the core performance characteristics are known from the startup and power-ascension test programs. The NRC bears the final responsibility for designating which material in Tier 2 is Tier 2*.		
	The following issues are identified to ensure comprehensive and consistent treatment of Tier 1 based on the safety significance of the system being reviewed: a. System purpose and functions b. Location/functional arrangement of system c. Key design features of the system d. System operation in various modes e. Seismic and ASME code classifications f. Materials—weld quality and pressure-boundary integrity g. Controls, alarms, and displays h. Logic i. Interlocks j. Class 1E electrical power sources and divisions k. Equipment to be qualified for harsh environments l. Valve qualification and operation m. Interface requirements with other systems n. Numeric performance values (flow rates, capacities, etc.) o. Accuracy and quality of figures p. Active systems that provide defense-in-depth functions designated as	ITAAC	Tier 1

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	non-safety systems Appendix C to SRP 14.3 provides “checklists” for the fluid systems as an aid for establishing consistency and comprehensiveness in the review of the system.		
14.3.4-SAC-02	The source of information used to determine safety significance of SSCs for the design of reactor and core cooling systems include applicable rules and regulations, general design criteria, unresolved safety issues, and generic safety issues, NRC generic correspondence, PRA, insights from the standard design’s safety and severe accident analyses, and operating experience. Inputs from the PRA review, including shutdown safety evaluations, and severe accident analyses ensure important insights and design features from these analyses are incorporated into Tier 1. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.	Y ITAAC	14.3 Tier 1
14.3.4-SAC-03	The passive-designed reactors use safety systems that employ passive means (natural forces), such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident mitigation. These designs also include active systems that provide defense-in-depth capabilities for reactor-coolant makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. SECY-95-132, “Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)”	N/A-PAS	N/A

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	provides certain guidance and positions for ensuring consistent and complete treatment of those systems that might be classified as non-safety-related by the designer or applicant but are important to safety or otherwise provide defense-in-depth functions.		
14.3.4-SAC-04	Applicable regulatory guidance from the Commission for selected policy and technical issues related to particular design should be followed. For the severe accident analyses, the basis for the staff's review for the evolutionary and passive standard designs was the Commission guidance related to **SECY-90-016**, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements.” **SECY-93-087**, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs” generically presents guidance and NRC positions on evolutionary and passive LWR design certification issues. For guidance, positions, and issues related to specific designs, guidance is available in such documents as **SECY-97-044**, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design” or **SECY-92-137**, “Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) Requirements for the General Electric (GE) Advanced Boiling Water Reactor (ABWR).” Regarding DAC, **SECY-02-059**, “Use of Design Acceptance Criteria for the AP1000 Standard Plant Design,” presents staff conclusions on acceptable use of DAC for I&C, control room, and piping design areas, contingent upon Westinghouse’s and the staff’s agreeing on adequate DAC during the design certification review. In **SECY-92-053**, “Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Process,” the staff noted that DAC is defined as “a set of prescribed limits, parameters, procedures, and	Y	Throughout FSAR

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	attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification.”		
	In some instances, an applicant may employ DAC to provide the staff with information to support its safety determination process. In **SECY-92-053**, the staff noted that “the concept of DAC would enable the staff to make a final safety determination, subject only to satisfactory design implementation and verification by the COL licensee through appropriate use of ITAAC.” The staff defined DAC as “a set of prescribed limits, parameters, procedures, and attributes upon which the NRC relies, in a limited number of technical areas, in making a final safety determination to support a design certification. The DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design. That is, the acceptance criteria for DAC become the acceptance criteria for ITAAC, which are part of the design certification.” The use of DAC by applicants use for I&C is considered acceptable given the rapidly changing technology for digital I&C systems. For many of the design features, it might be impractical to test their functionality because of the absence of simulated severe accident conditions. An example might be the ability of the reactor cavity to absorb the heat and radiation effects of a molten core. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown and confirmatory test reports or analysis, may be considered sufficient Tier 1 treatment. Another example in which passive designs would be difficult to verify prior to fuel loading as related to normal operations involves natural circulation.	Y ITAAC	14.3 Tier 1

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	Passive designs, compared to previous designs, can include elongated-reactor-core designs to create the pressure differential for establishing natural circulation. Evidence of prior testing and analysis providing conclusive results may have to suffice for suitable acceptance criteria for ITAAC purposes.		
14.3.4-SAC-05	Appendix D of SRP 14.3 lists acceptable “Standard ITAAC Entries” in the standard three-column format for ITAAC entries for configuration of systems, hydrostatic tests, net positive suction head for pumps, divisional power supply, etc., that should be contained in the overall set of ITAAC entries, as appropriate.	ITAAC	Tier 1
	RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition),” contains guidance for developing ITAAC assuming that a COL applicant does not reference a certified design and/or an early site permit. Guidance in Section III for COLs referencing a certified design notes that the ITAAC contained in the certified design must apply to those portions of the facility design that have been approved. Appendix C.II.2-A provides “general ITAAC development guidance” on fluid, I&C, and electrical systems.	N/A-COL	N/A
SRP 14.3.5	**Instrumentation and Controls – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.5-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	Y ITAAC	14.3 Tier 1

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	the Atomic Energy Act, and the NRC's regulations;		
14.3.5-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.5-SAC-01	The methodology for selecting SSCs that will be subject to ITAAC as well as the criteria for establishing the necessary and sufficient ITAAC should be appropriate for and consistently applied to I&C systems.	Y	14.3
14.3.5-SAC-02	Tier 1 Design Descriptions (for DC and for COL referencing DC) and ITAAC Design Descriptions or ITAAC references to the FSAR (for COL not referencing DC) should describe the top-level I&C design features and performance characteristics that are significant to safety. For safety systems, this should include a description of system purpose, safety functions, equipment quality (e.g., meet the functional requirements of **IEEE Std. 603-1991** and the digital system life cycle design process), equipment qualification, automatic decision-making and trip logic functions, manual initiation functions, and design features (e.g., system architecture) provided to achieve high functional reliability. The functions and characteristics of other I&C systems important to safety should also be discussed to the extent that the functions and characteristics are necessary to support remote shutdown, support required operator actions or assessment of plant conditions and safety	Y ITAAC	14.3 Tier 1, Section 2.4

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	system performance, maintain safety systems in a state that assures their availability during an accident, minimize or mitigate control system failures that would interfere with or cause unnecessary challenges to safety systems, or provide diverse back-up to protection systems.		
	SRP Section 14.3, Appendix A, Subsection B.1, provides additional guidance on the content of Tier 1 Design Descriptions, ITAAC Design Descriptions, or ITAAC references to the FSAR.	ITAAC	Tier 1
14.3.5-SAC-03	ITAAC should identify the significant features of the I&C systems on which the Staff is relying to assure compliance with each NRC requirement identified in **SRP Appendix 7.1-A**. Tests, analyses, and acceptance criteria associated with each design commitment should, when taken together, be sufficient to provide reasonable assurance that the final as-built I&C system fulfills NRC requirements. **SRP Appendix 7.1-C** provides an expanded discussion of SRP acceptance criteria for safety system compliance with 10 CFR 50.55a(h). **SRP Appendix 7.1-D** further discusses SRP acceptance criteria for safety and protection systems using digital computer-based technology. SRP Section 14.3, Appendix A, Subsection B.2, provides additional guidance on the expected scope, content, and format of ITAAC.	Y	Tier 1, Section 2.4
14.3.5-SAC-04	For DC or for COL applications referencing a DC, Tier 1 Design	Y	14.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Descriptions and ITAAC design commitments should be based on and consistent with the Tier 2 material. For a COL application not referencing a DC, the ITAAC Design Descriptions (if provided) and ITAAC design commitments should be based on and consistent with the FSAR portion of the application.	ITAAC	Tier 1
14.3.5-SAC-05	The applicant may provide design acceptance criteria (DAC) in lieu of detailed system design information. In this case, the DAC should be sufficiently detailed to provide an adequate basis for the Staff to make a final safety determination regarding the design, subject only to satisfactory design implementation and verification of the DAC by the COL applicant or licensee. Implementation of the DAC should be verified as part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design.	Y	Tier 1, Section 2.4
SRP 14.3.6	**Electrical Systems – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.6-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.6-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance	N/A-COL	N/A

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	that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
14.3.6-AC-03	**GDC 17**, in part, requires that an onsite and an offsite electric power systems be provided to permit functioning of SSCs important to safety. It further requires that the onsite electric power system have independence and redundancy and the electric power supplied by the offsite system be supplied by two physically independent circuits. Also, GDC 17 requires that provisions be included to minimize the likelihood of losing all electric power as a result of or coincident with, loss of power generated by the nuclear power unit, from the transmission network, or the onsite electric power supplies.	Y	8.1 14.3
		ITAAC	Tier 1
14.3.6-AC-04	**10 CFR 50.49** as it relates to EQ of electrical equipment important to safety for nuclear power plants. Applicants must ensure that safety-related, certain nonsafety-related, and certain post-accident monitoring equipment can perform their intended functions in various anticipated environments.	ITAAC	Tier 1
14.3.6-AC-05	**10 CFR 50.63** requires that a nuclear power plant be able to withstand and recover from a station blackout event.	ITAAC	Tier 1
14.3.6-SAC	The staff's review of the standard plant is conducted to ensure, in part, that Tier 1 contains top level design, fabrication, testing, and performance requirements for SSCs important to safety. Design Descriptions and ITAAC should be established to verify that these top level requirements (or design commitments) are met when the plant is built. IEEE nuclear standards should be used, as appropriate, to further establish top level	N/A-INFO	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	requirements. IEEE Std. 308, "IEEE Standard Criteria for Class 1E power Systems for Nuclear Power Generating Stations," in conjunction with other related IEEE standards, establish specific design criteria for nuclear power plant electrical systems and equipment. The standard design Class 1E electrical systems may include: (1) the Class 1E electrical power distribution system, (2) the emergency diesel generators (EDGs), (3) the Class 1E direct current power supply, and (4) the Class 1E vital ac and Class 1E instrument and control power supplies. Using the above regulations, IEEE standards, operating experience, and PRA as its bases, the applicant should establish top-level design commitments for the Class 1E electrical systems of the standard design to be included in the design descriptions and verified by ITAAC.		
	The top-level design commitments for the Class 1E electrical systems include design aspects related to:		
14.3.6-SAC-1E-01	Equipment qualification for seismic and harsh environment To ensure that the seismic design requirements of GDC 2 and the EQ requirements of **10 CFR 50.49** have been adequately addressed, a "basis configuration" standard ITAAC may be established for applicable systems to verify these design aspects of electrical equipment important to safety. The Design Description should identify that Class 1E equipment is seismic Category 1 and equipment located in a harsh environment is qualified. The basic configuration standard ITAAC may be used to verify these areas. EQ of safe-shutdown equipment may be verified as part of the basic	ITAAC	Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	configuration ITAAC for safety-related systems. EQ treatment in the ITAAC would then be discussed in the General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class IE electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.		
	Qualification of systems and components for seismic and harsh environments should be verified by ITAAC. Electrical equipment located in a "mild" environment should be discussed in the applicable sections of the COL application only. An exception is made for state-of-the-art digital instrumentation and control (I&C) equipment and digital control and protection systems located in an "other than harsh" environment. Operational experience has shown these state-of-the-art equipment and systems to be sensitive to temperature. ITAAC should be included to verify the qualification of equipment whose performance may be impacted by sensitivity to particular environmental conditions not considered by regulations to be harsh.	Y ITAAC	14.3 Tier 1
14.3.6-SAC-1E-02	Redundancy and independence To ensure that the Class 1E electric systems meet the single failure requirements of **GDC 17** (and other GDC), ITAAC may be established to verify the redundancy and independence of the Class 1E portion of the electrical design. For the electrical systems, ITAAC should verify the Class 1E divisional assignments and independence of electric power by both inspections and tests. The independence may be established by both electrical isolation and physical separation. Identification of the Class 1E divisional equipment should be included to aid in demonstrating	Y ITAAC	14.3 Tier 1

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	the separation. (The detailed requirements are specified in Tier 2. For example, separation distances and identification are outlined in Tier 2). These attributes should be verified all the way to the electrically powered loads by a combination of the electrical system ITAAC and the ITAAC of the individual fluid, I&C, and heating, ventilation and air conditioning (HVAC) systems which also cover the electrical independence and divisional power supply requirements. ITAAC should be included to verify adequate separation, required inter-ties (if any), required identification (e.g., color coding), proper routing/termination (i.e., location), separation of non-Class 1E loads from 1E buses. Post-fire safe shutdown separation of electrical circuits should be addressed in the fire protection system ITAAC.		
14.3.6-SAC-1E-03	Capacity and Capability To ensure that the electrical systems have the capacity and capability to supply the safety-related electrical loads, ITAAC should be established to verify the adequate sizing of the electrical system equipment and its ability to respond (e.g., automatically in the times needed to support the accident analyses) to postulated events. This includes the Class 1E portion and the non-Class 1E portion to the extent that it is involved in supporting the Class 1E system. ITAAC should be included to analyze the as-built electrical system and installed equipment (diesel generators, transformers, switchgear, batteries, etc.) to verify its ability to power the loads. In addition, the ITAAC should also include tests to demonstrate the operation of the equipment. Testing should be included in ITAAC to verify EDG capacity and capability based on the Technical Specifications. In some cases regulatory guidance specifies the need for margin in capacity to allow for	Y	Tier 1, Chapter 3

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	future load growth. If it is only for future load growth, ITAAC does not need to check for the additional margin. ITAAC should be developed to verify the initiation of the Class 1E equipment necessary to mitigate postulated events for which the equipment is credited (e.g., loss of coolant accident (LOCA), loss of offsite power (LOOP), and degraded voltage conditions). ITAAC should be included to analyze the as-built electrical power system for its response to a LOCA, LOOP, combinations of LOCA and LOOP (including LOCA with delayed LOOP and LOOP with delayed LOCA), and degraded voltage, including tests to demonstrate the actuation of the electrical equipment in response to postulated events.		
	Analyses to demonstrate the acceptability of a voltage drop should be included in ITAAC to verify adequacy for supporting the accomplishment of a direct safety function. The applicable section of the COL application should include a discussion of how the voltage analyses will be performed, i.e., reference to industry standards. Testing should be included in ITAAC to verify the EDG voltage and frequency response is acceptable and is the same as that specified in the Technical Specifications.	ITAAC	Tier 1
14.3.6-SAC-1E-04	Electrical protection features To ensure that the electrical power system is protected against potential electrical faults, ITAAC should be established to verify the adequacy of the electrical circuit protection included in the design. Operating experience and NRC Electrical Distribution System Functional Inspections (EDSFIs) have indicated some problems with the short circuit rating of some electrical equipment and breaker and protective device	ITAAC	Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	coordination. Inclusion in ITAAC should be based on the potential for preventing safety functions and the operating experience. ITAAC should be included to analyze the as-built electrical system equipment for its ability to withstand and clear electrical faults. ITAAC should also be included to analyze the protection feature coordination to verify its ability to limit the loss of equipment due to postulated faults. Equipment short circuit capability and breaker coordination should be verified by specifying ITAAC for analyses. The description of the analyses should be included in the applicable section of the application. Similarly, diesel generator protective trips (and bypasses if applicable) should be considered.		
14.3.6-SAC-1E-05	Displays/controls/alarms To help ensure that the electrical power system is available when required, ITAAC should be included to verify the existence of monitoring and controls for the electrical equipment. The minimum set of displays, alarms, and controls is based on the emergency procedure guidelines. In some cases, additional displays, alarms, and controls may be specified based on special considerations in the design and/or operating experience. ITAAC should be included to inspect for the ability to retrieve the information (displays and alarms), and to control the electrical power system in the main control room and/or at locations provided for remote shutdown. Detection of undervoltage conditions along with the starting and loading of EDG should be included in ITAAC. This is a direct safety function in response to design basis event of loss of power. Problems with relay settings should be considered in this requirement.	ITAAC	Tier 1

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	Other Electrical Equipment Important to Safety		
14.3.6-SAC-OTHER-01	Offsite Power To ensure that the requirements of **GDC 17** for the adequacy and independence of the preferred offsite power sources within the standard design scope were met, ITAAC should verify the capacity and capability of the offsite sources to feed the Class 1E divisions, and the independence of those sources. ITAAC should be included to inspect the direct connection of the offsite sources to the Class 1E divisions and to inspect for the independence/separation of the offsite sources. ITAAC should be developed to inspect for appropriate lightning protection and grounding features. In addition, the Design Description includes "interface" requirements for the portions of the offsite power outside of the standard design scope; however, no ITAAC are included for the interfaces. The interfaces define the requirements that the offsite portion of the design (that is out-of-scope) must meet to support and not degrade the in-scope design (See also Appendix A to SRP Section 14.3)	ITAAC	Tier 1, Chapter 4
14.3.6-SAC-OTHER-02	Containment Electrical Penetrations To ensure the containment electrical penetrations (both those containing Class 1E circuits and those containing Non Class 1E circuits) do not fail due to electrical faults and potentially breach the containment, ITAAC should verify that all electrical containment penetrations are protected against postulated currents greater than their continuous current rating.	ITAAC	Tier 1
14.3.6-SAC-OTHER-03	Alternate AC Power Source (if applicable)	ITAAC	Tier 1

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SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	To ensure the availability of the alternate AC (AAC) power source for station blackout events, ITAAC should be developed to verify, through inspection and testing, the AAC power source (combustion gas turbines, diesel generators, or hydro units) and its auxiliaries along with its independence from other AC sources.		
14.3.6-SAC-OTHER-04	Lighting ITAAC should be included to verify the continuity of power sources for plant lighting systems to ensure that portions of the plant lighting remain available during accident scenarios and power failures. The basis for inclusion may be more related to defense-in-depth, support function, operating experience, or PRA rather than "accomplishing a direct safety function.”	ITAAC	Tier 1
14.3.6-SAC-OTHER-05	Electrical Power For Non-Safety Plant Systems To ensure that electrical power is provided to support the non-safety plant systems, Design Descriptions cover portions of the non-Class 1E electrical systems. ITAAC should be included to verify the functional arrangement of electrical power systems provided to support non-safety plant systems to the extent that those systems perform a significant safety function.	ITAAC	Tier 1
SRP 14.3.7	**Plant Systems – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.7-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and	Y ITAAC	14.3 Tier 1

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	will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;		
14.3.7-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.7-SAC-01	The reviewer should utilize the SRP in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. Tier 1 should be reviewed for consistency with the initial test program described in DCD Tier 2 Chapter 14.2. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in his review of the systems. If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to particular design. Examples of these are contained in **SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." The SRM related to this is dated July 21, 1993.	Y ITAAC	14.3 Tier 1
14.3.7-SAC-02	Tier 1 should be reviewed for treatment of design information proportional	Y	Tier 1

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	to the safety significance of the SSC for that system. Many items may be judged to be important to safety, and thus should be included in Tier 1. The following issues are identified to ensure comprehensive and consistent treatment in Tier 1 based on the safety significance of the system being reviewed: (1) System purpose and functions (2) Location of system (3) Key design features of the system (4) Seismic and ASME code classifications (5) System operation in various modes (6) Controls, alarms, and displays (7) Logic (8) Interlocks (9) Class 1E electrical power sources and divisions (10) Equipment to be qualified for harsh environments (11) Interface requirements (12) Numeric performance values (13) Accuracy and quality of figures		
14.3.7-SAC-03	Standard ITAAC entries should be utilized to verify selected issues, where appropriate. The reviewer should ensure consistent application and treatment of the standard ITAAC entries for basic configuration ITAAC, net positive suction head, and physical separation for appropriate systems in Tier 1. In particular, the general provision for environmental qualification aspects of SSCs invoked by the basic configuration ITAAC should be reviewed to ensure appropriate treatment in Tier 1.	ITAAC	Tier 1

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14.3.7-SAC-04	Environmental qualification (EQ) of safe-shutdown equipment may be verified as part of the basic configuration ITAAC for safety-related systems. EQ treatment in the ITAAC would then be discussed in the General Provisions section of Tier 1. Verification may include type tests or a combination of type tests and analyses of Class 1E electrical equipment identified in the Design Description or accompanying figures to show that the equipment can withstand the conditions associated with a design basis accident without loss of safety function for the time that the function is needed.	ITAAC	Tier 1
14.3.7-SAC-05	The design features in Tier 1 should be selected to ensure that the integrity of the analyses are preserved in an as-built facility. For example, 3-hour fire boundaries and divisional separation may be shown in the building figures. Also, flooding features such as structure elevations should be specified in the site parameters, flood doors may be shown on the building figures, and elevations are shown on the buildings to verify that the approximate physical location of components and relative elevations of buildings minimize the effects of flooding. As-built reconciliation reports for fires and floods to ensure consistency with Tier 2 analyses should be required by the appropriate system ITAAC (e.g., fire protection system) and selected building ITAAC, respectively.	Y ITAAC	14.3 Tier 1
14.3.7-SAC-06	Other specific issues that should be addressed include heat removal capabilities for design-basis accidents and tornado and missile protection. Heat removal capabilities may be verified through heat removal requirements for core cooling system heat exchangers and interface requirements for site-specific systems. Tornado and missile protection may be provided by inlet and outlet dampers in ventilation	Y	Tier 1

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	systems, and through the structural design of buildings.		
14.3.7-SAC-07	The areas of review for radioactive waste systems include design objectives, design criteria, identification of all expected releases of radioactive effluents, methods of treatment, methods used in calculating effluent source terms and releases of radioactive materials in the environment, and operational programs in controlling and monitoring effluent releases and for assessing associated doses to members of the public. The radioactive waste systems include the liquid waste management system (LWMS), gaseous waste management system (GWMS), and the solid waste management system (SWMS). These systems deal with the management of radioactive wastes, as liquid, wet, and dry solids, produced during normal operation and anticipated operational occurrences. In addition, the review includes an evaluation of the process and effluent radiological monitoring instrumentation and sampling systems (PERMISS) which are used to monitor liquid and gaseous process streams and effluents and solid wastes generated by these systems. The PERMISS includes subsystems used to collect process and effluent samples during normal operation, anticipated operational occurrences, and under post-accident conditions. The lead branch responsible in implementing the review should coordinate the review of these systems and operational programs and receive input on the design and compliance with acceptance criteria listed in **SRP Sections 11.2 to 11.5** from other branches, including, balance of plant, structural, instrumentation and controls, HVAC, quality assurance, technical specifications, and emergency planning.	Y	11.2 11.3 11.3 11.5 14.3
14.3.7-SAC-08	The reviewer should receive inputs on the treatment of issues identified above from other branches such as the structural, electrical and I&C	Y	14.3

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	branches. In addition, the secondary review branches specified in SRP Section 14.3 should provide inputs on selected issues. These issues include key insights and assumptions from PRA and severe accident analyses, as well as inputs for issues such as treatment of alarms, displays and controls, and functionality of MOVs. Cross-references from Tier 2 to Tier 1 for key insights and assumptions from PRA and severe accidents should be provided by applicants in Tier 2 together with these analyses.	ITAAC	Tier 1
14.3.7-SAC-09	Tier 1 should address and verify at least the minimum inventory of alarms, controls, and indications as derived from the Emergency Procedure Guidelines, the requirements of **RG 1.97**, and probabilistic risk assessment insights. These may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and verified to exist. Other controls, indications and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system design and function. The ability of these controls, indications, and alarms to function should be checked during operation of the system for the functional tests required by the system ITAAC. Because the intent of the ITAAC is to verify the final as-built condition of the plant, the operation of the system during the completion of the functional tests required in the system ITAAC should be conducted from the MCR. Therefore, the verification that the system can be operated from the MCR need not be a separate ITAAC. Also, because the operation of the equipment from the control room demonstrates the control function, continuity checks between the RSS and the equipment demonstrates that the control signal will be received by the component	Y ITAAC	14.3 Tier 1

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	and provides adequate assurance that the equipment can be operated by the RSS. The results of the pre-operational test program may be utilized to demonstrate the ability to operate plant equipment by the RSS.		
SRP 14.3.8	**Radiation Protection – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.8-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.8-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.8-SAC-01	The reviewer should primarily use the applicable rules and regulations, general design criteria, regulatory guides, unresolved safety issues, and generic safety issues in the review of Tier 1 to determine the safety significance of SSCs with respect to the radiation protection for occupational workers and the general public they provide. Other sources include the SRP and applicable U.S. Nuclear Regulatory Commission	Y	14.3

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(NRC) generic correspondence. The reviewer should use the guidance in Appendix C to SRP Section 14.3 as an aid for ensuring the comprehensiveness and consistency of this review.		
14.3.8-SAC-02	Radiation Protection: The reviewer should ensure that Tier 1 identifies and describes, commensurate with their safety significance, those SSCs that provide radiation shielding, confinement or containment of radioactivity, ventilation of airborne contamination, or radiation (or radioactivity concentration) monitoring for normal operations and during accidents. Tier 1 identifies and describes the measures that need to be employed during first-of-a-kind engineering to ensure that final design details (i.e., materials and component selection, equipment placement, and pipe routing) are consistent with the radiation protection commitments (including the commitment that radiation exposures will be as low as is reasonably achievable (ALARA)) in the certified design. Tier 1 contains ITAAC that ensure that the identified SSCs will function in a manner consistent with the certified design.	Y ITAAC	14.3 Tier 1
14.3.8-SAC-03	Design Processes and Design Acceptance Criteria: A DC applicant may not provide sufficient detail in selected aspects of the design, including sufficient information to stipulate the source terms needed to verify the design of the shielding, ventilation, and airborne radioactivity monitoring systems. The applicant may choose to provide design processes and DAC for this material, as discussed in Appendix A to **SRP Section 14.3**. The applicant should document in DCD Tier 2, Section 14.3, its rationale for determining which areas of the design should use design processes	Y ITAAC	14.3 Tier 1

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and acceptance criteria. Essentially, the applicant should extract the most important design processes and acceptance criteria from DCD Chapter 12 of Tier 2 and identify them in Tier 1. This may be done either in a separate section of Tier 1 or in the applicable systems of Tier 1. A COL applicant or licensee must meet these criteria in the design of the plant, and the staff can audit the facility's design documentation to ensure that the criteria are met. The following discussion is specific to the review of design processes and acceptance criteria in this area. DC applicants may not provide the complete design information in this design area before the design is certified because the radiation shielding design and the calculated concentrations of airborne radioactive material depend on as-built and as-procured information about plant systems and components. Therefore, applicants may be unable to describe the standard design's radiation source terms (i.e., the quantity and concentration of radioactive materials contained in, or leaking from, plant systems) in sufficient detail to allow the staff to verify the adequacy of the shielding design, ventilation system designs, or the design and placement of the airborne radioactivity monitors. Instead, applicants may provide the processes and acceptance criteria by which the details of the design in this area are to be developed, designed, and evaluated. The design description should state the scope of the material in Tier 1. The application could, for example, encompass the radiological shielding and ventilation design of the reactor building, turbine building, control building, service building, and radwaste building. The COL applicant or licensee is responsible for the implementation of the process and the design. The DAC may be taken from the acceptance criteria in the applicable sections of Chapter 12 of the SRP. The analysis methods and source		

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	term assumptions specified in the DAC should be consistent with the approved methods and assumptions listed in the SRP. The SRP is the basis for the staff's safety review of the standard design. Therefore, demonstrating that the final design meets these DAC with the methods and assumptions specified in Tier 1 ensures that the as-built design will meet the applicable acceptance criteria of the SRP and the associated regulations and staff technical positions. The DAC in Tier 1 should address the verification of the plant radiation shielding design and the plant airborne concentrations of radioactive materials (e.g., the ventilation system and airborne monitoring system designs). The DAC should require the COL applicant to calculate radiation levels and airborne radioactivity levels within the plant rooms and areas to verify the adequacy of these design features during plant construction (concurrently with the verification of the ITAAC). The plant rooms and areas to which the DAC apply may be given in figures in Tier 1. The appropriate section of DCD Tier 2, Chapter 12, should include detailed supporting information for the DAC. The criteria in Tier 1 should ensure that the radiation shielding design (as provided by the plant structures or by permanent or temporary shielding included in the design) is adequate so that the maximum radiation levels in plant areas are commensurate with the areas' access requirements. This will allow radiation exposures to plant personnel to be maintained ALARA during normal plant operations and maintenance. Tier 1 should ensure that adequate shielding is provided for those plant areas that may require occupancy to permit an operator to aid in the mitigation of or the recovery from an accident. Tier 1 should ensure that the contribution of gamma shine to the radiation dose (particularly from the turbine building)		

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	to a member of the public (off site) will be a small fraction of the U.S. Environmental Protection Agency's dose limits in found at **40 CFR Part 190**. The criteria in Tier 1 should ensure that the plant provides adequate containment and ventilation flow rates to control the concentrations of airborne radioactivity to levels commensurate with the access requirements of areas in the plant. Tier 1 should ensure that once the concentrations of airborne radioactivity are determined, the required airborne monitors are placed in the appropriate locations in the plant.		
SRP 14.3.9	**Human Factors Engineering – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.9-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.9-AC-02	10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
14.3.9-SAC-01	**SRP Chapter 18** provides guidance for the NRC staff to use in determining whether an applicant has proposed an acceptable HFE design. The applicant's HFE program will be evaluated in accordance with the review criteria of SRP Chapter 18 and **NUREG-0711**, "Human Factors Engineering Program Review Model." As indicated in Chapter 18, the HFE program technical information for the DC or COL review may be based on a design and implementation process plan. Therefore, the DC or COL ITAAC may be based on a design and implementation process plan. For example, acceptance criteria for the task analysis program element may be stated as "a report exists and concludes that function-based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions . . ."	ITAAC	Tier 1, Section 3.4
14.3.9-SAC-02	If an implementation plan, rather than a completed HFE element, was accepted as part of the design certification process, then ITAAC should address the completion of the HFE program element.	ITAAC	Tier 1, Section 3.4
14.3.9-SAC-03	If an implementation plan was not reviewed and approved as part of the design certification, then the ITAAC should address both the development of the plan as well as item 2 above.	ITAAC	Tier 1, Section 3.4
14.3.9-SAC-04	The reviewer will verify that HFE-related ITAAC information is provided based on accepted HFE principles and program elements as discussed in **SRP Chapter 18** and incorporated into the plant's design.	ITAAC	Tier 1, Section 3.4
14.3.9-SAC-05	HFE-related ITAAC should primarily address verification of products (e.g., the control room, the human-system interfaces, etc.) or results reports from implementing the HFE program element implementation plan.	ITAAC	Tier 1, Section 3.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
14.3.9-SAC-06	Minimum Inventory of Displays, Alarms and Controls: Tier 1 includes a minimum inventory of displays, controls, and alarms that are necessary to carry out the vendor's emergency procedure guidelines (i.e., Owners' Groups Generic Technical Guidelines) and critical actions identified from the applicant's PRA and task analysis of operator actions. The reviewers evaluation of the minimum inventory will encompass a multi-disciplinary effort consisting of human factors, I&C, PRA, and plant, reactor, and electrical system engineering. The minimum inventory list has been implemented through the rule-making process for four certified designs (**10 CFR Part 52 Appendixes A, B, C, and D**). The criteria used to determine acceptability of the inventory includes assuring that: (1) the scope of these items in the Generic Technical Guidelines and PRA effort are adequately considered, (2) the task analysis is detailed and comprehensive, (3) **RG 1.97**, Revision 3, Category 1 variables or RG 1.97, Revision 4, Type A, B, and C variables for accident monitoring are included, and (4) important system displays and controls described in Tier 1 system design descriptions necessary for transient mitigation are included.	ITAAC	Tier 1, Section 3.4
SRP 14.3.10	**Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.10-AC-01	**10 CFR 52.17** and **10 CFR 52.18**, as they relate to emergency planning information submitted in an ESP application. 10 CFR 52.17(b)(3) provides the requirement for ITAAC in an ESP application that includes major features of the emergency plans or complete and integrated emergency plans in accordance with 10 CFR 52.17(b)(2).	N/A-ESP	N/A
14.3.10-AC-02	**10 CFR 52.47** and **10 CFR 52.48**, as they relate to emergency planning	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	information submitted in a standard design certification application. **10 CFR 52.47(b)(1)** provides the requirement for ITAAC in a design certification application.		
14.3.10-AC-03	**10 CFR 52.77**, **10 CFR 52.79**, **10 CFR 52.80**, **10 CFR 52.81**, and **10 CFR 52.83**, as they relate to emergency planning and preparedness associated with a COL application. 10 CFR 52.80(a) provides the requirement for ITAAC in a combined license.	N/A-COL	N/A
14.3.10-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.	Y ITAAC	Tier 2, Section 14.3 Tier 1
14.3.10-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.10-SAC-01	The reviewer should ensure that for a design certification where an applicant has chosen to address emergency response facilities that the information provided adequately discusses facilities for emergency response. These include a habitable technical support center (TSC) with	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	space, data retrieval capabilities and dedicated communications equipment, and an operational support center (OSC) with adequate communications, consistent with the applicable criteria in **Supplement 1 to NUREG-0737** and **NUREG-0696**.		
14.3.10-SAC-02	A generic set of acceptable emergency planning EP-ITAAC was developed through coordination efforts between the NRC and the Nuclear Energy Institute (NEI) and resulted in the development of generic EP-ITAAC that are provided in Table 14.3.10-1(Table C.II.2-B11 of **RG 1.206**). These EP-ITAAC were established on a generic basis; they are not associated with any particular site or design. As such, several of the generic EP-ITAAC require the COL applicant to provide more specific acceptance criteria that reflect the plant-specific design and site-specific emergency response plans and facilities. This generic set is applicable to ESP applications that include ITAAC information. The reviewer should consider this set of EP-ITAAC in the review of application-specific EP-ITAAC that is tailored to the specific reactor design and emergency planning program requirements for the proposed plant and site. A smaller set of EP-ITAAC is acceptable if the application contains information that fully addresses emergency preparedness requirements associated with any of the generic ITAAC contained in Table 14.3.10-1 which is not all-inclusive, or exclusive of other ITAAC an applicant may propose. Additional plant-specific EP-ITAAC (i.e., beyond those listed in Table 14.3.10-1) may be proposed, and they will be examined to determine their acceptability on an applicant-specific basis. Table 14.3.10-1 also includes ITAAC associated with emergency response facilities that are within the scope of the design certification. COL applications referencing a certified design must include these	N/A-COL	N/A

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	design certification ITAAC on emergency response facilities. EP-ITAAC are proposed by the COL applicant and, except for EP-ITAAC from the referenced design certification or ESP, are subject to NRC review and a hearing with respect to whether they satisfy the “necessary and sufficient” requirement of 10 CFR 52.80(a). The complete set of EP-ITAAC will be incorporated into the COL as a license condition to be satisfied prior to fuel load. A COL holder may request a change in one or more of the EP-ITAAC, except those provided in the referenced certified design, via the license amendment process applicable to 10 CFR Part 52.		
SRP 14.3.11	**Containment Systems and Severe Accidents – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.11-AC-01	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;	Y ITAAC	14.3 Tier 1
14.3.11-AC-02	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic	N/A-COL	N/A

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Energy Act, and the NRC's regulations.		
14.3.11-SAC-01	The reviewer should primarily utilize the SRP sections related to containment systems in its review of Tier 1 to determine the safety significance of SSCs. Other sources include applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from the standard design's safety and severe accident analyses, and operating experience. The reviewer should also use the review checklists provided in Appendix C to SRP Section 14.3 as an aid for establishing consistency and comprehensiveness in the review of the systems.	Y ITAAC	14.3 Tier 1
14.3.11-SAC-02	Tier 1 should be reviewed to verify that key parameters and insights from containment safety analyses, such as loss of coolant accident, main steamline break, main feedline break, subcompartment analyses, and suppression pool bypass are adequately addressed. Applicants should provide cross references in DCD Tier 2 Section 14.3 to show how the important input parameters used in the transient and accident analyses for the design are verified by the ITAAC. The reviewer should ensure that appropriate treatment of severe accident design features and containment design features are included in Tier 1. The supporting information regarding the detailed design and analyses should remain in Tier 2. For many of the design features, it may be impractical to test their functionality because of the absence of simulated severe accident conditions. Consequently, the existence of the feature on a figure, subject to a basic configuration walkdown, may be considered sufficient Tier 1 treatment. Applicants should provide cross references in the appropriate sections of Tier 2 to show how the important parameters from PRA, including shutdown risk, and severe accident analyses are verified by the	Y ITAAC	14.3 Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	ITAAC. For both PRA and severe accident analyses, although large uncertainties and unknowns may be associated with the event phenomena, design features important for severe accident prevention and mitigation resulting from these analyses should be selected for treatment in Tier 1.		
14.3.11-SAC-03	If applicable, the reviewer should utilize regulatory guidance from the Commission for selected policy and technical issues related to the particular design. Examples of these are contained in **SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." The SRM related to this is dated July 21, 1993.	Y ITAAC	14.3 Tier 1
14.3.11-SAC-04	Containment isolation may be addressed by a combination of the system ITAACs or in a single system ITAAC. The containment isolation valves (CIVs) should be specified in Tier 1, and are most clearly shown on the system figures. The verification of the design qualification of the motor operated CIVs may be verified by the basic configuration check in each system ITAAC. In addition, in-situ tests should be required for containment isolation motor operated valves (MOV) and check valves in each system ITAAC. The ITAAC should verify that the CIVs close on receipt of an isolation signal. Actual closure of the containment isolation valves may be checked using the manual isolation switches in the main control room (MCR). Other ITAAC may verify that a containment isolation signal is generated for each of the process variables that will cause a containment isolation; the intent is to preclude multiple cycling of the containment isolation valves during the testing.	ITAAC	Tier 1
14.3.11-SAC-05	Tier 1 should address and verify at least the minimum inventory of	ITAAC	Tier 1, Section 3.4

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SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	alarms, displays, and controls in Design Control Document (DCD) Tier 2 Chapter 18. These are derived from Generic Technical Guidelines (e.g., Emergency Procedure Guidelines, Emergency Response Guidelines), the guidance of **RG 1.97**, and severe accident and PRA insights. They may be specified in the MCR and the Remote Shutdown System (RSS) ITAAC, or addressed in the appropriate ITAAC, and are verified to exist. Other controls, displays, and alarms should be identified in the system ITAAC based on their safety significance. Locations for these should be shown on system figures if important to system design and function.		
SRP 14.3.12	**Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria (03/2007)**		
14.3.12-AC-01	**10 CFR 73.1** as it relates to the prescribed requirements for the establishment and maintenance of a physical protection system and to protect against the design basis threat of radiological sabotage.	ITAAC	Tier 1, Section 3.1
14.3.12-AC-02	**10 CFR 73.55** as it relates to the requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.	ITAAC	Tier 1, Section 3.1
14.3.12-AC-03	**10 CFR 73.70(f)** as it relates to the requirements specific to design for alarm annunciation records.	ITAAC	Tier 1, Section 3.1
14.3.12-AC-04	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of	Y ITAAC	14.3 Tier 1

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	the Atomic Energy Act, and the NRC's regulations;		
14.3.12-AC-05	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.	N/A-COL	N/A
14.3.12-SAC-01	Appendix A to this SRP section provides an acceptable set of generic PS-ITAAC that an applicant may use to develop application-specific PS-ITAAC, tailored to specific physical security hardware requirements. Appendix A is not all-inclusive, or exclusive of other PS-ITAAC an applicant may propose. Additional plant-specific PS-ITAAC (i.e., beyond those listed in Appendix A) may be proposed and will be examined to determine their acceptability on a case-by-case basis.	ITAAC	Tier 1, Section 3.1

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 15.0	**Introduction – Transient and Accident Analysis (R3, 03/2007)**	Refer to SRP 15.0.1 through SRP 15.9	
SRP 15.0.1	**Radiological Consequence Analyses Using Alternative Source Terms (07/2000)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.0.2	**Review of Transient and Accident Analysis Methods (01/2006)**		
15.0.2-AC	The acceptance criteria are based on meeting the requirements of the regulations in 10 CFR Part 50 that govern the evaluation models for the specific accident under consideration (e.g., 10 CFR 50.46 for a LOCA).	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
15.0.2-SAC-01	Documentation The submittal must identify the specific accident scenarios and plant configurations for which the codes will be used. The evaluation model documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. Consistent nomenclature must be used throughout the entire model documentation. Any referenced material must be readily available from a technical library. Copies of any referenced documents that are not readily obtainable from a technical library or the NRC Public Document Room, including proprietary reports, must be included with the documentation or provided upon request. The code documentation must be sufficiently detailed that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of **Appendix B to 10 CFR Part 50**, and the documentation requirement in **Appendix K to 10 CFR Part 50**. It is desirable that the documentation include the responses to requests for additional information, sorted according to the review issue so that it is easy to	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2

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CHAPTER 15 **Accident Analysis**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	follow the entire review history for a single issue. The reviewer can help obtain this goal by issuing RAI's organized by review issue. The documentation must include the following components:		
	A. An overview of the evaluation model which provides a clear roadmap describing all parts of the evaluation model, the relationships between them, and where they are located in the documentation.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
	B. A complete description of the accident scenario including plant initial conditions, the initiating event and all subsequent events and phases of the accident, and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
	C. A complete description of the code assessment comprising a description of each assessment test, why it was chosen, success criteria, diagrams of the test facility that show the location of instrumentation that is used in the assessment, a code model nodalization diagram, and all code options used in the calculation.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
	D. A determination of the code uncertainty for a sample plant accident calculation. (Appendix K models do not require a determination of the code uncertainty.)	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
	E. A theory manual that is a self-contained document and that describes (a) field equations, (b) closure relationships, (c) numerical solution techniques, (d) simplifications and approximations (including limitations) inherent in the chosen field equations and numerical	Y (Per AREVA Topical Report EMF-2100P)	15.0.2

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	methods, (e) pedigree or origin of closure relationships used in the code, and (f) limits of applicability for all models in the code.		
	F. A user manual that provides (a) detailed instructions about how the computer code is used, (b) a description of how to choose model input parameters and appropriate code options, (c) guidance about code limitations and options that should be avoided for particular accidents, components, or reactor types, and (d) if multiple computer codes are used, documented procedures for ensuring complete and accurate transfer of information between different elements of the evaluation model.	Y (Per AREVA Topical Report EMF-CC-097P)	15.0.2
	G. A quality assurance plan that describes the procedures and controls under which the code was developed and assessed, and the corrective action procedures that are followed when an error is discovered.	Y (Per AREVA Topical Report ANP-10266-A)	3.1 15.0.2 17.5
	It is not important that the documentation be provided in exactly the format stated above but the information in the review package must be clearly organized in a reasonable manner.	N/A-INFO	N/A
15.0.2-SAC-02	Evaluation Model Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. The degree of imprecision that is allowed in the models will ultimately be determined by the amount of uncertainty that can be tolerated in the calculation. Models that cause non-physical predictions to	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the extent that misinterpretation of the calculated results or trends in the results may occur, are not acceptable. For Appendix K LOCA analyses, emergency core cooling system (ECCS) evaluation models must meet the specific requirements contained in **Appendix K to 10 CFR Part 50**.		
15.0.2-SAC-03	Accident Scenario Identification Process The purpose of the accident scenario identification process is to identify and rank the reactor component and physical phenomena modeling requirements based on (a) their importance to the modeling of the scenario and (b) their impact on the figures of merit for the calculation. The accident scenario identification process must be a structured process. It must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario. The models that are present in the code and their degree of fidelity in predicting physical phenomena must be consistent with the results of this process. For example, if the accident scenario identification process determines that a certain physical phenomenon is important to the scenario under consideration, the code must have a relatively accurate model for that phenomenon and a detailed assessment of that model must be provided. Phenomena that have lower ranking may be represented by models with larger inherent uncertainty. The formality and complexity of this process should be commensurate with the complexity and importance of the event under consideration.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2
15.0.2-SAC-04	Code Assessment Assessments of all code models intended to be used in the evaluation model must be provided. All assessments must be performed with the frozen version of the evaluation model that has been submitted for	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	review. Assessments performed with other versions of the evaluation model should be justified on a case by case basis because even “small” changes to the evaluation model can have unintended consequences on calculation results that were thought to not be impacted by the changes. Separate effects testing must be performed to demonstrate the adequacy of the physical models to predict physical phenomena that were determined to be important by the accident scenario identification process. Separate effects testing must also be used to determine the uncertainty bounds of individual physical models. Integral effects testing must be performed to demonstrate that the interactions between different physical phenomena and reactor coolant system components and subsystems are identified and predicted correctly. Assessments against both separate effects tests and integral effects tests must be performed with the code. All models need to be assessed over the entire range of conditions encountered in the transient or accident scenario. Assessments must also compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models. Code options used in the assessment calculations must be the same as those used in plant accident calculations. A scaling analysis must be performed that identifies important non-dimensional parameters related to geometry and key phenomena. Scaling distortions and their impact on the code assessment must be identified and evaluated in the assessment. Calculations of actual plant transients or accidents can be considered, but only as confirmatory		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	supporting assessments for the evaluation model. This is because the data available from plant instrumentation is usually not detailed enough to support code assessment of specific models. Plant data can be used for code assessment if it can be demonstrated that the available instrumentation provides measurements of adequate resolution to assess the code. The assessment cases must compare code predictions to all important measured variables in order to show that good predictions of one test variable do not result from compensating errors. Assessments must include a description of all assessment cases, specific models that are being assessed in each case, and acceptance criteria used. Acceptance criteria must be supported by quantitative analysis whenever possible. ECCS evaluation models must include a specific assessment to meet the criteria in **Appendix K to 10 CFR Part 50**. Small-break ECCS evaluation models must also meet the assessment requirements of TMI Action Item II.K.3.30, where applicable.		
15.0.2-SAC-05	Uncertainty Analysis The uncertainty analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident sequence identification process. When the code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application. In some cases, bounding values are used for input parameters as	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.0.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	described in SRP sections or Regulatory Guides and are used for plant operating conditions such as accident initial conditions, set points, and boundary conditions.		
15.0.2-SAC-06	Quality Assurance Plan The code must be maintained under a quality assurance program that meets the requirements of **Appendix B to 10 CFR Part 50**.	Y (Per AREVA Topical Report ANP-10266-A)	3.1 15.0.2 17.5
SRP 15.0.3	**Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors (03/2007)**		
15.0.3-AC-01	**Section 50.34(a)(1) of 10 CFR Part 50**, “Contents of applications; technical information,” as it relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with fission product release.	Y	15.0.3
15.0.3-AC-02	**General Design Criterion (GDC) 19** of Appendix A to 10 CFR Part 50, “Control room,” as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.	Y	15.0.3
15.0.3-AC-03	**Section 100.21 of 10 CFR Part 100**, “Non-seismic siting criteria,” as it relates to the evaluation and analysis of the radiological consequences of postulated accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.	Y	15.0.3
15.0.3-AC-04	**Paragraph IV.E.8 of Appendix E, to 10 CFR Part 50**, “Emergency Planning and Preparedness for Production and Utilization Facilities,” as it relates to adequate provisions for an onsite technical support center (TSC) from which effective direction can be given and effective control can be exercised during an emergency.	Y	15.0.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
15.0.3-SAC-01	<u>Offsite Radiological Consequences of Postulated Design Basis Accidents</u>. The acceptance criteria are based on the requirements of **10 CFR 50.34(a)(1)** as related to mitigating the radiological consequences of an accident in accordance with 10 CFR 52.17(a)(1) [early site permits], **10 CFR 52.47(a)(1)** [standard design certifications] and 10 CFR 52.79(b) [combined licenses]. The plant design features intended to mitigate the radiological consequences of accidents, site atmospheric dispersion characteristics and the distances to the exclusion area boundary (EAB) and to the low population zone (LPZ) outer boundary are acceptable if the total calculated radiological consequences for the postulated fission product release fall within the following exposure acceptance criteria specified in **10 CFR 50.34(a)(1)(ii)(D)**:	Y	15.0.3
	A. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE), and	Y	15.0.3
	B. An individual located at any point on the outer boundary of the LPZ, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE.	Y	15.0.3
	For CP, OL, DC and COL reviews, the application is acceptable with regard to the radiological consequences of analyzed DBAs if the calculated TEDEs at the EAB and the LPZ outer boundary do not exceed the dose acceptance criteria listed in Table 1 below.	Y	15.0.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

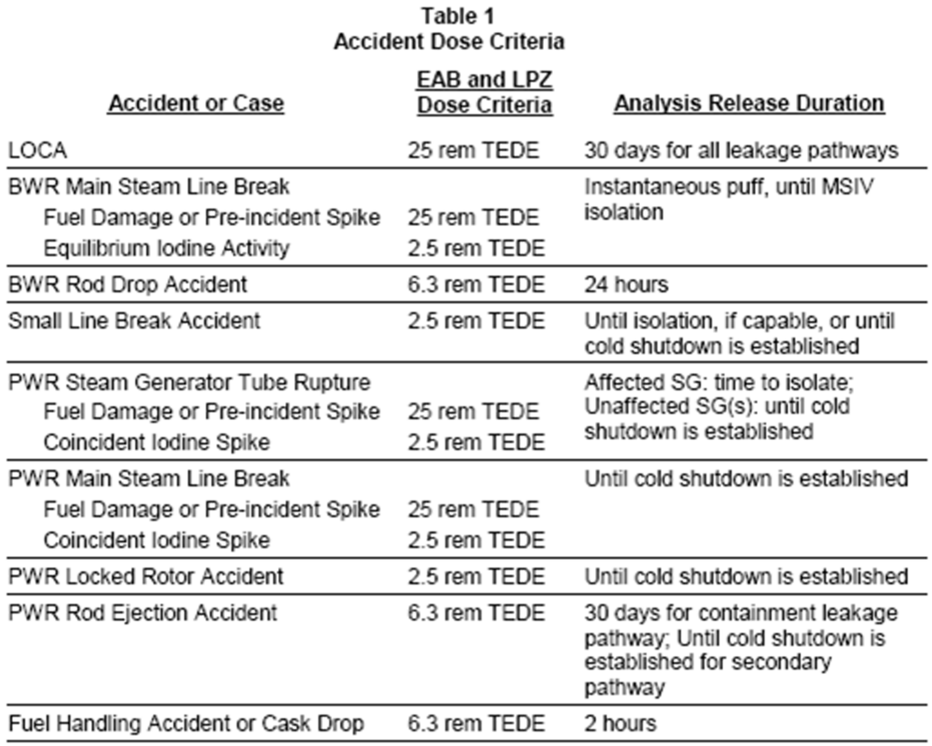
<table>
<tr><th colspan="4">CHAPTER 15
Accident Analysis</th></tr>
<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>
Table 1
Accident Dose Criteria
<table>
<tr><th>Accident or Case</th><th>EAB and LPZ Dose Criteria</th><th>Analysis Release Duration</th></tr>
<tr><td>LOCA</td><td>25 rem TEDE</td><td>30 days for all leakage pathways</td></tr>
<tr><td>BWR Main Steam Line Break</td><td></td><td>Instantaneous puff, until MSIV isolation</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Equilibrium Iodine Activity</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>BWR Rod Drop Accident</td><td>6.3 rem TEDE</td><td>24 hours</td></tr>
<tr><td>Small Line Break Accident</td><td>2.5 rem TEDE</td><td>Until isolation, if capable, or until cold shutdown is established</td></tr>
<tr><td>PWR Steam Generator Tube Rupture</td><td></td><td>Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Coincident Iodine Spike</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>PWR Main Steam Line Break</td><td></td><td>Until cold shutdown is established</td></tr>
<tr><td>Fuel Damage or Pre-incident Spike</td><td>25 rem TEDE</td><td></td></tr>
<tr><td>Coincident Iodine Spike</td><td>2.5 rem TEDE</td><td></td></tr>
<tr><td>PWR Locked Rotor Accident</td><td>2.5 rem TEDE</td><td>Until cold shutdown is established</td></tr>
<tr><td>PWR Rod Ejection Accident</td><td>6.3 rem TEDE</td><td>30 days for containment leakage pathway; Until cold shutdown is established for secondary pathway</td></tr>
<tr><td>Fuel Handling Accident or Cask Drop</td><td>6.3 rem TEDE</td><td>2 hours</td></tr>
</table></td><td></td><td></td></tr>
<tr><td></td><td>For ESP applications that neither reference the standard reactor designs certified by NRC nor use the PPE approach, the staff may establish dose acceptance criteria lower than those stated above for certain DBAs based on the probability of occurrence. Examples of such criteria are</td><td>N/A-ESP</td><td>N/A</td></tr>
</table>

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	illustrated in Table 1.		
	For COL applications using an ESP with a PPE approach, these acceptance criteria may be applied at that time. Such applicants bear the burden of ensuring sufficient margin is provided in the design parameters (for example, PPE values) in the ESP application to compensate for uncertainty in those parameters. The margin should be large enough such that the actual design submitted at the COL stage, coupled with the site characteristics as described in the ESP, will comply with NRC regulations.	N/A-COL	N/A
15.0.3-SAC-02	Control Room Radiological Habitability. The acceptance criterion is based on the requirements of **GDC 19** that mandate a control room design providing adequate radiation protection to permit access and occupancy of the control room under accident conditions for the duration of the accident, without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. These requirements are incorporated by reference in **10 CFR 52.47(a)(1)** [standard design certifications] and 10 CFR 52.79(b) [combined licenses]. The radiation protection design of the control room is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified in **GDC 19** of 5 rem TEDE for the duration of the accident.	Y	15.0.3
15.0.3-SAC-03	Technical Support Center Radiological Habitability. This acceptance criterion is based on the requirement of **Paragraph IV.E.8 of Appendix E to 10 CFR Part 50** to provide an onsite TSC from which effective direction can be given and effective control can be	Y	15.0.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room of 5 rem TEDE for the duration of the accident.		
SRP 15.1.1 - 15.1.4	**Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve (R2, 03/2007)**	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-AC-01	**General Design Criterion 10 (GDC 10)**, as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-AC-02	**General Design Criterion 13 (GDC 13)**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-AC-03	**General Design Criterion 15 (GDC 15),** as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-AC-04	**General Design Criterion 20 (GDC 20)**, as it relates the reactor protection system being designed to initiate automatically the operation of	Y	15.1.1 15.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences.		15.1.3 15.1.4
15.1.1-15.1.4-AC-05	**General Design Criterion 26 (GDC 26)**, as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences. This is accomplished by ensuring that appropriate margin for malfunctions such as stuck rods are accounted for.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC-Objectives	The basic objectives of the review of the transients which result from an increase in heat removal are:		
	1. Identify which of the moderate-frequency initiating events that result in increased heat removal are the most limiting.	Y	15.1.1 15.1.2 15.1.3 15.1.4
	2. Verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC-Specifc Criteria	The specific criteria necessary to meet the requirements of **General Design Criteria 10, 15, 20, and 26** for incidents of moderate frequency are:		
15.1.1-15.1.4-SAC- 01	Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.	Y	15.1.1 15.1.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
			15.1.3 15.1.4
15.1.1-15.1.4-SAC- 02	Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see **SRP Section 4.4**).	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC- 03	An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC- 04	To meet the requirements of **General Design Criteria 10, 13, 15, 20, and 26** the positions of **Regulatory Guide 1.105**, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC- 05	The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of **Appendix A to 10 CFR Part 50**, shall be identified and assumed in the analysis and shall satisfy the positions of **Regulatory Guide 1.53**.	Y	15.1.1 15.1.2 15.1.3 15.1.4
15.1.1-15.1.4-SAC-Tranients	**The analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model and NRC approved methodologies and the computer codes.** If the applicant proposes to use analytical methods which have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer performs an evaluation based on SRP	Y (Per AREVA Topical Reports ANP-10263 & ANP-10278)	15.1.1 15.1.2 15.1.3 15.1.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	section 15.0.2, "Transient and Accident Analysis methods."		
15.1.1-15.1.4-SAC-Parameters	The values of the parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:		
	1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event	Y	15.1.1 15.1.2 15.1.3 15.1.4
	2. Conservative scram characteristics are assumed, i.e., for a PWR - maximum time delay with the most reactive rod held out of the core, and for a BWR - a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR or DCD) Section 4.4.	Y	15.1.1 15.1.2 15.1.3 15.1.4
	3. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, doppler coefficient, axial power profile, and radial power distribution	Y	15.1.1 15.1.2 15.1.3 15.1.4
	4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with **Regulatory Guide 1.105**. Compliance with Regulatory Guide	Y	15.1.1 15.1.2 15.1.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	1.105 is determined by the Instrumentation and Control Systems		15.1.4
SRP 15.1.5	**Steam System Piping Failures Inside and Outside of Containment (PWR) (R3, 03/2007)**		
15.1.5-AC-01	**General Design Criterion 13 (GDC 13)**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.1.5
15.1.5-AC-02	**General Design Criterion 17 (GDC 17),** as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.	Y	15.1.5
15.1.5-AC-03	**General Design Criteria 27 (GDC 27) and 28 (GDC 28)**, as they relate to the reactor coolant system being designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.	Y	15.1.5
15.1.5-AC-04	**General Design Criterion 31 (GDC 31)**, as it relates to the reactor coolant system being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.	Y	15.1.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.1.5-AC-05	**General Design Criterion 35 (GDC 35)**, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.	Y	15.1.5
	Requirements for ensuring adequate decay heat removal and reactor coolant pump Integrity and operation are specified in **10 CFR 50.34(f)(2)(xii)** and **10 CFR 50.34(f)(1)(iii)**, respectively.	Y	15.1.5
15.1.5-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.	Y	15.1.5
15.1.5-SAC-02	The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see **SRP Section 4.4**). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see **SRP Section 4.2**), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.	Y	15.1.5
15.1.5-SAC-03	The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in **SRP section 15.0.3**.	Y	15.1.5
15.1.5-SAC-04	The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage.	Y	15.1.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.1.5-SAC-05	The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated.	Y	15.1.5
15.1.5-SAC-06	Tripping of the reactor coolant pumps should be consistent with the resolution to **Task Action Plan item II.K.3.5**.	Y	15.1.5
SRP 15.1.5.A	**Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR (R2, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.2.1 – 15.2.5	**Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed) (R2, 03/2007)**	(Refer to entries below)	15.2.1 15.2.2 15.2.3 15.2.4
		N/A-BWR	15.2.5
15.2.1-15.2.5-AC-01	**General Design Criterion (GDC) 10**, as to reactor coolant system design with appropriate margin so specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs).	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-AC-02	**GDC 13**, as to the availability of instrumentation to monitor variables an systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-AC-03	**GDC 15**, as to design of the reactor coolant system and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.	Y	15.2.1 15.2.2 15.2.3 15.2.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.2.1-15.2.5-AC-04	**GDC 17**, as to onsite and offsite electric power systems so safety-related structures, systems, and components (SSCs) function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-AC-05	**GDC 26**, as to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (*e.g.*, stuck rods).	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-SAC-01	The basic objectives of the review of the initiating events listed in subsection I of this SRP section:		
	A. Identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	B. Verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	C. Verify whether the plant protection systems' setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in **RG 1.105**.	Y	15.2.1 15.2.2 15.2.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
			15.2.4
	D. Verify whether the event evaluation considers single failures, operator errors, and performance of nonsafety-related systems consistent with the **RG 1.206** regulatory guidelines.	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-SAC-02	With the ANS standards as guidance, specific criteria meet the relevant requirements of **GDCs 10, 13, 15, 17, and 26** for events of moderate frequency.		
	A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	B. Fuel cladding integrity must be maintained by the minimum departure from nucleate boiling ratio (DNBR) remaining above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently	Y	15.2.1 15.2.2 15.2.3 15.2.4
	D. The requirements in **RG 1.105**, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs	Y	15.2.1 15.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	addressed in this SRP section.		15.2.3 15.2.4
	E. The most limiting plant system single failure, as defined in "Definitions and Explanations," **10 CFR Part 50, Appendix A**, must be assumed in the analysis according to the guidance of **RG 1.53** and **GDC 17**.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	F. Performance of nonsafety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of **SECY 77-439**, **SECY 94-084**, and **RG 1.206**.	Y	15.2.1 15.2.2 15.2.3 15.2.4
15.2.1-15.2.5-SAC-03	The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:	Y (Per AREVA Topical Report ANP-10263)	15.2.1 15.2.2 15.2.3 15.2.4
	A. The reactor is initially at 102 percent of the rated (licensed) core thermal power (to account for a 2 percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (see SAR (or DCD) Section 4.4)), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.	Y	15.2.1 15.2.2 15.2.3 15.2.4
	B. Conservative scram characteristics are assumed (*i.e.*, for a PWR maximum time delay with the most reactive rod held out of the core,	Y	15.2.1 15.2.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is otherwise accounted for (see SAR (or DCD) Section 4.4).		15.2.3 15.2.4
	C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Y	15.2.1 15.2.2 15.2.3 15.2.4
	D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with **Regulatory Guide 1.105**.	Y	15.2.1 15.2.2 15.2.3 15.2.4
SRP 15.2.6	**Loss of Non-Emergency AC Power to the Station Auxiliaries (R2, 03/2007)**		
15.2.6-AC-01	**General Design Criterion (GDC) 10**, as to RCS design with appropriate margin so specified acceptable fuel design limits are not exceeded during normal operation including anticipated operational occurrences.	Y	15.2.6
15.2.6-AC-02	**GDC 13**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.2.6
15.2.6-AC-03	**GDC 15**, as to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal	Y	15.2.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	operation including anticipated operational occurrences.		
15.2.6-AC-04	**GDC 26**, as to reliable control of reactivity changes so specified acceptable fuel design limits are not exceeded in anticipated operational occurrences. This control is accomplished by appropriate margin for malfunctions like stuck rods.	Y	15.2.6
15.2.6-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.	Y	15.2.6
15.2.6-SAC-02	Fuel cladding integrity should be maintained by keeping the minimum departure from nucleate boiling ratio (DNBR) above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) above the minimum critical power ratio safety limit for BWRs based on acceptable correlations (see **SRP Section 4.4**).	Y	15.2.6
15.2.6-SAC-03	An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	Y	15.2.6
15.2.6-SAC-04	For the requirements of **GDCs 10 and 15**, the positions of **Regulatory Guide (RG) 1.105**, "Instrument Setpoints for Safety-Related Systems," have impact on the plant response to the type of transient addressed in this SRP section.	Y	15.2.6
15.2.6-SAC-05	The most limiting plant system single failure, as defined in the "Definitions and Explanations" of **10 CFR Part 50, Appendix A**, must be assumed in the analysis and must satisfy the positions of **RG 1.53**.	Y	15.2.6
	The applicant's analysis of the loss of ac power transient should be based on an acceptable and NRC-approved model. If the applicant proposes analytical methods not approved, these are evaluated by the staff for acceptability and approval. For new generic methods, the	Y (Per AREVA Topical Reports ANP-10263 & ANP-10287)	15.2.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	reviewer requests an appropriate evaluation.		
	The parameter values in the analytical model should be suitably conservative. The following values are acceptable:		
	A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The number of loops (RCS loop requirements as applicable for BWR design) operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.	Y	15.2.6
	B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate).	Y	15.2.6
	C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile, and radial power distribution.	Y	15.2.6
	D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with **RG 1.105**. Compliance with RG 1.105 is determined.	Y	15.2.6
SRP 15.2.7	**Loss of Normal Feedwater Flow (R2, 03/2007)**		
15.2.7-AC-01	**General Design Criterion (GDC) 10**, as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal	Y	15.2.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	operations including anticipated operational occurrences (AOOs).		
15.2.7-AC-02	**GDC 13**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.2.7
15.2.7-AC-03	**GDC 15**, as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.	Y	15.2.7
15.2.7-AC-04	**GDC 17**, as it relates to providing onsite and offsite electric power systems to ensure that SSCs important to safety will function during normal operation, including anticipated operational occurrences. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded during an AOO.	Y	15.2.7
15.2.7-AC-05	**GDC 26**, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by assuring that appropriate margin for malfunctions, such as stuck rods, are accounted for.	Y	15.2.7
15.2.7-AC-06	TMI Action Plan item II.K.2.19 of **NUREG-0737** and **10 CFR 50.34(f)(1)(ii)** and **10 CFR 50.34(f)(2)(xii)** as they relate to the performance requirements of the auxiliary feedwater system for the loss of normal feedwater flow event.	Y	15.2.7
15.2.7-SAC-01	The basic objective in the review of the loss of normal feedwater transient		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	is to confirm that the following criteria are met:		
	A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.	Y	15.2.7
	B. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition.	Y	15.2.7
	C. The plant protection systems' setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in **Regulatory Guide 1.105**.	Y	15.2.7
	D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety related systems that are consistent with regulatory guidelines set forth in **RG 1.206**.	Y	15.2.7
15.2.7-SAC-02	Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of **GDCs 10, 13, 15, 17, and 26** for events of moderate frequency and they are as follows:	Y	15.2.7
	A. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.	Y	15.2.7
	B. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs, and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SAR (or DCD) Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design.	Y	15.2.7
	C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.	Y	15.2.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	D. To meet the requirements of **GDCs 10 and 15**, the positions of **Regulatory Guide 1.105**, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.	Y	15.2.7
	E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of **Appendix A to 10 CFR Part 50**, shall be identified and assumed in the analysis and shall satisfy the positions of **Regulatory Guide 1.53** and **GDC 17**.	Y	15.2.7
	F. The guidance provided in **SECY 77-439**, **SECY 94-084** and **RG 1.206** with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.	Y	15.2.7
15.2.7-SAC-03	The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10287)	15.2.7
	A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.	Y	15.2.7

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	B. Conservative scram characteristics are assumed, i.e., for a PWR – maximum time delay with the most reactive rod held out of the core and for a BWR – a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) the uncertainty has otherwise been accounted for (see SAR (or DCD) Section 4.4).	Y	15.2.7
	C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution.	Y	15.2.7
	D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with **Regulatory Guide 1.105**.	Y	15.2.7
SRP 15.2.8	**Feedwater System Pipe Breaks Inside and Outside Containment (R2, 03/2007)**		
15.2.8-AC-01	**General Design Criterion (GDC) 13**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.2.8
15.2.8-AC-02	**GDC 17**, as to onsite and offsite electric power systems for safety-related SSCs to function. The safety function for each power system (assuming the other system is not functioning) must be of sufficient capacity and capability so design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in postulated accidents.	Y	15.2.8
15.2.8-AC-03	**GDCs 27 and 28**, as to the RCS design with appropriate margin so	Y	15.2.8

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	acceptable fuel design limits are not exceeded and core cooling capability is maintained.		
15.2.8-AC-04	**GDC 31**, as to RCS design with sufficient margin so the boundary is nonbrittle and the probability of fracture propagation is minimized.	Y	3.1 15.2.8
15.2.8-AC-05	**GDC 35**, as to design of the RCS and its auxiliaries for abundant emergency core cooling.	Y	15.2.8
15.2.8-AC-06	**10 CFR Part 100**, as to calculated doses at the site boundary.	Y	15.2.8
15.2.8-SAC-01	Requirements for maintenance of adequate decay heat removal by the AFWS are in **10 CFR 50.34(f)(1)(ii)**, (TMI issue II E 1.1) and **10 CFR 50.34(f)(2)(xii)**, (TMI issue II E 1.2). Requirements for reactor coolant pump (RCP) operation are in **10 CFR 50.34(f)(1)(iii)**, (TMI issue 2 K 2). See Chapter 20 of the NRC FSAR for AP1000 to see how these post TMI requirements are met by the PRHR, the non-safety related start-up feedwater system (SUFWS) and the canned-motor RCPs of AP1000.	Y	15.2.8
15.2.8-SAC-02	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (**American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III**) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.	Y	15.2.8
15.2.8-SAC-03	The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations (see **SRP Section 4.4**). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an	Y	15.2.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	acceptable fuel damage model (see **SRP Section 4.2**) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.		
15.2.8-SAC-04	Calculated doses at the site boundary from any activity release must be a small fraction of the **10 CFR Part 100** guidelines.	Y	15.2.8
15.2.8-SAC-05	The integrity of the RCPs should be maintained so loss of alternating current power and containment isolation do not result in seal damage.	Y	15.2.8
15.2.8-SAC-06	The AFWS must be safety grade and automatically initiated when required.	Y	15.2.8
15.2.8-SAC-07	Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures:		
	A. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. These assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable.	Y	15.2.8
	B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine	Y	15.2.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of auxiliary feedwater and the consequent modification of the sequence of events.		
	C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of **Branch Technical Positions (BTP) 3-3** and **BTP 3-4**.	Y	15.2.8
	D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP in combination with a single, active failure. (This position is based upon interpretation of **GDC 17** as documented in the FSER for the ABB-CE System 80+ DC.)	Y	15.2.8
	E. The maximum rod worth should be assumed to be held in the fully withdrawn position per **GDC 25**. An appropriate rod reactivity worth versus rod position curve should be assumed.	Y	15.2.8
	F. The core burn-up (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.	Y	15.2.8
	G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the	Y	15.2.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

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SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	minimum DNBR margin is the result for a feedwater line rupture inside containment; however, this assumption may not be the most conservative. For example, maximum initial core flow increases RCS cool-down and depressurization, decreases shutdown margin, and increases the possibility that the core will become critical and return to power. As it is not clear which initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.		
	H. During the initial 10 minutes of the transient, if credit for operator action is required (i.e., RCP trip), an assessment for the limiting consequence must account for operator delay and/or error.	Y	15.2.8
SRP 15.3.1 – 15.3.2	**Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions (R2, 03/2007)**		
15.3.1-15.3.2-AC-01	**10 CFR 50, Appendix A, General Design Criteria (GDCs) 10 and 20** as to design of the reactor coolant system with appropriate margin so SAFDLs are not exceeded during normal operations, including anticipated operational occurrences (AOOs).	Y	15.3.1 15.3.2
15.3.1-15.3.2-AC-02	**10 CFR 50, Appendix A, GDC 13** as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.3.1 15.3.2
15.3.1-15.3.2-AC-03	**10 CFR 50, Appendix A, GDC 15** as to design of the reactor coolant system and its auxiliaries appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.	Y	15.3.1 15.3.2
15.3.1-15.3.2-AC-04	**10 CFR 50, Appendix A, GDC 17** as to onsite and offsite electric power	Y	15.3.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	systems so structures, systems, and components (SSCs) important to safety function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) must be to provide sufficient capacity and capability so SAFDLs and design conditions of the reactor coolant pressure boundary are not exceeded during AOOs.		15.3.2
15.3.1-15.3.2-AC-05	**10 CFR 50, Appendix A, GDC 26** as to the reliable control of reactivity changes so SAFDLs are not exceeded, including during AOOs. This control is accomplished by accounting for appropriate margin for malfunctions (e.g., stuck rods).	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-Objectives	Identify the most limiting transients	Y	15.3.1 15.3.2
	Verify whether, for the most limiting transients, the plant response to the loss of flow transients satisfies fuel damage and system pressure criteria.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-Specific Criteria	The following specific criteria are necessary to meet the regulatory requirements for incidents of moderate frequency:		
15.3.1-15.3.2-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-02	Fuel-cladding integrity must be maintained by the minimum DNBR remaining above the 95 percent probability/95 percent confidence DNBR limit for PWRs and the critical power ratio (CPR) remaining above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see **SRP Section 4.4**).	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-03	An incident of moderate frequency should not generate an aggravated	Y	15.3.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	plant condition without other faults occurring independently.		15.3.2
15.3.1-15.3.2-SAC-04	The requirements stated in **RG 1.105**, "Instrument Spans and Setpoints," are evaluated for their impact on the plant response to AOOs addressed in this SRP section.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-05	Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-06	The most limiting plant system single failure, as defined in the "Definitions and Explanations" of **10 CFR 50, Appendix A**, must be assumed in the analysis and should follow the guidance of **RG 1.53**.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-07	The performance of nonsafety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of **SECY 77-439**, **SECY 94-084** and **RG 1.206**.	Y	15.3.1 15.3.2
15.3.1-15.3.2-SAC-08	The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10287)	15.3.1 15.3.2
15.3.1-15.3.2-SAC-09	Parameter values in the analytical model should be suitably conservative. The following values are acceptable:		
	A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of 2% to account for power measurement uncertainty unless (i) a lower number can be justified through the measurement uncertainty methodology and evaluation or (ii) the uncertainty is accounted for	Y	15.3.1 15.3.2

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	otherwise. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.		
	B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core for a PWR, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR), unless (i) a different conservatism factor can be justified through the uncertainty methodology and evaluation or (ii) the uncertainty is accounted for otherwise.	Y	15.3.1 15.3.2
	C. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.	Y	15.3.1 15.3.2
	D, Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with **RG 1.10**5 and as determined by the organization responsible for instrumentation and controls.	Y	15.3.1 15.3.2
SRP 15.3.3 – 15.3.4	**Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break (R3, 03/2007)**		
15.3.3-15.3.4-AC-01	**General Design Criterion (GDC) 17**, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in the event of postulated accidents.	Y	15.3.3 15.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.3.3-15.3.4-AC-02	**General Design Criteria 27 (GDC 27) and 28 (GDC 28**), as they relate to the reactor coolant system being designed with appropriate margin to ensure that the capability to cool the core is maintained.	Y	15.3.3 15.3.4
15.3.3-15.3.4-AC-03	**General Design Criterion 31 (GDC 31)**, as it relates to the reactor coolant system being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.	Y	15.3.3 15.3.4
15.3.3-15.3.4-AC-04	**10 CFR Part 100**, as it relates to the calculated doses at the site boundary.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-Objectives	Identify which of these accidents is the more limiting.	Y	15.3.3 15.3.4
	Verify that, for the accident, the plant responds in such a way that the criteria regarding fuel damage, radiological consequences, and system pressure are met.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-Specific Criteria	The specific criteria necessary to meet the relevant requirements of **General Design Criteria 27, 28, and 31** and **10 CFR Part 100** for the rotor seizure and shaft break event are:		
15.3.3-15.3.4-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-02	The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR)	Y	15.3.3 15.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	safety limit for BWRs based on acceptable correlations (see **SRP Section 4.4**). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see **SRP Section 4.2**), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.		
15.3.3-15.3.4-SAC-03	Any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the **10 CFR Part 100** guidelines.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-04	The integrity of the reactor coolant pumps should be maintained such that loss of ac power and containment isolation will not result in pump seal damage.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-05	The auxiliary feedwater system must be safety grade and, when required, automatically initiated.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-06	A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-07	Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component.	Y	15.3.3 15.3.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	For new applications, loss of offsite power should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a loss of off-site power in combination with a single active failure.		
15.3.3-15.3.4-SAC-08	The ability to achieve and maintain long-term core cooling should be verified.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-09	This event should be analyzed assuming turbine trip and coincident loss of offsite power and coastdown of undamaged pumps.	Y	15.3.3 15.3.4
15.3.3-15.3.4-SAC-Analytical Model	**The applicant's analysis should be performed using an acceptable analytical model.** The equations, sensitivity studies, and models described in References 8 through 12 are acceptable. The NRC staff found References 13 and 14 to be acceptable transient analysis computer codes for design analysis of the Advanced Boiling Water Reactor (ABWR). References 15 through 19 were found to be acceptable computer codes for transient analyses (i.e., except for loss-of-coolant accidents, or LOCAs) for the Combustion Engineering System 80+ final safety evaluation report staff review. In addition, NUREG-1465 contains guidance on accident source terms for light-water nuclear power plants. When conducting transient analyses, the NUREG-1465 guidance is particularly important for reviewing fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine assumed to exist within the gap between fuel pellets and cladding. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10287)	15.3.3 15.3.4
	There are certain assumptions regarding important parameters used to		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	describe the initial plant conditions and postulated system failures which should be used. These are listed below:		
	1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance to account for power measurement uncertainties. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.	Y	15.3.3 15.3.4
	2. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N–1 reactor coolant pumps (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution will change as more pumps begin to coastdown following turbine trip and coincident loss of offsite power	Y	15.3.3 15.3.4
	3. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core, and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.	Y	15.3.3 15.3.4
	4. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.	Y	15.3.3 15.3.4
SRP 15.4.1	**Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (R3, 03/2007)**		
15.4.1-AC-01	The requirements of **GDC 10, 20, and 25** concerning the specified acceptable fuel design limits are assumed to be met for this event when:		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	A. **Criterion 10**, which requires that specified acceptable fuel design limits are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.	Y	15.4.1
	B. **Criterion 13**, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.1
	C. **Criterion 17**, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety.	Y	15.4.1
	D. **Criterion 20**, which requires that the protection system initiate automatically appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.	Y	15.4.1
	E. **Criterion 25**, which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system.	Y	15.4.1
15.4.1-SAC-01	The requirements of **GDC 10, 20, and 25** concerning the specified acceptable fuel design limits are assumed to be met for this event when:		
	A. The thermal margin limits (DNBR for PWRs and MCPR for BWRs) as specified in **SRP Section 4.4** are met.	Y	15.4.1
	B. Fuel centerline temperatures (for PWRs) as specified in **SRP Section 4.2** do not exceed the melting point.	Y	15.4.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 does not exceed 1%.	N/A-BWR	N/A
SRP 15.4.2	**Uncontrolled Rod Assembly Withdrawal at Power (R3, 03/2007)**		
15.4.2-AC-01	**General Design Criterion 10 (GDC 10)**, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not to be exceeded during any condition of normal operation, including the effects of AOOs.	Y	15.4.2
15.4.2-AC-02	**General Design Criterion 13 (GDC 13)**, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.2
15.4.2-AC-03	**General Design Criterion 17 (GDC 17)**, which requires, in part, that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety	Y	15.4.2
15.4.2-AC-04	**General Design Criterion 20 (GDC 20)**, which requires, in part, that the protection system shall be designed to initiate automatically the operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of AOOs.	Y	15.4.2
15.4.2-AC-05	**General Design Criterion 25 (GDC 25)**, which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout)	Y	15.4.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	of control rods.		
15.4.2-SAC-01	The requirements of **General Design Criteria 10, 17, 20, and 25** concerning the specified acceptable fuel design limits are assumed to be met for this event when:	Y	15.4.2
	A. The thermal margin limits departure from nucleate boiling ratio for PWRs and maximum critical power ratio for BWRs as specified in **SRP Section 4.4**, subsection II.1, are met.	Y	15.4.2
	B. Fuel centerline temperatures (for PWRs) as specified in **SRP Section 4.2**, subsection II.A.2(a) and (b), do not exceed the melting point.	Y	15.4.2
	C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1%.	N/A-BWR	N/A
SRP 15.4.3	**Control Rod Misoperation (System Malfunction or Operator Error) (R3, 03/2007)**		
15.4.3-AC-01	**General Design Criterion 10 (GDC 10)**, which requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not to be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	Y	15.4.3
15.4.3-AC-02	**General Design Criterion 13 (GDC 13)**, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.3
15.4.3-AC-03	**General Design Criterion 20 (GDC 20)**, which requires, in part, that the protection system shall be designed to initiate automatically the operation	Y	15.4.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.		
15.4.3-AC-04	**General Design Criterion 25 (GDC 25)**, which requires that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.	Y	15.4.3
15.4.3-SAC-Specific Criteria	The requirements of **General Design Criteria 10, 20, and 25** concerning the specified acceptable fuel design limits are assumed to be met for this event when:		
15.4.3-SAC-01	The thermal margin limits (departure from nucleate boiling ratio for PWRs) as specified in **SRP Section 4.4**, subsection II.1, are met.	Y	15.4.3
15.4.3-SAC-02	Fuel centerline temperatures as specified in **SRP Section 4.2**, subsection II.A.2(a) and (b), do not exceed the melting point.	Y	15.4.3
15.4.3-SAC-03	Uniform cladding strain as specified in **SRP Section 4.2**, subsection II.A.2(b), does not exceed 1%.	Y	15.4.3
SRP 15.4.4 – 15.4.5	**Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate (R2, 03/2007)**	(Refer to entries below) N/A-BWR	15.4.4 15.4.5
15.4.4-15.4.5-AC-01	**General Design Criterion (GDC) 10** and **GDC 20**, as they relate to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations and AOOs.	Y	15.4.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.4.4-15.4.5-AC-02	**GDC 13**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.4
15.4.4-15.4.5-AC-03	**GDC 15** and **GDC 28**, as they relate to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations and AOOs.	Y	15.4.4
15.4.4-15.4.5-AC-04	**GDC 26**, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs. This is accomplished by ensuring that appropriate margins for malfunctions, such as stuck rods, is accounted for.	N/A-BWR	N/A
15.4.4-15.4.5-AC-05	The basic objectives of the review of the AOOs described above are:		
	A. Identify which of the AOOs are the most limiting.	Y	15.4.4
	B. Verify that, for the most limiting AOOs, the plant responds in such a way that the criteria regarding fuel damage and system pressure are satisfied.	Y	15.4.4
15.4.4-15.4.5-SAC-Specific Criteria	Using the ANS standards as guidance, the specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows:		
15.4.4-15.4.5-SAC-A	Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values.	Y	15.4.4
15.4.4-15.4.5-SAC-B	Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95%	Y	15.4.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	probability/95% confidence DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs, based on acceptable correlations (see **SRP Section 4.4**).		
15.4.4-15.4.5-SAC-C	An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently	Y	15.4.4
15.4.4-15.4.5-SAC-D	The requirements stated in **Regulatory Guide 1.105**, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section.	Y	15.4.4
15.4.4-15.4.5-SAC-E	The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, shall be identified and assumed in the analysis and should satisfy the guidance stated in Regulatory Guide 1.53.	N/A-BWR	N/A
15.4.4-15.4.5-SAC-F	The guidance provided in **SECY 77-439**, **SECY 94-084** and **RG 1.206** with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified.	N/A-BWR	N/A
15.4.4-15.4.5-SAC-Analytical Models	The applicant's analysis of the most limiting AOOs should be performed using an acceptable model. If analytical methods that have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability.	Y	15.4.4
15.4.4-15.4.5-SAC-Parameters	The values of parameters used in the analytical model are to be suitably conservative. The following values are considered acceptable:		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	1. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty, unless (a) a lower number can be justified through the measurement uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see **SRP Section 4.4**). An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case.	Y	15.4.4
	2. Conservative scram characteristics are assumed, e.g., maximum time delay with the most reactive rod held out of the core for a PWR and a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate for a BWR, unless (a) a different conservatism factor can be justified through the uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see **SRP Section 4.4**).	N/A-OTHER (No reactor trip occurs for this event)	15.4.4
	3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile and radial power distribution.	Y	15.4.4
	4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with **Regulatory Guide 1.105** as determined by the organization responsible for instrumentation and controls.	N/A-OTHER (No mitigating systems used)	15.4.4
	The reviewer shall verify that the protection system (1) automatically initiates the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded for this event,	Y	15.4.4

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CHAPTER 15 **Accident Analysis**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and (2) senses the plant conditions and initiates the operation of SSCs important to safety.		
	For BWR plants where flow control is part of the reactivity control system, GDCs 26 and 28 must be satisfied for this event; otherwise, GDCs 26 and 28 are not applicable. Where applicable, GDCs 26 and 28 are satisfied if compliance with GDCs 10 and 15 is demonstrated.	N/A-BWR	N/A
SRP 15.4.6	**Inadvertent Decrease in Boron Concentration in the Reactor Coolant (PWR) (R2, 03/2007)**		
15.4.6-AC-01	**General Design Criterion (GDC) 10**, as it relates to the reactor core and its coolant, control, and protection systems with appropriate design margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	Y	15.4.6
15.4.6-AC-02	**GDC 13** as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.6
15.4.6-AC-03	**GDC 15**, as it relates to the RCS and its auxiliary, control, and protection systems with sufficient design margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences.	Y	15.4.6
15.4.6-AC-04	**GDC 26**, as it relates to the capability of control rods to reliably control reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences (AOOs), and with	Y	15.4.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	appropriate margin for malfunctions like stuck rods, specified acceptable fuel design limits are not exceeded.		
15.4.6-AC-05	The general objective of the review of moderator dilution events is to confirm either of the following conditions is met:	Y	15.4.6
	A. The consequences of these events are less severe than those of another transient that results in an uncontrolled increase in reactivity and has the same anticipated frequency classification.	Y	15.4.6
	B. The plant responds to events such that the criteria regarding fuel damage and system pressure are met and the dilution transient is terminated before the shutdown margin is eliminated.	Y	15.4.6
15.4.6-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.	Y	15.4.6
15.4.6-SAC-02	Fuel cladding integrity must be maintained so the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) based on acceptable correlations with **SRP Section 4.4**.	Y	15.4.6
15.4.6-SAC-03	An incident of moderate frequency should not generate a more serious than moderate plant condition without other faults occurring independently.	Y	15.4.6
15.4.6-SAC-04	If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:	Y	15.4.6

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	A. During refueling: 30 minutes. B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.		
15.4.6-SAC-05	The applicant's analysis of moderator dilution events should use an acceptable analytical model. Staff must evaluate any proposed unreviewed analytical methods. The reviewer initiates an evaluation of new generic methods.	Y	15.4.6
	The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown.	Y	15.4.6
	Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are acceptable: A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of 2 percent to account for power-measurement uncertainty. The analysis may use a smaller power- measurement uncertainty if justified adequately.	Y	15.4.6
	B. The boron dilution is assumed to occur at the maximum possible rate.	Y	15.4.6
	C. Core burnup and corresponding boron concentration must yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup must be justified by either analysis or evaluation.	Y	15.4.6
	D. All fuel assemblies are installed in the core.	Y	15.4.6
	E. A conservatively low value is assumed for the reactor coolant volume.	Y	15.4.6

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	F. For analyses during refueling, all control rods are withdrawn from the core. An alternate assumption requires adequate justification and delineation of necessary controls so the alternate assumption remains valid.	Y	15.4.6
	G. For analyses during power operation, the minimum shutdown margin allowed by the technical specifications (usually 1 percent) is assumed prior to boron dilution.	Y	15.4.6
	H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution.	Y	15.4.6
	I. Conservative scram characteristics are assumed (i.e., maximum time delay with the most reactive rod out of the core)	Y	15.4.6
SRP 15.4.7	**Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (R2, 03/2007)**		
15.4.7-AC-01	**General Design Criterion 13 (GDC 13)**, as it relates to providing instrumentation to monitor variables over anticipated ranges for normal operations, for anticipated operational occurrences, and for accident conditions.	Y	15.4.7
15.4.7-AC-02	**10 CFR Part 100**, as it relates to offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly.	Y	15.4.7
15.4.7-SAC-01	To meet the requirements of **GDC 13**, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations.	Y	15.4.7
15.4.7-SAC-02	In the event the error is not detectable by the instrumentation system and	Y	15.4.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the **10 CFR Part 100** criteria. A small fraction is interpreted to be less than 10% of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of any fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.		
SRP 15.4.8	**Spectrum of Rod Ejection Accidents (PWR) (R3, 03/2007)**		
15.4.8-AC-01	**General Design Criterion (GDC) 13**, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.4.8
15.4.8-AC-02	Acceptance criteria are based on meeting **GDC 28** requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficient damage to impair significantly core cooling capacity. Regulatory positions and specific guidelines necessary to meet the relevant requirements of **GDC 28** are in **Regulatory Guide 1.77** and **SRP Section 4.2**. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that exceed the "Service Limit C" as defined in the **American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code**.	Y	15.4.8

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.4.8-AC-03	**10 CFR 100.11** and **10 CFR 50.67** establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under **SRP Section 15.0.3**. **SRP Section 4.2** describes fuel rod failure mechanisms. Guidance for calculating radiological consequences is in **Regulatory Guides 1.183 and 1.195**.	Y	15.4.8
SRP 15.4.8.A	**Radiological Consequences of a Control Rod Ejection Accident (PWR) (R1, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.4.9	**Spectrum of Rod Drop Accidents (BWR) (R3, 03/2007)**	N/A-BWR	N/A
SRP 15.4.9.A	**Radiological Consequences of Control Rod Drop Accident (BWR) (R2, 07/1981)**	N/A-BWR	N/A
SRP 15.5.1 – 15.5.2	**Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (R2, 03/2007)**		
15.5.1-15.5.2-AC-01	**GDC 10**, which requires that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.	Y	15.5.1 15.5.2
15.5.1-15.5.2-AC-02	**GDC 13**, which requires in part that the effect of instrumentation shall be provided to monitor variables and systems over their anticipated ranges for anticipated operational occurrences to assure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.	Y	15.5.1 15.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15
Accident Analysis

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
15.5.1-15.5.2-AC-03	**GDC 15**, which requires that the reactor coolant system and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operations, including anticipated operational occurrences.	Y	15.5.1 15.5.2
15.5.1-15.5.2-AC-04	**GDC 26**, which requires, in part that the reliable control of reactivity changes to assure that specified acceptable fuel design limits are not exceeded under conditions of normal operation, including anticipated operational occurrences, with appropriate margin for malfunctions, such as stuck rods.	Y	15.5.1 15.5.2
15.5.1-15.5.2-AC-05	For plants with licensing bases that incorporate RG 1.70, ANS 51.1 (for PWRs), or ANSI/ANS-52.1-1978 (for BWRs), there are acceptance criteria, in SRP 15.0, "Condition II" events, or events of moderate frequency, "Condition III events, or infrequent events, and "Condition IV" events, or postulated accidents of low probability. Acceptance criteria are also defined for Condition II, III, and IV events. Regulatory Issue Summary (RIS) 2005-29, which relates to the escalation of a Condition II event into a Condition III or IV event, is also applicable to these plants.	Y	15.5.1 15.5.2
15.5.1-15.5.2-SAC-Objectives	The basic objectives in reviewing the events leading to an increase in reactor coolant inventory are:		
	1. Identify which of the AOOs leading to an RCS inventory increase are the most limiting.	Y	15.5.1 15.5.2
	2. Verify that for the most limiting transients the plant responds to the RCS inventory increase in such a way that the criteria regarding fuel damage, RCS pressure, and escalation to a more serious event are	Y	15.5.1 15.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	met.		
15.5.1-15.5.2-SAC-Specific Criteria	The specific acceptance criteria derived from GDC 10, 13, 15, and 26, and from the aforementioned ANS standards, are:		
15.5.1-15.5.2-SAC-01	Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the **ASME Boiler and Pressure Vessel Code**.	Y	15.5.1 15.5.2
15.5.1-15.5.2-SAC-02	Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs based on acceptable correlations (see **SRP Section 4.4**).	Y	15.5.1 15.5.2
15.5.1-15.5.2-SAC-03	An AOO should not generate a more serious plant condition without other faults occurring independently.	Y	15.5.1 15.5.2
15.5.1-15.5.2-SAC-Evaluation Models	**The applicant's analysis of events leading to an increase of reactor coolant inventory should be performed using an acceptable analytical model. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability.** For new generic methods, the reviewer performs an evaluation of the new method as part of its review under this SRP section.	Y (Per AREVA Topical Reports ANP-10263 & ANP-10287)	15.5.1 15.5.2
15.5.1-15.5.2-SAC-Parameters	The values of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:		
	1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties,	Y	15.5.1 15.5.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.		
	2. Conservative scram characteristics are assumed, i.e., for a PWR maximum time delay with the most reactive rod held out of the core and for a BWR a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.	Y	15.5.1 15.5.2
	3. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.	Y	15.5.1 15.5.2
SRP 15.6.1	**Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve (R2, 03/2007)**		
15.6.1-AC-01	**General Design Criterion (GDC) 10**, as it relates to designing the RCS with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs.	Y	15.6.1
15.6.1-AC-02	**GDC 13**, as it relates to providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation to assure adequate safety, and to providing appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.6.1
15.6.1-AC-03	**GDC 15**, as it relates to designing the RCS and associated auxiliary systems with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operations, including AOOs.	Y	15.6.1
15.6.1-AC-04	**GDC 26**, as it relates to providing a reactivity control system capable of	Y	15.6.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

<table>
<tr><th colspan="4">CHAPTER 15
Accident Analysis</th></tr>
<tr><th>SRP Criterion</th><th>Description
(AC – Acceptance Criteria Requirement,
SAC – Specific SRP Acceptance Criteria)</th><th>U.S. EPR Assessment</th><th>FSAR Section(s)</th></tr>
<tr><td></td><td>reliably controlling reactivity changes during manual operations and AOOs so the specified fuel design limits are not exceeded.</td><td></td><td></td></tr>
<tr><td>15.6.1-AC-05</td><td>10 CFR 52.47(a) and 52.79(a), as they relate to demonstrating compliance with any technically relevant portions of the Three Mile Island (TMI)-related requirements set forth in 10 CFR 50.34(f)(1)(vi) and 10 CFR 50.34(f)(1)(iii), for DC and COL reviews, respectively.</td><td>Y</td><td>15.6.1</td></tr>
<tr><td>15.6.1-SAC-01</td><td>Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values</td><td>Y</td><td>15.6.1</td></tr>
<tr><td>15.6.1-SAC-02</td><td>Fuel cladding integrity is maintained if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4).</td><td>Y</td><td>15.6.1</td></tr>
<tr><td>15.6.1-SAC-03</td><td>An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.</td><td>Y</td><td>15.6.1</td></tr>
<tr><td>SRP 15.6.2</td><td>Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (R2, 07/1981)</td><td>N/A-SUP
(Refer to SRP 15.0.3)</td><td>N/A</td></tr>
<tr><td rowspan="2">SRP 15.6.3</td><td rowspan="2">Radiological Consequences of Steam Generator Tube Failure (PWR) (R7, 07/1981)</td><td>(Entries below address Accident Analysis of Tube Failure)</td><td></td></tr>
<tr><td>N/A-SUP
(Refer to SRP 15.0.3 for radiological consequences)</td><td>N/A</td></tr>
</table>

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.6.3-AC-01	The acceptance criteria are based on the relevant requirements of **10 CFR Part 100** as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating engineered safety features are acceptable with respect to the radiological consequences of a postulated steam generator tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the following exposure guidelines:	Y	15.6.3
	(1) for the postulated accident with an assumed pre-accident iodine spike in the reactor coolant and for the postulated accident with the highest worth control rod stuck out of the core the calculated doses should not exceed the guideline values of **10 CFR Part 100**, Section 11 (Ref. 1), and	Y	15.6.3
	(2) for the postulated accident with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses.	Y	15.6.3
15.6.3-AC-02	The methodology and assumptions for calculating the radiological consequences should reflect the regulatory positions of **Regulatory Guide 1.4** (Ref. 2) except for the atmospheric dispersion factors which are reviewed under **SRP Section 2.3.4**. Plant technical specifications are required for iodine activity in the primary and secondary coolant systems. These specifications are acceptable if the calculated potential radiological consequences from the steam generator tube failure accident are within	Y	15.6.3
		N/A-SUP (Refer to SRP 15.0.3 relative to RG 1.4 replacement by RG 1.183)	15.0.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the exposure guidelines for the above two cases.		
SRP 15.6.4	**Radiological Consequences of Main Steam Line Failure Outside Containment (BWR) (R2, 07/1981)**	N/A-BWR	N/A
SRP 15.6.5	**Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary (R3, 03/2007)**		
15.6.5-AC-01	**10 CFR 50.46** as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.	Y	15.6.5
15.6.5-AC-02	**GDC 13** as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.	Y	15.6.5
15.6.5-AC-03	**General Design Criterion (GDC) 35** as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts. The analyses should reflect that the ECCS has suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities available such that the safety functions could be accomplished assuming a single failure. In addition, consideration should be given to the availability of onsite power (assuming offsite electric power is not available with onsite electric power	Y	15.6.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	available; or assuming onsite electric power is not available with offsite electric power available).		
15.6.5-AC-04	**10 CFR 100** or **10 CFR 50.67** as they relate to mitigating the radiological consequences of an accident.	Y	15.6.5
15.6.5-SAC-01	An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of **10 CFR 50.46**. **Regulatory Guide 1.157** and **Section I of Appendix K to 10 CFR Part 50** provide guidance on acceptable evaluation models. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for reactor coolant pump operation during a small break loss-of-coolant accident, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied. This also includes analyses of a spectrum of large break and small break LOCAs to assure boric acid precipitation is precluded for all break sizes and locations.	Y	15.6.5
	The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in **10 CFR 50.46(a)(1) or (2)**. The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available).	Y	15.6.5
	Additionally, the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying **10 CFR**	Y (Per AREVA Topical	15.6.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	50.46(c)(2), and the analysis results should meet the performance criteria in **10 CFR 50.46(b)**.	Report ANP-10278)	
	A. The calculated maximum fuel element cladding temperature does not exceed 1200 °C (2200 °F).	Y	15.6.5
	B. The calculated total local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.	Y	15.6.5
	C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.	Y	15.6.5
	D. Calculated changes in core geometry are such that the core remains amenable to cooling.	Y	15.6.5
	E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.	Y	15.6.5
15.6.5-SAC-02	The radiological consequences of the most severe LOCA are within the guidelines of and **10 CFR 100** or **10 CFR 50.67**. For applications under 10 CFR Part 52, reviewers should use **SRP Section 15.0.3**, “Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications.”	Y	15.6.5
15.6.5-SAC-03	The TMI Action Plan requirements for **II.E.2.3**, **II.K.2.8**, **II.K.3.5**, **II.K.3.25**,		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	II.K.3.30, **II.K.3.31**, and **II.K.3.40** have been met.		
	A. II.E.2.3 - Uncertainties in Performance Predictions Small-break LOCA analyses performed by LWR vendors to develop operator guidelines had shown that large uncertainties may exist in system thermal-hydraulic response due to modeling assumptions or inaccuracies. It is necessary to establish that these assumptions or inaccuracies are properly accounted for in determining the acceptability of ECCS performance pursuant to Appendix K of 10 CFR 50.	Y	15.6.5
	B. II.K.2.8 - The licensees of B&W reactors were required to comply with the Commission Orders regarding certain short-term and long-term auxiliary feedwater system (AFWS) modifications. The staff evaluated the short-term actions, and safety evaluations were prepared before the plants were allowed to return to operation. The staff evaluation of the additional (long-term) items will be performed in conjunction with item II.E.1.1 in NUREG-0660, Auxiliary Feedwater System Evaluation and item II.E.1.2, AFWS Automatic Initiation and Flow Indicator.	Y	15.6.5
	C. II.K.3.5 – Tripping of the reactor coolant pumps in case of a LOCA is not an ideal solution. Licensees should consider other solutions to the small-break LOCA problem (for example, an increase in safety injection flow rate). In the meantime, until a better solution is found, the reactor coolant pumps should be tripped automatically in case of a small-break LOCA. The signals designated to initiate the pump trip are	Y	15.6.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 **Accident Analysis**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	discussed in NUREG-0623.		
	D. II.K.3.25 - Provide by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.	Y	15.6.5
	E. II.K.3.30 – The analysis methods used for SBLOCA analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.	Y (Per AREVA Topical Report ANP-10278)	15.6.5
	F. II.K.3.31 - Use NRC-approved models for SBLOCAs as described in TMI Action Item II.K.3.30 to show compliance with 10 CFR 50.46.	Y	15.6.5
	G. II.K.3.40 – Provide an evaluation of the potential for RCP seal damage and leakage during SBLOCA.	Y	15.6.5
SRP 15.6.5.A	**Radiological Consequences of a Design Basis Loss-of-Coolant Accident including Containment Leakage Contribution (R2, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.6.5.B	**Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment (R2, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 15.6.5.D	**Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR) (R2, 07/1981)**	N/A-BWR	N/A
SRP 15.7.3	**Postulated Radioactive Releases Due to Liquid-Containing Tank Failures**	N/A-SUP (Refer to BTP 11-6 and SRP 11.2)	N/A
SRP 15.7.4	**Radiological Consequences of Fuel Handling Accidents (R2, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.7.5	**Spent Fuel Cask Drop Accidents (R2, 07/1981)**	N/A-SUP (Refer to SRP 15.0.3)	N/A
SRP 15.8	**Anticipated Transients Without Scram (R2, 03/2007)**		
15.8-AC-01	**10 CFR 50.62** (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy	Y	15.8
15.8-AC-02	**10 CFR 50.46**, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-AC-03	**GDC 12**, found in Appendix A to 10 CFR Part 50, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-AC-04	**GDC 14**, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
15.8-AC-05	**GDC 16**, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-AC-06	**GDC 35**, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reactor must be limited to negligible amounts	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-AC-07	**GDC 38**, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-AC-08	**GDC 50**, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.	N/A-OPT (Refer to SRP 15.8, 15.8-AC-01)	15.8
15.8-SAC-01	Acceptance criteria for Boiling Water Reactors (BWRs)	N/A-BWR	N/A
15.8-SAC-02	Acceptance criteria for Pressurized Water Reactors (PWRs):		
	A. Provide measures to automatically initiate the auxiliary (or emergency) feedwater system and a turbine trip under conditions indicative of an ATWS. This equipment shall be independent and diverse from the reactor trip system from sensor output to the final actuation device.	Y	15.8
	B. Combustion Engineering or Babcock and Wilcox reactors applicants shall have provision for a scram system that is independent and diverse from the reactor trip system, from sensor output to the points of interruption of power to the control rods.	Y	15.8

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CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	C. These system and equipment shall be demonstrated to provide reasonable assurance that unacceptable plant conditions do not occur in the event of an anticipated transients	Y	15.8
	D. The reactor coolant system (RCS) pressure shall not exceed ASME Service Level C limits (approximately 22 MPa or 3200 psig) containment safety parameters (e.g., temperature or pressure) should not exceed design limits	Y	15.8
15.8-SAC-03	Acceptance criteria for Evolutionary Plants:		
	A. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either of the following: i. A diverse scram system satisfying the design and quality assurance criteria specified in **SRP Section 7.2** ii. Demonstrate that the consequences of an ATWS event are within acceptable values.	Y	15.8
	B. For evolutionary plants, some of the equipment required to satisfy the rule may not be apply. For example, passive BWRs do not have recirculation pumps; therefore, these designs cannot provide equipment to trip them as required by the rule. For these designs provision of an equivalent action such as reducing the vessel water level may be acceptable.	N/A-BWR	N/A
	C. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small because either: (1) the criteria are met, or (2) a **diverse scram system** is installed that reduces significantly the probability of a failure to scram. The analysis	Y	15.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	leading to the ATWS rule in **NUREG-0460** used the following ATWS success criteria, which have their bases in the Commission regulations and GDC listed above. Applicant's design shall maintain :		
	i. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221"C (2200"F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen.	N/A-OPT (Refer to SRP 15.8, 15.8-SAC-03.C)	15.8
	ii. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the 'emergency conditions' as defined in the ASME Nuclear Power Plant, Components Code, Section III." The acceptance criteria for reactor coolant pressure, based upon the ASME Service Level C limits, are approximately 10.3 MPa (1500 psig) for BWRs and approximately 22MPa (3200 psig) for PWRs.	N/A-OPT (Refer to SRP 15.8, 15.8-SAC-03.C)	15.8
	iii. Maintain containment Integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment	N/A-OPT (Refer to SRP 15.8, 15.8-SAC-03.C)	15.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 15 Accident Analysis			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.		
SRP 15.9	**Boiling Water Reactor Stability (03/2007)**	N/A-BWR	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 16
Technical Specifications

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 16.0	**Technical Specifications (R2, 03/2007)**		
16.0-AC	Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations: **10 CFR 50.34(b)(6)(vi)**, **10 CFR 50.36**, **10 CFR 50.36a**, **10 CFR 52.47(a)(11)**, and **10 CFR 52.79(a)(30)** require that applications include proposed TS.	(See SAC responses below)	
16.0-SAC-Specifc Criteria	The proposed plant-specific TS satisfy 10 CFR 50.34, 10 CFR 50.36, and 10 CFR 50.36a and are therefore acceptable if consistent with the regulatory guidance of the following STS documents and present plant-specific values for parameters at the indicated level of detail: • NUREG-1430, STS, Babcock and Wilcox Plants • NUREG-1431, STS, Westinghouse Plants • NUREG-1432, STS, Combustion Engineering Plants • NUREG-1433, STS, General Electric Plants, BWR/4 • NUREG-1434, STS, General Electric Plants, BWR/6 In TS change requests for facilities with TS based on previous STS, licensees should comply with comparable provisions in these STS NUREGs to the extent possible or justify deviations from the STS. Acceptable justifications for deviation would include retention of existing TS requirements, nonadoption of STS requirements not represented in existing TS (e.g., an LCO in STS but not in existing TS), editorial preference, facility design, and a technically justified alternative presentation equivalent to the STS intent. In some cases, comparison to the previous STS may help evaluate the proposed changes by clarifying the TS intent. The previous STS NUREGs are as follows:	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 16 Technical Specifications			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	• NUREG-0103, STS, Babcock and Wilcox Plants • NUREG-0452, STS, Westinghouse Plants • NUREG-0212, STS, Combustion Engineering Plants • NUREG-0123, STS, General Electric Plants		
	For applicants referencing a certified design, the certified generic TS of the referenced design provide the guidelines for the evaluation of proposed plant-specific TS.	(See SAC responses below)	
16.0-SAC-01	**10 CFR 50.34** requires applicants at the construction permit stage to justify the selection of those variables, conditions, or other items identified through preliminary safety analysis as probable subjects for plant-specific TS. Special attention should be given to items that could influence the final design significantly. At the operating license stage, compliance with 10 CFR 50.34 requires applicants to propose TS in accordance with **10 CFR 50.36**. 10 CFR 50.34 requirements provide assurance that items requiring special attention at the construction permit stage are identified sufficiently early to preclude the need for any significant design change to support the facility's final TS. To meet 10 CFR 50.34 requirements at the operating license stage requires that the applicant propose TS in accordance with 10 CFR 50.36. The safety benefit of such a process is addressed in subsection II.2 of this SRP section.	N/A-CP/OL	N/A
16.0-SAC-02	**10 CFR 52.47(a)(11)** requires DC applications license to include TS prepared in accordance with **10 CFR 50.36** and **10 CFR 50.36a**.	Y	16.0
16.0-SAC-03	**10 CFR 52.79(b)** requires COL applications license to include TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a.	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 16
Technical Specifications

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
16.0-SAC-04	**10 CFR 50.36** requires proposed TS to include the following:		
	A. **10 CFR 50.36(d)(1)(i)(A)** Safety Limits. Safety limits apply to important process variables necessary for an appropriate level of protection for the integrity of certain physical barriers that guard against the uncontrolled release of radioactive material.	Y	16.2
	B. **10 CFR 50.36(d)(1)(ii)(A)** Limiting Safety System Settings. Limiting safety system settings are for automatic protective devices affecting variables with significant safety functions.	Y	16.3.3
	C. **10 CFR 50.36(d)(2)** LCOs. LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a LCO of a nuclear reactor is not met, the licensee must shut down the reactor or follow any remedial action permitted by the TS until the condition can be met. A TS LCO of a nuclear reactor must be established for each item meeting one or more of the following **10 CFR 50.36(d)(2)(ii)** criteria: (i) Criterion 1. Installed instrumentation used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. (ii) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. (iii) Criterion 3. An SSC part of the primary success path functioning	Y	16.3

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 16
Technical Specifications

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	or actuating to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. (iv) Criterion 4. An SSC which operating experience or PRA has shown to be significant to public health and safety.		
	D. **10 CFR 50.36(d)(3)** Surveillance Requirements (SRs). SRs relate to test, calibration, or inspection to maintain the necessary system and component quality, to operate the facility operation within safety limits, and to meet LCOs.	Y	16.3
	E. **10 CFR 50.36(d)(4)** Design Features. Design features affect aspects of the facility (e.g., construction materials and geometric arrangements) not covered in the categories described above that, if altered or modified, would have significant effects on safety.	Y	16.3
	F. **10 CFR 50.36(d)(5)** Administrative Controls. Administrative controls are provisions for organization and management, procedures, record-keeping, review and audit, and reporting necessary for safe operation of the facility.	Y	16.5
16.0-SAC-05	**10 CFR 50.36** requirements provide assurance that essential safety-related items or issues of facility design and operation (i.e., those derived from analyses and evaluations included in the safety analysis report) are identified.	N/A-INFO	N/A
16.0-SAC-06	**10 CFR 50.36a** requires each licensee of a nuclear power reactor to include TS that require (A) operating procedures for the control of effluents and (B) annual reports of the quantity of principal radionuclides	Y	16.5

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 16 Technical Specifications			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	released to unrestricted areas in both gaseous and liquid effluents. The STS contain model TS for radiological effluents in Section 5.5, “Programs and Manuals,” and Section 5.6, “Reporting Requirements.”		
SRP 16.1	**Risk-Informed Decision Making: Technical Specifications (R1, 03/2007)**	N/A-OTHER (Risk-based Technical Specifications not utilized)	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 17
Quality Assurance

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 17.1	**Quality Assurance During the Design and Construction Phases (R2, 07/1981)**	N/A-SUP (See SRP 17.5)	17.1
SRP 17.2	**Quality Assurance During the Operations Phase (R2, 07/1981)**	N/A- SUP (See SRP 17.5)	17.2
SRP 17.3	**Quality Assurance Program Description (09/1990)**	N/A-SUP (See SRP 17.5)	17.3
SRP 17.4	**Reliability Assurance Program (RAP) (03/2007)**		
17.4-AC-01	Per Commission policy in the **staff requirement memorandum (SRM) on SECY 95-132**, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (**SECY 94-084**),” Item E, Reliability Assurance Program, the requirement to provide a reliability assurance program is codified by incorporation within the design-specific rulemaking for an applicant for design certification. In addition, this becomes part of an application for a combined license that references that certified design. Furthermore, the RAP for the design stage will be verified using the ITAAC process.	Y	17.4
17.4-AC-02	**10 CFR 52.47(b)(1)**, which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations	ITAAC	Tier 1
17.4-AC-03	**10 CFR 52.80(a)**, which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 17 Quality Assurance			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.		
17.4-SAC-A	<u>Design Certification</u> The application describes the following RAP information:		
	1. The scope and purpose. The scope and purpose of the RAP are described in Subsections I and II of this SRP section.	Y	17.4
	2. The application of the quality elements associated with organization, design control, procedures and instructions, records, corrective action, and audit plans as follows: a. Organization (1) The organizations responsible for formulating and implementing the RAP and the coordination of RAP program activities, including those performed within the design, PRA, RAP, and risk and reliability organizations as well as work completed by the architect-engineers and other supporting organizations that develop deterministic and other methods used to identify SSCs that are significant contributors to plant safety. (2) How the reliability and design organizations manage interface issues. For example, how does the risk and reliability organization keep the design staff cognizant of SSCs that are significant contributors to plant safety, program needs, and	Y	17.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 17 Quality Assurance			
SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	status, and how does the feedback process ensure that significant design assumptions related to equipment reliability are realistic and achievable. (3) The risk and reliability organization participation in the design change control process for the purpose of providing RAP related inputs in the design process. (4) The risk and reliability organization involvement in design reviews. b. Design Control (1) The configuration control process for maintaining the list of SSCs within the scope of RAP similar to the control of a quality list. (2) How the design control and change processes provide a feedback mechanism for notifying the appropriate organization of changes in the design of SSCs within the scope of the RAP that could affect the probabilistic/PRA, deterministic, or other methods used to identify SSCs that are significant contributors to plant safety. (3) The interface between the risk and reliability and the design organizations for determining that the performance of SSCs within the scope of the RAP relate to the reliability assumptions in the probabilistic/PRA, deterministic, or other methods used to identify SSCs that are significant contributors to plants safety. (4) Engineering design controls applied for determining the SSCs within the scope of the RAP.		

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 17 Quality Assurance			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	(5) The process for proposing an alternative design to reliability performance. For example, is the revised design reviewed to provide confidence that the current reliability assumptions are still valid. c. The controls for procedures and instructions used to implement the RAP. d. The controls for records of activities involving SSCs within the scope of RAP. e. The corrective action process applied to SSCs within the scope of RAP. f. The audit plans for conducting QA audits of RAP activities.		
	3. The expert panel qualifications in the areas of personnel knowledgeable in the design, operation and maintenance of a plant, and experience necessary to perform the SSC selections if an expert panel is utilized.	Y	17.4
	4. Deterministic or other methods of analysis used to identify SSCs included in the RAP and the SSCs affected.	Y	17.4
	5. A non-system-based ITAAC for the RAP that provides reasonable assurance that the design of SSCs within the scope of the RAP is consistent with their assumed design reliability. The ITAAC acceptance criteria should ensure that the estimated reliability of each as-built SSC is at least equal to the assumed design reliability and that industry experience including operations, maintenance, and monitoring activities were assessed in estimating the reliability of these SSCs.	Y	17.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 17 Quality Assurance			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	6. A COL action item that a COL applicant referencing a certified design will identify the site-specific SSCs within the scope of the RAP.	N/A-COL	17.4
17.4-SAC-B	<u>COL Applicant</u>	N/A-COL	N/A
SRP 17.5	**Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants (03/2007)**	Y (Per AREVA Topical Report ANP-10266-A)	17.5
SRP 17.6	**Maintenance Rule (03/2007)**	N/A-COL	17.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 18.0	**Human Factors Engineering (R2, 03/2007)**		
	Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:		
18.0-AC-01	**10 CFR 50.34(f)(1)(i)**	Y	18.6
18.0-AC-02	**10 CFR 50.34(f)(2)**	(See below)	
	10 CFR 50.34(f)(2)(i)	Y	18.7
	10 CFR 50.34(f)(2)(ii)	Y	18.8
		N/A-COL	18.8
	10 CFR 50.34(f)(2)(iii)	Y	18.7
	10 CFR 50.34(f)(2)(iv)	Y	18.7
	10 CFR 50.34(f)(2)(v)	Y	18.7
	10 CFR 50.34(f)(2)(vi)	Y	18.7
	10 CFR 50.34(f)(2)(xi)	Y	18.7
	10 CFR 50.34(f)(2)(xii)	Y	18.7
	10 CFR 50.34(f)(2)(xiii)	Y	18.7
	10 CFR 50.34(f)(2)(xv)	Y	18.7
	10 CFR 50.34(f)(2)(xvii)	Y	18.7
	10 CFR 50.34(f)(2)(xviii)	Y	18.7
	10 CFR 50.34(f)(2)(xix)	Y	18.7
	10 CFR 50.34(f)(2)(xxv)	Y	18.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	10 CFR 50.34(f)(2)(xxvii)	Y	18.7
18.0-AC-03	**10 CFR 50.34(f)(3)(i)**	Y	18.2
		N/A-COL	18.2
18.0-AC-04	**10 CFR 50.34(f)(3)(vii)**	Y	18.1
		N/A-COL	18.1
18.0-AC-05	**10 CFR 50.54 (i) to (m)**	Y	18.5
		N/A-COL	18.5
18.0-AC-06	**10 CFR 50.120**	N/A-COL	18.9
18.0-AC-07	**10 CFR 52.47**	Y	18.2
		N/A-COL	18.6
18.0-AC-08	**10 CFR 52.79**	N/A-COL	18.6
			18.9
18.0-AC-09	**10 CFR 52.80**	N/A-COL	N/A
18.0-AC-10	**10 CFR Part 55**	N/A-COL	18.9
18.0-SAC-A	<u>Review of the HFE Aspects of a New Plant</u>		
18.0-SAC-A.01	HFE Program Management The objective of this review is to confirm that the applicant has adequately considered the role of HFE and the means by which HFE activities will be accomplished. The review should verify that: • The applicant has identified plans to oversee design and construction of the nuclear facility in accordance with the requirements of **10 CFR 50.34(f)(3)(vii)**, as described in **SRP Section 13.1.1**, “Management	Y	18.1
		N/A-COL	18.1

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 **Human Factors Engineering**			
SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	and Technical Support Organization.”		
	• The applicant has an HFE design team with the responsibility, authority, placement within the organization, and composition to ensure that the design commitment to HFE is achieved. There is, however, no assumption that HFE is the responsibility of a single organization or that there is an organizational unit called the HFE design team. • The team is guided by an HFE program plan to ensure the proper development, execution, oversight, and documentation of the HFE program.	Y (Per AREVA Topical Report ANP- 10279)	18.1
	• The overall HFE program appropriately considers and addresses the deterministic aspects of the design, as discussed in **RG 1.174**.	Y	18.1
	The HFE program plan should describe the technical program in sufficient detail to ensure that all aspects of the HSIs, procedures, and training are developed, designed, and evaluated on the basis of a structured top-down systems analysis using accepted HFE principles. The applicant’s HFE program management should be evaluated in accordance with the review criteria of **NUREG-0711**, “Human Factors Engineering Program Review Model.”	N/A-COL	18.1
18.0-SAC-A.02	Operating Experience Review The objective of this review is to verify that the applicant has identified and analyzed HFE-related problems and issues in previous designs so that these problems and issues may be avoided in the development of the new design. This review should also verify that the applicant has retained positive	Y (Per AREVA Topical Report ANP- 10279)	18.2

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	features of previous designs. The operating experience review (OER) should be evaluated in accordance with the review criteria of **NUREG-0711** and should satisfy the requirements of **10 CFR 50.34(f)(3)(i)** and **52.47(a)(22)**.		
18.0-SAC-A.03	Functional Requirements Analysis and Function Allocation Functional requirements analysis is the identification and analysis of those functions that must be performed to satisfy plant safety objectives; that is, to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Function allocation analysis is the analysis of requirements for plant control and the assignment of control functions to (1) personnel (e.g., manual control), (2) system elements (e.g., automatic control and passive, self-controlling phenomena), and (3) combinations of personnel and system elements (e.g., shared control, automatic systems with manual backup). The objective of this review is to verify that (1) the plant's functions that must be performed to satisfy plant safety objectives have been defined, and (2) that the allocation of those functions to human and system resources has resulted in a role for personnel that takes advantage of human strengths and avoids human limitations. Functional requirements analysis and function analysis should be evaluated in accordance with the review criteria of **NUREG-0711**	Y (Per AREVA Topical Report ANP- 10279)	18.3
18.0-SAC-A.04	Task Analysis Task analysis is the analysis of human performance that results from the allocation of functions to personnel and the identification of HSI characteristics needed to support personnel task accomplishment. The	Y (Per AREVA Topical Report ANP- 10279)	18.4

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	objective of this review is to ensure that the applicant's task analysis identifies the specific tasks that are needed for function accomplishment and their information, control, and task support requirements. The task analysis should be evaluated in accordance with the review criteria of **NUREG-0711**.		
18.0-SAC-A.05	Staffing and Qualifications The objective of this review is to verify that the applicant has analyzed the requirements for the number and qualifications of personnel in a systematic manner that includes a thorough understanding of task requirements and applicable regulatory requirements. The applicant's staffing and qualifications analyses should be evaluated in accordance with the review criteria of **NUREG-0711** and should satisfy the requirements of **10 CFR 50.54 (i)** through (m). If an exemption from these requirements is being sought, the analysis and justifications should be presented [see also **NUREG/CR-6838**, "Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)" and **NUREG-1791**, "Guidelines for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operating Staff Requirements Specified in 10 CFR 50.54(m) — Final Report"]. The full staffing program is considered to be an operational program as discussed in **SECY-05-197** and in **RG-1.206** "Combined License Applications for Nuclear Power Plants (LWR Edition)", Section C.IV.4, "Operational Programs."	Y (Per AREVA Topical Report ANP- 10279)	18.5
		N/A-COL	18.5
18.0-SAC-A.06	Human Reliability Analysis Human reliability analysis (HRA) is an evaluation of the potential for and	Y (Per AREVA Topical	18.6

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	mechanisms of human error that may affect plant safety. The objectives of this review are to ensure that (1) the applicant has addressed human-error mechanisms in the design of the HFE aspects of the plant to minimize the likelihood of personnel error, and detect errors and recover from them; and (2) the HRA activity effectively integrates the HFE program and PRA. A design-specific PRA/HRA is required by **10 CFR 50.34(f)(1)(i)**, **10 CFR 52.47(b)(1)** and **10 CFR 52.79**, and is addressed in **SRP Chapter 19** and **RG 1.206 Section C.II.1**. RG 1.206 Section C.II.1 specifies the purpose and objectives of the PRA, as well as the required scope and level of detail. In order to accomplish the above objectives, the HRA/PRA and the modeling of HAs must be of sufficient quality (see SRP Chapter 19 and RG 1.206 Section C.II.1**)**. Review of the HRA should be coordinated with **SRP Section 6.3**, III.19 and **RG 1.206 Section C.I.6.3.2.8** as they relate to manual actions for ECCS. The integration of the applicant's HRA with the HFE program should be evaluated in accordance with the review criteria of **NUREG-0711**.	Report ANP- 10279)	
18.0-SAC-A.07	Human-System Interface Design The HSI design process represents the translation of function and task requirements into HSI characteristics and functions. The objective of this review is to evaluate the process by which HSI design requirements are developed and HSI designs are identified and refined. The review should verify that the applicant has appropriately translated functional and task requirements to the detailed design of alarms, displays, controls, and other aspects of the HSI through the systematic application of HFE principles and criteria. The applicant's HSI design process should be	Y	18.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	evaluated in accordance with the review criteria of **NUREG-0711**, and the final design evaluated in accordance with the review criteria of **NUREG-0700**, “Human-System Interface Design Review Guidelines.” The HSI design should address those subsections of **10 CFR 50.34(f)(2)** that are applicable to the plant's design from the following list: **10 CFR 50.34(f)(2)(i), (iii), (iv), (v), (xi), (xii), (xiii), (xv), (xvii), (xviii), (xix), (xxi), (xxiv), (xxv), & (xxvii)**.		
	In addition to the HFE considerations discussed above, the following specific HSI design guidance should also be addressed: 1. Safety parameter display system requirements, as described in **10 CFR 50.34(f)(2)(iv)**, **NUREG-0835**, **NUREG-1342**, and **Supplement 1 of NUREG-0737**.	Y	18.7
	2. Periodic testing of protection systems actuation functions, as described in **Regulatory Guide 1.22**.	Y	18.7
	3. Bypassed and inoperable status indication for NPP safety systems, as described in **Regulatory Guide 1.47**.	Y	18.7
	4. Manual initiation of protective actions, as described in **Regulatory Guide 1.62**.	Y	18.7
	5. Instrumentation for light-water-cooled nuclear power plants to access plant and environmental conditions during and following an accident, as described in **Regulatory Guide 1.97**.	Y	18.7
	6. Instrumentation setpoints, as described in **Regulatory Guide 1.105**.	Y (Per AREVA Topical Report ANP- 10279)	18.7

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	7. Functional criteria for emergency response facilities, as described in **NUREG-0696**.	Y	18.7
		N/A-COL	18.7
	8. A minimum inventory of controls, displays and alarms.	Y	18.7
	The HSI design should describe the process, after the plant is in operation, by which (1) HSIs are modified and updated,(2) temporary HSI changes are made (such as set point modification) and (3) operator defined HSIs are created (such as temporary displays defined by operators for monitoring a specific situation). The HSI design review should be coordinated with the instrumentation and controls review in **SRP Chapter 7**.	Y	18.7
		N/A-COL	18.7
18.0-SAC-A.08	**Procedure Development** The objective of this review is to confirm that the applicant's procedure development program incorporates HFE principles and criteria, along with all other design requirements, to develop procedures that are technically accurate, comprehensive, explicit, easy to utilize, validated, and in conformance with **10 CFR 50.34(f)(2)(ii)**. Because procedures are considered an essential component of the HFE design, they should be derived from the same design process and analyses as the other components of the HSI (e.g., displays, controls, operator aids) and subject to the same evaluation processes. The applicant's procedure development program should be evaluated in accordance with the review criteria of **NUREG-0711**. The review should be coordinated with the review of procedures described in **SRP Section 13.5**. The full procedures program is considered to be an operational program as discussed in **SECY-05-197** and in **RG-1.206 Section C.IV.4**	Y (Per AREVA Topical Report ANP- 10279)	18.8
		N/A-COL	18.8

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18
Human Factors Engineering

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
18.0-SAC-A.09	**Training Program Development** The objective of this review is to ensure that the applicant has a systematic approach for the development of personnel training. The training development should include the following five activities: • A systematic analysis of tasks and jobs to be performed • Development of learning objectives derived from an analysis of desired • performance following training • Design and implementation of training based on the learning objectives • Evaluation of trainee mastery of the objectives during training • Evaluation and revision of the training based on the performance of trained personnel in the job setting The training program should be developed in accordance with **10 CFR 50.120**, **10 CFR 52.79**, and **10 CFR Part 55** to ensure that personnel's qualifications are commensurate with the performance requirements of their jobs. The applicant's training program should be evaluated in accordance with the review criteria of **NUREG-0711** and should address applicable guidance provided in **SRP Section 13.2**, "Training." The full training program is considered to be an operational program as discussed in **SECY-05-197** and in **RG-1.206 Section C.IV.4**.	Y (Per AREVA Topical Report ANP- 10279) N/A-COL	18.9 18.9
18.0-SAC-A.10	**Verification and Validation** Verification and validation (V&V) evaluations seek to comprehensively determine that the design conforms to HFE design principles and that it enables plant personnel to successfully perform their tasks to achieve	Y (Per AREVA Topical Report ANP- 10279)	18.10

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	plant safety and other operational goals. The overall scope for V&V should include the main control room, the remote shutdown panel, and local control stations (including the central alarm system (CAS) and secondary alarm system (SAS) associated with the risk important HAs. The applicant's V&V activities include operational condition sampling, design verification, integrated system validation, and human engineering discrepancy (HED) resolution. The objectives of the staff review of each of these activities are identified in the following subsections.		
	1. **Operational Conditions Sampling** The applicant's sampling methodology identifies the range of operational conditions that guide V&V activities. The objectives of the review are to ensure that the applicant has identified a sample of operational conditions that (1) includes conditions that are representative of the range of events that could be encountered during operation of the plant, (2) reflects the characteristics that are expected to contribute to system performance variation, and (3) considers the safety significance of HSI components. The use of risk importance to help select failure events, transients, and accidents for use in V&V is appropriate. The applicant's operational conditions sampling should be evaluated in accordance with the review criteria of **NUREG-0711**.	Y	18.10
	2. **Design Verification** The applicant's verification should demonstrate that the design meets task and human requirements. Verification activities require a characterization of the HSI. The staff's review of design verification has the following objectives:	Y (Per AREVA Topical Report ANP- 10279)	18.10

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	• Inventory and Characterization Review - The objective of this review is to evaluate whether the applicant's HSI inventory and characterization accurately describes all HSI displays, controls, and related equipment that are within the defined scope of the HSI design review. • HSI Task Support Verification Review - The objective of this review is to evaluate whether the applicant verifies that the HSI provides all alarms, information, and control capabilities required for personnel tasks. • HFE Design Verification Review - The objective of this review is to evaluate whether the applicant verifies that the characteristics of the HSI and the environment in which it is used conform to HFE guidelines. The applicant's design verification should be evaluated in accordance with the review criteria of **NUREG-0711**.		
	3. **Integrated System Validation** The objective of integrated system validation is to confirm that the integrated system design (i.e., hardware, software, and personnel elements) acceptably supports safe operation of the plant. Validation is based on performance-based tests. The applicant's integrated system validation should be evaluated in accordance with the review criteria of **NUREG-0711**.	Y (Per AREVA Topical Report ANP- 10279)	18.10
	4. **Human Engineering Discrepancy (HED) Resolution** HED resolution is the process of evaluating and resolving issues that are identified in V&V evaluations. The objectives of the staff's review are to verify that the applicant's HED evaluation acceptably prioritizes	Y (Per AREVA Topical Report ANP- 10279)	18.10

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	HEDs in terms of their need for improvement and that design solutions and a realistic schedule for implementation is developed to address those HEDs selected for correction. The applicant's HED resolution should be evaluated in accordance with the review criteria of **NUREG-0711**.		
18.0-SAC-A.11	**Design Implementation** The objective of this review is to verify that the applicant’s as-built design will conform to the verified and validated design that resulted from the HFE design process. The applicant's design implementation process should be evaluated in accordance with the review criteria of **NUREG-0711**. This review should also ensure the acceptability of the applicant’s plans for determining the operability of the MCR, RSP, LCSs, Technical Support Center and Emergency Operations Facility.	Y (Per AREVA Topical Report ANP- 10279)	18.11
18.0-SAC-A.12	**Human Performance Monitoring** The objective of this review is to assure that the applicant has prepared a human performance monitoring strategy for ensuring that no significant safety degradation occurs because of any changes that are made in the plant and to verify that the conclusions that have been drawn from the evaluation remain valid over the life of the plant. The applicant’s performance monitoring strategy should be evaluated in accordance with the review criteria of **NUREG-0711**.	Y (Per AREVA Topical Report ANP- 10279)	18.12
		N/A-COL	18.12
18.0-SAC-B	**Review of the HFE Aspects of Control Room Modifications** License amendments involving major changes to the HSIs, such as control room modernization, should be reviewed using the guidance contained in Section II.A of this SRP chapter. However, since the extent of such modifications can vary, the staff's review should be tailored using	N/A-COL	N/A

Table 1-2 U.S. EPR Conformance with Standard Review Plan (NUREG-0800)

CHAPTER 18 Human Factors Engineering			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	the additional guidance from **NUREG-0711** and presented in this section.		
18.0-SAC-C	**Review of the HFE Aspects of Modifications Affecting Risk-Important Human Actions** The staff's review of license amendments and actions involving plant changes that affect important human actions (HAs) use a graded, risk-informed approach in conformance with **Regulatory Guide (RG) 1.174**. The staff's review uses a two-phase approach. The first phase is a screening analysis to determine the risk associated with the plant modification and its associated HAs using both quantitative and qualitative information (see Section C.1 below). This approach can be accomplished for submittals by licensees that are either risk-informed or non-risk-informed. Plan modifications and HAs are categorized into regions of high, medium, and lower risk. This categorization is used to determine the level of HFE review needed. The second phase of the review is performed by the human factors analyst and consists of the HFE review. Changes that involve more risk-significant HAs receive a detailed review (see Section C.2.1 below), while those of moderate risk significance receive a less detailed review (see Section C.2.2 below). HAs in the lowest risk region receive minimal HFE review (see Section C.2.3 below).	N/A-COL	N/A

CHAPTER 19
Severe Accidents

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
SRP 19.0	**Probabilistic Risk Assessment and Severe Accident Evaluation (R2, 06/2007)**		
	For a DC		
19.0-AC-01	**10 CFR 52.47(8) -** The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), specifically **10 CFR 50.34(f)(1)(i)**.	Y	6.2.5 19.1 19.2.3 19.2.4
19.0-AC-02	**10 CFR 52.47(a)(23) -** For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment.	Y	6.2.5 19.2.2 19.2.3
19.0-AC-03	**10 CFR 52.47(a)(27)** - A description of the design-specific probabilistic risk assessment (PRA) and its results.	Y	19.0 19.1 19.2
	For a COL		
19.0-AC-04	**10 CFR 52.79(a)(17) -** Information with respect to compliance with a number of the technically relevant positions of the Three Mile Island requirements in 10 CFR 50.34(f), specifically **10 CFR 50.34(f)(1)(i)**.	N/A-COL	N/A
19.0-AC-05	**10 CFR 52.79(a)(38) -** For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection,	N/A-COL	N/A

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Severe Accidents

SRP Criterion	**Description** **(AC – Acceptance Criteria Requirement,** **SAC – Specific SRP Acceptance Criteria)**	U.S. EPR Assessment	FSAR Section(s)
	hydrogen combustion, and containment bypass.		
19.0-AC-06	**10 CFR 52.79(a)(46)** - A description of the plant-specific PRA and its results.	N/A-COL	N/A
19.0-AC-07	**10 CFR 52.79(c)(1), (d)(1), and (e)(1)** - If a COL application references a standard design approval, standard DC, or the use of one or more manufactured nuclear power reactors licensed under **Subpart F of 10 CFR Part 52**, then, the plant-specific PRA information must use the PRA information for the design approval, design certification, or manufactured reactor, respectively, and must be updated to account for site-specific design information and any design changes or departures.	N/A-COL	N/A
19.0-SAC-01	**NRC Policy Statement**, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," **50 FR 32138**, August 8, 1985.	Y	19.2.1
			19.2.2
			19.2.3
19.0-SAC-02	**NRC Policy Statement**, "Safety Goals for the Operations of Nuclear Power Plants," **51 FR 28044**, August 4, 1986.	Y	19.0
			19.1.1
19.0-SAC-03	**NRC Policy Statement**, "Nuclear Power Plant Standardization," **52 FR 34884**, September 15, 1987.	Y	Throughout Chapter 19
19.0-SAC-04	**NRC Policy Statement**, "Regulation of Advanced Nuclear Power Plants," **59 FR 35461**, July 12, 1994.	Y	Throughout Chapter 19
19.0-SAC-05	**NRC Policy Statement**, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," **60 FR 42622**, August 16, 1995.	Y	Throughout Chapter 19
19.0-SAC-06	**SECY-90-016**, "Evolutionary Light-Water Reactor (LWR) Certification	Y	6.2.5

CHAPTER 19
Severe Accidents

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	Issues and Their Relationship to Current Regulatory Requirements," ADAMS Accession No. ML003707849, January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, June 26, 1990.		Throughout Chapter 19
19.0-SAC-07	**SECY-93-087**, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, July 21, 1993.	Y	6.2.5 Throughout Chapter 19
19.0-SAC-08	**SECY-96-128**, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708224, June 12, 1996, and the related SRM, ADAMS Accession No. ML003708192, January 15, 1997.	N/A-VEN	N/A
19.0-SAC-09	**SECY-97-044**, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708316, February 18, 1997, and the related SRM, ADAMS Accession No. ML003708232, June 30, 1997.	N/A-VEN	N/A

CHAPTER 19
Severe Accidents

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	The first five NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur. For the first aspect of the review, the staff's acceptance criteria consists of a determination that the applicant has adequately demonstrated that the design properly balances preventive and mitigative features and represents a reduction in risk when compared to existing operating plants. For the second aspect of the review, the staff should ensure that the applicant has used the PRA results and insights, including those from uncertainty analyses, importance analyses, and sensitivity studies, in an integrated fashion to identify and establish specifications and performance objectives (e.g., ITAAC, technical specifications, RAP, RTNSS, and COL action items) for the design, construction, testing, inspection, and operation of the plant. The specific programs establish the staff's acceptance criteria. For example, Section C.I.17.4 of Regulatory Guide 1.206 presents the RAP submittal guidance and SRP Section 17.4 gives the associated staff review guidance, including acceptance criteria.	N/A-INFO	N/A

CHAPTER 19
Severe Accidents

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	For designs that have evolved from current plant technology through the incorporation of several features intended to make the plant safer, more available, and easier to operate, the results of the PRA should indicate that the design represents a reduction in risk compared to existing operating plants. The staff review should include a broad (qualitative and quantitative) comparison of risks, by initiating event category, between the proposed design and existing operating plant designs (from which the proposed design evolved) to identify the major design features that contribute to the reduced risk of the proposed design compared to existing plant designs (e.g., passive systems, less reliance on offsite and onsite power for accident mitigation, and divisional separation) The staff review should also consider the impact of data uncertainties on the risk estimates. The uncertainty analysis should identify major contributors to the uncertainty associated with the estimated risks. In addition, the staff review should address the applicant's risk importance studies that are performed at the system, train, and component level to provide insights about (1) the systems that contribute the most in achieving the low risk level assessed in the PRA, (2) events (e.g., component failures or human errors) that contribute the most to decreases in the built-in plant safety level, and (3) events that contribute the most to the assessed risk. The staff should also review the applicant's sensitivity studies performed to gain insights about the impact of uncertainties (and the potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies should include (1) determining the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities, (2) determining the impact of the potential lack of modeling details on the estimated risk, and (3) determining the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). For designs using passive safety systems and active defense-in-depth systems, the staff should review the sensitivity studies performed to	Y	19.1.4.2 Throughout Chapter 19

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Severe Accidents

SRP Criterion	Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)	U.S. EPR Assessment	FSAR Section(s)
	To have confidence that the applicant’s PRA and severe accident evaluation results and insights are adequate, the PRA staff must also determine that the scope, level of detail, and technical adequacy of the design-specific and plant-specific PRA are appropriate for the DC and COL, respectively, and any identified uses and risk-informed applications, as follows:	Y	Throughout Chapter 19
	1. The applicant’s analyses should be comprehensive in scope, and address all applicable internal and external events and all plant operating modes. Since some aspects of the applicant’s approach may involve non-PRA techniques to address specific events (e.g., PRA-based seismic margins), the PRA staff review should ensure that the scope of the applicant’s analyses is appropriate for their identified uses and applications, which may involve a scope, level of detail, and/or technical adequacy for the affected areas that is greater than that needed for a COL application.	Y	Chapter 19
	2. The level of detail of the applicant’s PRA should be commensurate with the identified uses and applications of the PRA (e.g., sufficient to gain risk-informed insights and use such insights, in conjunction with assumptions made in the PRA, to identify and support requirements important to the design and plant operation). The PRA should reasonably reflect the actual plant design, construction, operational practices, and relevant operational experience of the applicant and the industry. **The burden is on the applicant to justify that the PRA approach, methods, and data, as well as the requisite level of detail necessary for the NRC staff’s review and assessment, are appropriate. Regulatory Guides 1.174 and 1.200 provide additional guidance on the level of detail that should be included**	Y	Throughout Chapter 19

CHAPTER 19 Severe Accidents			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	in the PRA. If detailed design information (e.g., regarding cable and pipe routing) is not available or if it can be shown that detailed modeling does not provide significant additional information, it is acceptable to make bounding-type assumptions consistent with the guidelines in Regulatory Guide 1.200. However, the risk models should still be able to identify vulnerabilities as well as design and operational requirements such as ITAAC and COL action items. In addition, the bounding assumptions should not mask any risk-significant information about the design and its operation.		
	3. Consistent with the guidance in Section 2.5 of Regulatory Guide 1.174 regarding QA, the staff expects that the applicant will have subjected its PRA to quality control. The following methods are acceptable to the NRC staff to ensure that the pertinent QA requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is sufficient: A. Use of personnel qualified for the analysis B. Use of procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses C. Documentation and maintenance of records, including archival documentation as well as submittal documentation D. Use of procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used previously are changed or determined to be in error	Y	Throughout Chapter 19

CHAPTER 19 Severe Accidents			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Toward this end, **the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the NRC (e.g., Regulatory Guide 1.200 and SRP Section 19.1).** In addressing the technical adequacy of the PRA, the applicant should include (1) a discussion of prior NRC staff review of the PRA (e.g., during the DC process), findings (i.e., facts and observations) from that review, disposition of those findings, and the relevance of that review to the technical adequacy of the current plant-specific PRA, (2) a discussion of the scope, level of detail, and technical adequacy needed to support the specific uses and risk-informed applications, (3) a discussion regarding the method used for determination of technical adequacy for pertinent PRA scope areas for which the NRC has not endorsed PRA standards (i.e., identify the guidance and good practices documents relied upon to determine the technical adequacy of the PRA), (4) a discussion on the use of and criteria for independent peer reviews, and (5) a discussion on the process for disposition independent peer review findings and maintaining or upgrading the PRA, as appropriate, to ensure that it reasonably reflects the as-designed, as-built, and as-operated plant, including the corrective action and feedback mechanisms involving the periodic evaluation of the PRA, consistent with its uses and risk-informed applications, on the basis of actual plant-specific equipment, train, and system performance and relevant industry operational experience. As noted in Element 1.1 of Table A-1 in Appendix A to Regulatory		

CHAPTER 19 Severe Accidents			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
	Guide 1.200, special emphasis should be placed on PRA modeling of novel and passive features in the design, as well as addressing issues related to those features, such as digital instrumentation and control, explosive (squib) valves, and the issue of T-H uncertainties. The staff should confirm that the assumptions made in the applicant's PRA during design development/certification, in which a specific site may not have been identified or all aspects of the design (e.g., balance of plant) may not have been fully developed, are identified in the DC application and either remain valid or are adequately addressed within the COL application. In addition, a DC and COL applicant may request NRC approval to implement one or more risk-informed applications. The applicant's submission and staff review of these risk-informed applications follow specific regulatory guidance, approved topical reports, and SRP sections. For example, if an applicant requests to implement a risk-informed inservice inspection program concurrent with its COL application, the application should address the guidance provided in Regulatory Guide 1.178, following a specific methodology contained in identified approved topical reports and approved industry code cases, and SRP Section 3.9.8 will guide the staff review of this program. Chapter 19 of the applicant's FSAR should identify this risk-informed application, with a cross-reference to Section 3.9.8 of the applicant's FSAR, which should describe the applicant's risk-informed inservice inspection program.		
SRP 19.1	**Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (R2, 06/2007)**		

CHAPTER 19 Severe Accidents			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
19.1-AC	Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations. The underlying regulations are identified in the SRP sections which invoke this SRP section.	Y	6.2.5 Throughout Chapter 19
19.1-SAC	Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.	Y	6.2.5 Throughout Chapter 19
	In order for the NRC staff to conclude that a PRA is of sufficient technical adequacy to support an application, the staff needs to be assured that (1) the parts of the PRA needed to support the application have been appropriately identified and (2) those parts have been performed in a manner consistent with current good PRA practice. The former needs to be addressed as part of the assessment of the application. The latter can be met by determining that the necessary parts of the PRA have been performed in accordance with the staff position on consensus PRA standards or industry programs as documented in the appendices to Regulatory Guide 1.200. Where there are differences in approach to performing a specific part, the staff must determine that the approach used by the applicant is either equivalent to, or better than, that supported by the staff position.	Y	6.2.5 Throughout Chapter 19

CHAPTER 19 Severe Accidents			
SRP Criterion	**Description (AC – Acceptance Criteria Requirement, SAC – Specific SRP Acceptance Criteria)**	**U.S. EPR Assessment**	**FSAR Section(s)**
SRP 19.2	**Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance (06/2007)**	N/A-COL	N/A