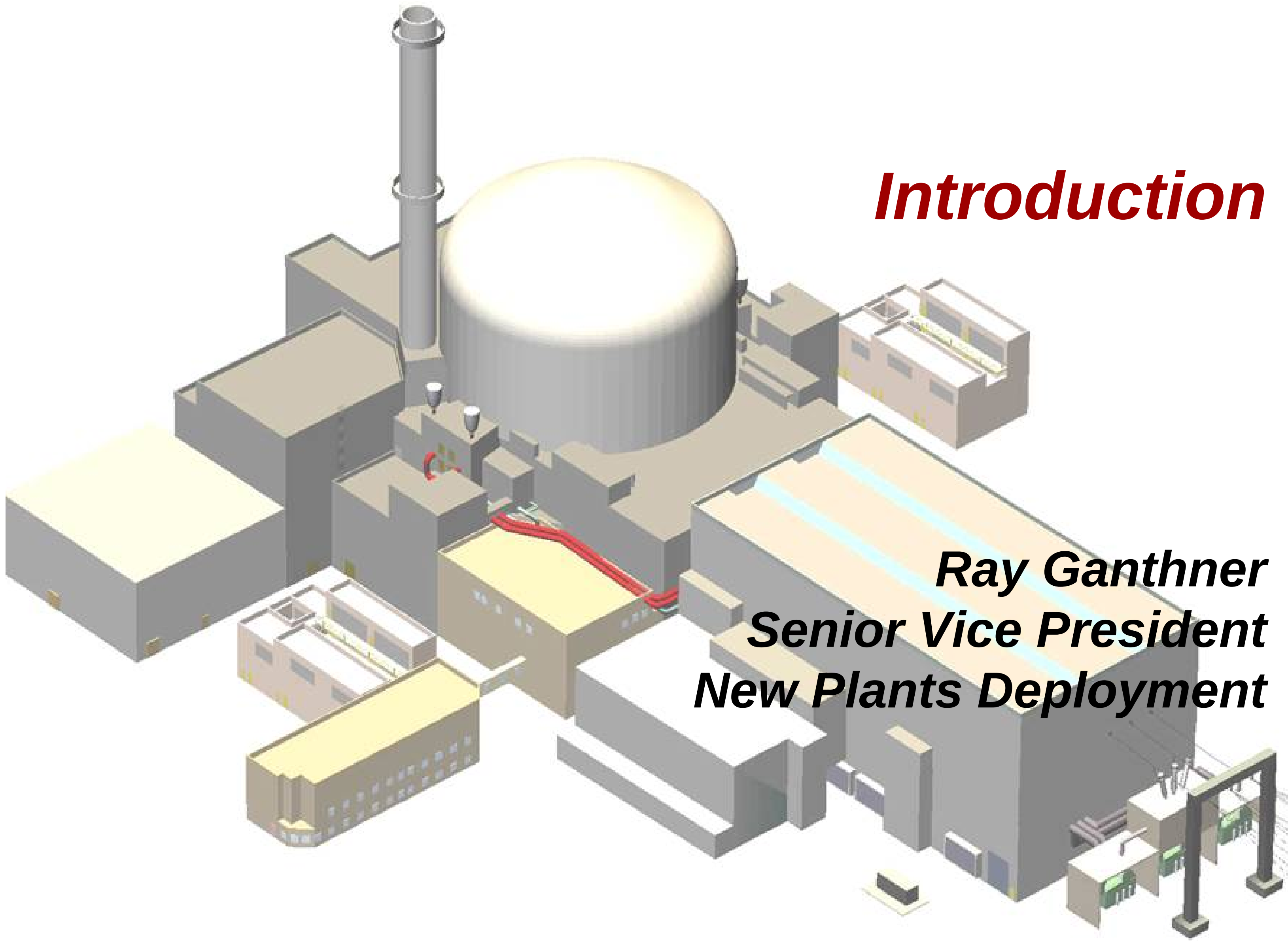
U.S. EPR Pre-Application Review Meeting: Overview of Regulatory Compliance Demonstration Process

AREVA NP Inc., Constellation Energy and the NRC
March 29, 2006

Introduction



Ray Ganthner
Senior Vice President
New Plants Deployment

Meeting Objectives

> To provide an update on global EPR-related activities

> To describe the process to demonstrate regulatory compliance for the U.S. EPR design certification application

> Obtain feedback from NRC on list of items for compliance evaluation

Outline

> **Overview of Regulatory Compliance Demonstration Process (Jim Kay)**

> **Summary and Next Steps (Sandra Sloan)**

Update on the Status of the EPR

> U.S. design conversion process proceeding according to schedule

> Former DOE Secretary Abraham appointed Chairman of the Board of AREVA Inc.

> UniStar Advisory Board announced

- John Gordon
- James Asselstine
- Richard Meserve
- Kyle McSlarrow
- Neil Todreas

> Framatome ANP, Inc. is now AREVA NP Inc.

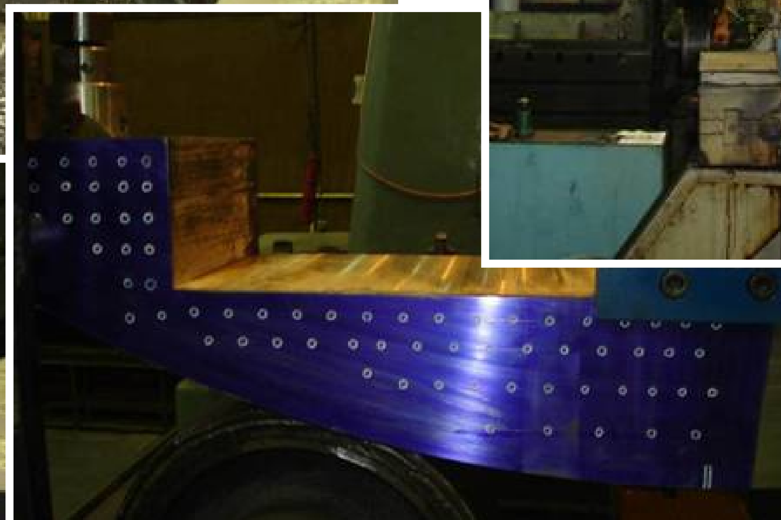
> International EPR activities continue (Finland, France, Canada, China)

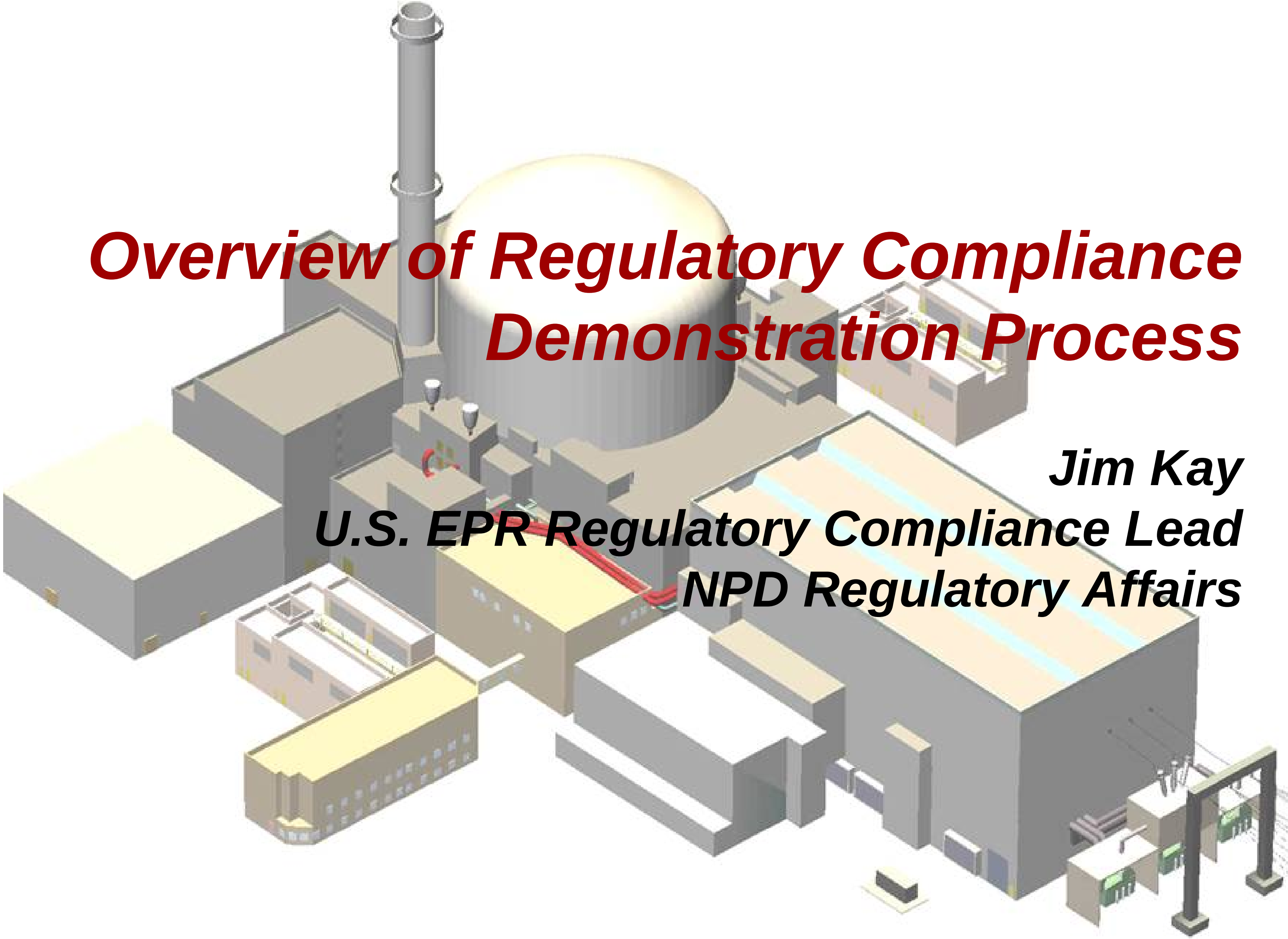
Olkiluoto 3: End of February 2006

General Overview of the Site: Turbine Island

General Overview of the Site: Water Inlet

RPV Internals: Heavy Reflector (SKODA)





Overview of Regulatory Compliance Demonstration Process

Jim Kay
U.S. EPR Regulatory Compliance Lead
NPD Regulatory Affairs

Overview

> Describe the process for identifying relevant items and demonstrating compliance

> Discuss the database for tracking compliance

> Describe the format & content of DCD Tier 2 Section 1.9

Basis of Process

> Identification of regulatory requirements & guidelines

> Incorporation of lessons learned from previous DC reviews (overall content, attention to detail, level of detail)
- RAIs
- NRC comment letters to applicants

> Develop compliance tracking database

Thorough, methodical process being applied to identify and address regulatory requirements and guidance.

Regulatory Compliance Requirements

> **Technical information required for CPs and OLs by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, which is technically relevant to the design and not site-specific *[10 CFR 52.47(a)(1)(i)]***

> **Demonstration of compliance with technically relevant portions of the TMI requirements in 10 CFR 50.34(f) except paragraphs (f)(1)(xii), (f)(2)(ix) and (f)(3)(v) *[10 CFR 52.47(a)(1)(ii)]***

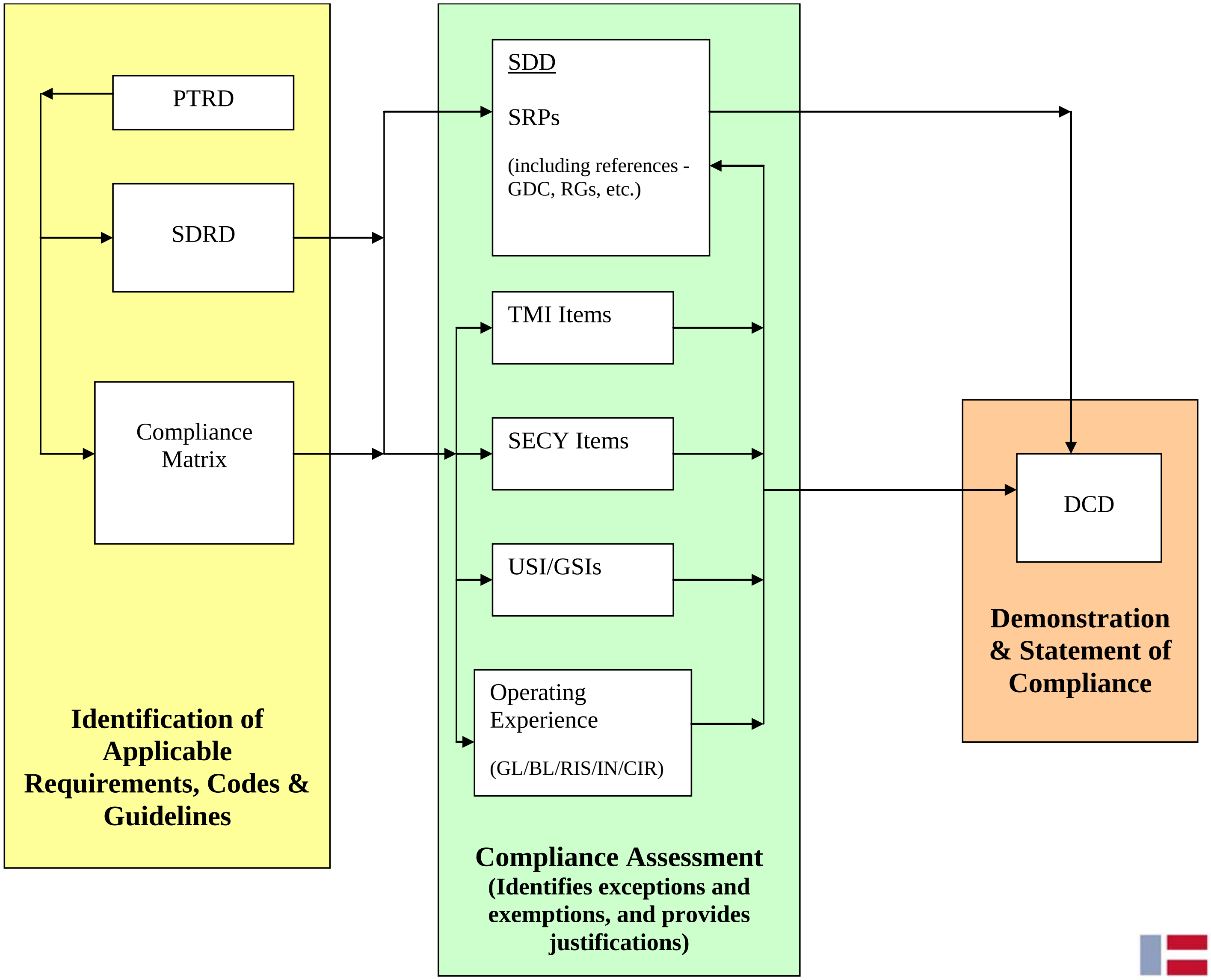
> **Proposed resolutions of USIs and medium- and high-priority GSIs in NUREG-0933 current 6 months prior to application which are technically relevant to the design (currently NUREG-0933 Supplement 29, published November 2005) *[10 CFR 52.47(a)(1)(iv)]***

> **Evaluation of the facility against the SRP in effect 6 months prior to the docket date of the application *[10 CFR 50.34(h)]***

Additional Technical Information

> **Evaluation of operating experience *[SRMs dated July 31, 1989, and February 15 and March 5, 1991]***
  - **Generic Letters and Bulletins issued between January 1, 1980, and 6 months prior to DCD submittal**

Compliance Demonstration Process



Process Integration

- Plant Technical Requirements Document (PTRD)
  - Top level U.S. EPR design requirements
  - Top level regulatory requirements
  - Design codes and standards
- System Design Requirements Document (SDRD)
  - System design requirements
- Compliance Matrix
  - List of applicable criteria for design certification
- System Description Document (SDD)
  - Defines the system design to permit verification that the design satisfies the design input
  - Provides the compliance assessment (input for DCD)
- DCD Control Procedure
- DCD Preparation Guideline
  - Guidance to develop Tier 1 & Tier 2 information (RG 1.70 format & SRP content conformance)

Regulatory Compliance Inputs

> Standard Review Plan (SRP)
- Using current drafts dated 1996
- Referenced regulations, General Design Criteria, and Regulatory Guides
- Following ongoing updates to SRP

> TMI Items (10CFR50.34)
- 32 out of 41 evaluated for U.S. EPR DC

> SECY Issues (90-016, 93-087)
- 38 out of 42 evaluated for U.S. EPR DC

> USI/GSIs (NUREG-0933)
- 94 issues evaluated for U.S. EPR DC
- 4 High (#156.6.1, #163, #185, #191)
- 1 Medium (#89 – for future plants only)

> OE Information (BLs, GLs, INs, CIRs, RISs)
- 1980 through present

> Consideration of SECY 05-0006

U.S. EPR will comply with U.S. regulatory requirements

Compliance Template

- List of Positions/Requirements, U. S. EPR Compliance, Justification & Supporting Statement and/or Comments

Position/Requirement	EPR Compliance (see Note 1)	Basis (include supporting statements/comments)

Note (1):

Does the design comply with the position? YES, NO, OPEN.

a. YES means all information is complete to make the determination, and the design fully meets the entire position. Explain the basis for the conclusion.
b. NO means all information is complete to make the determination, and the design does not meet the position. Explain the basis for the conclusion.
c. OPEN means that the design information is not available yet or a conclusion can not be made at this time. These items will be tracked to resolution.

If all regulatory positions are YES, the design is in full compliance.

If there is a NO or OPEN, an exception or exemption to the requirement/position might be necessary, with justification.

All items will be resolved by the time of the submittal of the DC application.

Regulatory Compliance Tracking

> Database developed
> Tracking compliance inputs (SECY, TMI, OE, RG, USI/GSI)
> Use to develop Tier 2 Section 1.9 of the DCD
> Shows relationship between DCD sections & source material (e.g., SDDs)

Compliance Database

EPR Complete List of Compliance Issues

Tuesday, March 21, 2006
9:10:40 AM

Issue	Issue No	Issue Description	Contact	Position No and Description	Assigned To	Due Date	DCD Section Affected	System Code	Source Documents	Program	Category
50.34(f)	(2)(xii)	AFW System Initiation and Indication	Reeside	1 Provide automatic and manual AFW system initiation and provide AFW flow indication in the control room.	Jones Meacham McCumber	1/7/2021	7.7; 10.4.9	LAR			Undetermined
50.34(f)	(2)(xiii)	Pressurizer Heater Power Supplies	Reeside	1 Provide PZR heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available.	Grundman	2/1/2007	5.4.5; 8	JEF			Undetermined
50.34(f)	(2)(xiv)	Containment Isolation System	Reeside	1 Provide containment isolation systems that: A) Ensure all non essential systems are isolated automatically by the containment isolation system. B) For each non-essential penetration (except instrument lines), have two isolation barriers in series. C) Do not result in re opening of the containment isolation valves on the resetting of the isolation signal. D) Utilize a containment set point pressure for initiating containment isolation as lo[illegible] compatible with normal operati[illegible] E) Include automatic clo[illegible] on [illegible]igh rad[illegible]ion signal for all systems that [illegible]vide [illegible] to the environs.	Stoudt Meacham Getz Orlakis	6/1/2007	6.2.3	[illegible]			Undetermined
50.34(f)	(2)(xix)	Post Accident Monitoring Instrumentation	Reeside	1 Prov[illegible] instrum[illegible]ta[illegible] a[illegible]equ[illegible]e for [illegible]onitoring plan[illegible] co[illegible]itions [illegible]llow[illegible] an ac[illegible]dent that includes core [illegible]ama[illegible]		3/1/2007	7; 19	CRU	SDD	EP	Compliant
50.34(f)	(2)(xv)	Containment Purging/Venting	Ree[illegible]	1 Prov[illegible] a capa[illegible]ity for containment purging/venting desi[illegible]ed to minimize the purging time consistent [illegible]ith ALARA principles for occupational exposures. Provide and demonstrate that the purge system will reliably isolate under accident conditions.	Orlakis Runowski Getz	12/1/2006	6	JMA30	SDD		Compliance Expected
50.34(f)	(2)(xvii)	Specific Accident Monitoring Instrumentation	Reeside	1 Provide instrumentation to measure, record and readout in the control room: containment pressure, containment water level, containment hydrogen concentration, containment radiation levels, and noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points and for onsite capability to analyze and measure these samples.	Wheeler Geiger Gruenewald Getz Jones Orlakis	12/1/2006	7	CRU	SDD	EP	Compliance Expected

Products of Compliance Assessment

- DCD Tier 2 Section 1.9 (conformance with regulatory criteria)
  - Regulatory Guides
  - Compliance with the SRP (NUREG-0800)
  - TMI Items
  - USI/GSIs
  - ALWR Certification Issues
  - Operating Experience
  - References

- Assessment results in appropriate DCD sections

Conclusions

- Identified applicable compliance requirements and guidance for design certification
- Using current draft SRP and associated RGs and will continue to follow the update process
- Developed integrated process to assess and demonstrate compliance
- Tracking compliance for input to DCD

Discussion of Specific Compliance Issues (See Handout)

> **Use of draft SRPs**
- **RG 1.97 Rev. 4 (vs. Rev. 3)**
- **SRP on Tornados (Rev. 3, dated January 2006)**
- **List of endorsed standards**

> **Confirmation of selection of USI/GSIs**
- **GIMCS, dated October 31, 2005**
- **Applicability of NUREG-0933, Appendix B**

Summary and Next Steps

Sandra M. Sloan
Manager, Regulatory Affairs
New Plants Deployment

Summary

> Approach for U.S. EPR regulatory compliance evaluation addresses 10CFR 52.47 requirements

> Sufficient information will be provided in the DCD, and supporting documentation, to demonstrate compliance

> Justification will be provided for exceptions and exemptions

Next Steps

> **Letter from NRC identifying list of relevant technical information, including applicable regulatory requirements, guidance documents and operating experience, for the U.S. EPR DC application by June 2006**

> **Next meeting**

- **April 2006, I&C RPS Design Philosophy and Concepts**

U.S. EPR Design Certification - Applicable Operating Experience

Bulletins

Bulletin	Title	Issue Addressed
80-04	Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition	1) Review the containment pressure response analysis to determine if the potential for containment overpressure for a main steam line break inside containment included the impact of runout flow from the auxiliary feedwater system and the impact of other energy sources, such as continuation of feedwater or condensate flow. 2) Review the analysis of the reactivity increase which results from a main steam line break inside or outside containment. This review should consider the reactor cooldown rate and the potential for the reactor to return to power with the most reactive control rod in the fully withdrawn position.
80-05	Vacuum Condition Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks	Review the design of all systems that contain low pressure or holdup tanks that can be valved to contain primary system water. Assure that adequate measures have been taken to protect against vacuum conditions that could result in tank damage with the potential for release of radioactive material or detrimental effects with regard to overall safety of plant operations.
80-06	Engineered Safety Feature (ESF) Reset Controls	Review the drawings for all systems serving safety-related functions at the schematic level to determine whether or not upon the reset of an ESF actuation signal, all associated safety-related equipment remains in its emergency mode.
80-08	Examination of Containment Liner Penetration Welds	Determine if your facility contains the flued head design for penetration connections, or other designs with containment boundary butt weld(s) between the penetration sleeve and process piping as illustrated in Figure NE 1120-1, Winter 1975 Addenda to the 1974 and later editions of the ASME B&PV Code.
80-12	Decay Heat Removal System Operability	Review the hardware capability of your facility to prevent DHR loss events, including equipment redundancy, diversity, power source reliability, instrumentation and control reliability, and overall reliability during the refueling and cold shutdown modes of operation.
80-18	Maintenance of Adequate Minimum Flow Thru Centrifugal Charging Pumps Following Secondary Side High Energy Line Rupture	Specific calculations outlined in the Bulletin were to be performed to determine if adequate minimum flow is assured under all conditions.
81-01 81-01 R1	Surveillance of Mechancial Snubbers	Several instances of failures of mechanical snubbers supplied by International Nuclear Safeguards Corporation (INC) have been identified
81-02	Failure of Gate Type Valves to Close Against Differential Pressure	The following valves failed to fully close during the U.S. EPRI PORV block valve testing: 1. Westinghouse Electro-Mechanical Division (W-EMD) 3-inch Valves 2. Borg-Warner Nuclear Valve Division (BW-NVD) 3-inch 1500-pound Motor- Operated Gate Valves 3. Anchor Darling 3-inch 1540-pound Double-Disc Valve
84-03	Refueling Cavity Water Seal	Requested that licensees evaluate the potential for and consequences of a refueling cavity water seal failure.
86-03	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	This bulletin informed addressees of single failures of minimum flow recirculation lines containing air-operated isolation valves which could result in a common-cause failure of all emergency core cooling system (ECCS) pumps in a system.
88-04	Potential Safety-Related Pump Loss	This bulletin addressed two miniflow design concerns. The first concerned the potential for the dead-heading of one or more pumps in

Bulletin	Title	Issue Addressed
		safety-related systems that have a miniflow line common to two or more pumps or other piping configurations that do not preclude pump-to-pump interaction during miniflow operation. A second concern was whether or not the installed miniflow capacity was adequate for even a single pump in operation.
88-08	Thermal Stresses in Piping Connected to Reactor Cooling Systems	This bulletin requested that licensees review their reactor coolant systems (RCSs) to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses.
88-11	Pressurizer Surge Line Thermal Stratification	This bulletin requested that addressees establish and implement a program to confirm pressurizer surge line integrity in view of the occurrence of thermal stratification.
01-01	Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles	This bulletin requested that addressees provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking.
02-01	Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	This bulletin required PWR addressees to submit information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head (follow-up to Bulletin 01-01).
02-02	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs	Follow-up to Bulletins 01-01 and 02-01
03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors	Notified licensees of the susceptibility of PWR recirc sump screens to debris blockage in the event of a high-energy line break requiring sump recirculation.
03-02	Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	Follow-up to Bulletins 01-01, 02-01, and 02-02.
04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors	Inspection of Alloy 82/182/600

Generic Letters

Generic Letter	Title	Issue Addressed
80-01	NUREG-0630 " Cladding, Swelling And Rupture - Models For LOCA Analysis"	Requested comments on NUREG-0630.
80-05	IEB 79-01b Environmental Qualification Of Class 1E Equipment	IE Bulletin No. 79-01 required licensees to perform a detailed review of the environmental qualification of Class IE electrical equipment to ensure that the equipment would function under (i.e. during and following) postulated accident conditions.
80-10	Issuance Of NUREG-0588, "Interim Staff Position On Equipment Qualifications Of Safety-Related Electrical Equipment	NUREG-0588 provides the NRC's technical positions on selected areas of environmental qualification.
80-11	IEB 80-03 Loss Of Charcoal From Standard Type II, 2 Inch, Tray Absorber Cells	
80-12	IEB 80-04 Analysis Of A PWR Main Steam Sine Break With Continued Feedwater Addition	

Generic Letter	Title	Issue Addressed
80-13	Qualification Of Safety Related Electrical Equipment	For each item of safety-related electrical equipment, licensees were requested to provide a tabular listing of all such equipment and appropriate qualification-related data.
80-14	LWR Primary Coolant System Pressure Isolation Valves	Describe the valve configuration at your plant and indicate if an Event V isolation valve configuration exists within the Class 1 boundary of the high pressure piping connecting RCS piping to low pressure system piping (1) two check valves in series, or (2) two check valves in series with an MOV.
80-16	IEB 79-01b Environmental Qualification Of Class 1E Equipment	Follow-up to GL 80-04 and -05
80-20	Actions Required From OL Applicants Of NSSS Designs By W and CE Resulting From NRC B&O Task Force Review Of TMI2 Accident	(a) provide an evaluation which shows how the AFW system meets each requirement in Standard Review Plan 10.4.9 and Branch Technical Position ASB-10-1, (b) perform a reliability evaluation similar to the method described in Enclosure 1, and (c) respond to Enclosure 2, which requests the information necessary to determine the design basis for your AFW system flow requirements and to verify that your AFW system will meet these requirements.
80-21	IEB 80-05 Vacuum Condition Resulting In Damage To Chemical Volume Control System Holdup Tanks	
80-25	IEB 80-06 Engineering Safety Feature (ESF) Reset Controls	
80-28	IEB 80-08 Examination Of Containment Liner Penetration Welds	
80-34	Clarification Of NRC Requirements For Emergency Response Facilities At Each Site	An onsite technical support center (TSC) shall be maintained by each operating nuclear power plant. The TSC shall be separate from, but in very close proximity to, the control room and be within the plant security boundary.
80-35	Effect Of A DC Power Supply Failure On ECCS Performances	It has generally been recognized that the loss of a direct current (DC) power supply could disable several emergency core cooling system components and thereby could result in a limiting single failure condition for some breaks.
80-37	Five Additional TMI-2 Related Requirements To Operating Reactors	(1) I.A.1.3-Shift Manning; (2) I.A.3.1 - Licensing Examinations: (3) I.C.5 - Licensee Dissemination of Operating Experiences; (4) II.K.3 - LOFW and Small Break LOCA Generic Review Matters; and (5) III.D.3.4 - Control Room Habitability.
80-45	Fire Protection Rule	Transmittal of proposed Fire Protection Rule.
80-46	Generic Technical Activity A-12 Fracture Toughness	Fracture toughness of RCS component supports
80-47	Additional Guidance On "Potential For Low Fracture Toughness And Laminar Tearing On PWR SG Reactor Coolant Pump Supports	Fracture toughness of RCS component supports
80-48	Revision To 5/19/80 Letter On Fire Protection	
80-52	Five Additional TMI-2 Related Requirements - Errata Sheets To 5/7/80 Letter	II.K.3.25, II.K.3.29 and II.K.3.44
80-69	IEB 80-18 Maintenance Of Adequate Minimum Flow Through Centrifugal Charging Pumps Following Secondary Side HELB	
80-73	"Functional Criteria For Emergency Response Facilities", NUREG-0696	Transmitted criteria for emergency response facilities
80-82	IEB 79-01b Supplement 2 Environmental Qualification Of Class 1E	EQ

Generic Letter	Title	Issue Addressed
	Equipment	
80-83	Environmental Qualification Of Safety-Related Equipment	EQ
80-85	Implementation Of Guidance From USI A-12 "Potential For LOW Fracture Toughness And Lamellar Tearing On Component Support	
80-88	Seismic Qualification Of Auxiliary Feedwater Systems	Requested review of seismic qualification of auxiliary feedwater system
80-89	IEB 79-01b Supplement 3 Environmental Qualification Of Class 1E Equipment	
80-96	Fire Protection	
80-102	Commission Memorandum And Order Of May 23, 1980 (Referencing IEB 79-01b Supplement 2 - q.2 & 3 - Sept 30, 1980	EQ
80-105	Implementation Of Guidance For USI A-12, Potential For Low Fracture Toughness And Lamellar Tearing On Component Supports	
80-113	Control Of Heavy Loads	Crane design, rigging and handling program.
81-04	Emergency Procedures And Training For Station Blackout Events	NRC request that you review your current plant operations to determine your capability to mitigate a station blackout event and promptly implement, as necessary, emergency procedures and a training program for station blackout events.
81-05	Information Regarding The Program For Environmental Qualification Of Safety-Related Electrical Equipment	EQ
81-07	Control of Heavy Loads	Crane design, rigging and handling program.
81-10	Post-TMI Requirements For The Emergency Operations Facility	1) Each operating nuclear plant shall maintain an onsite technical support center (TSC) separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. 2) An operational support center (OSC) shall be established separate from the control room and other emergency response facilities as a place where operations support personnel can assemble and report in an emergency situation to receive instructions from the operating staff. 3) Communications shall be provided between the OSC, TSC, EOF, and control room.
81-12	Fire Protection Rule (45 FR 76602, 11/19/80)	
81-14	Seismic Qualifications for Auxiliary Feedwater Systems	Follow-up to GL 80-88
81-19	Thermal Shock to Reactor Pressure Vessels	
81-21	Natural Circulation Cooldown	1) Review of procedures and training for natural circulation cooldown; 2) A demonstration (e.g. analysis and/or test) that controlled natural circulation cooldown from operating conditions to cold shutdown conditions, conducted in accordance with your procedure, should not result in reactor vessel voiding;

Generic Letter	Title	Issue Addressed
		3) Verification that supplies of condensate-grade auxiliary feedwater are sufficient to support your cooldown method;
81-28	Steam Generator Overfill	Operator training programs to address steam generator overfill
81-39	NRC Volume Reduction Policy	Notified licensees of NRC policy on solid waste volume reduction and waste minimization.
82-04	Use of INPO See-in Program	Item I.C.5 of the TMI Action Plan, requires licensees to develop procedures to assure that important operating experience originating both within and outside the utility organization is continually provided to operators and other personnel, and is incorporated into training and retraining programs.
82-09	Environmental Qualification of Safety Related Electrical Equipment	EQ
82-20	Guidance for Implementing the Standard Review Plan Rule	Implementation of NRC requirement to include an evaluation of the facility against the acceptance criteria of the Standard Review Plan (NUREG-0800).
82-26	NUREG-0744, REV. 1 - Pressure Vessel Material Fracture Toughness	Reactor vessel fracture toughness requirements.
82-33	Supplement 1 to NUREG-0737 - Emergency Response Capabilities	Additional guidance beyond 10CFR50.34(f) for Items I.C.1, I.D.1, I.D.2, III.A.1.2, III.A.2.2
83-10a 83-10b 83-10c 83-10d 83-10e 83-10f	Resolution of TMI Action Item II.K.3.5., "Automatic Trip of Reactor Coolant Pumps"	Addressed acceptability of CE, W, and B&W ECCS models and need for RCP trip
83-15	Implement. of Reg. Guide 1.150, "Ultrasonic Testing of RX Vessel Welds During Preservice & Inservice Examinations, REV.1	Transmitted Rev 1 of Reg Guide 1.150.
83-16 83-16a	Transmittal of NUREG-0977 Relative to the ATWS Events at Salem Generating Station, Unit No. 1	Transmittal of NUREG-0977
83-28	Required Actions Based on Generic Implications of Salem ATWS Events	Follow-up action for Salem ATWS
83-32	NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS	Recommended operator actions
83-33	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	Fire Protection
83-35	Clarification of TMI Action Plan Item II.K.3.31	NUREG-0737 Item II.K.3.30 requires that each licensee revise its current ECCS SBLOCA models.
83-42	Clarification to GL 81-07 Regarding Response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	Clarification with respect to single failure proof cranes.
84-01	NRC Use Of The Terms "Important To Safety" and "Safety Related"	Clarification of QA terms
84-04	Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04)	Provided that other PWR applicants may also request exemptions on the same basis from the requirements of GDC-4 with respect to asymmetric blowdown loads resulting from discrete breaks in the primary main coolant loop, if they can demonstrate … can provide an equivalent fracture mechanics based demonstration of the integrity of the primary main coolant loop in their facilities.
84-09	Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii)	Hydrogen control at Mark I BWRs with mention of PWRs.
84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	NRC information request : - Reduction of fast, cold starts - D/G reliability data

Generic Letter	Title	Issue Addressed
84-24	Certification of Compliance to 10CFR50.49, EQ of Electric Equipment Important to Safety for Nuclear Power Plants	EQ
85-05	Inadvertent Boron Dilution Events	Boron dilution mitigation and analysis required for new plants but not backfitted on operating plants
85-06	Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related	NRC guidance for non-safety-related ATWS equipment
85-11	Completion of Phase II of "Control Of Heavy Loads At Nuclear Power Plants" NUREG-0612	Heavy Load Handling
85-12	Implementation Of TMI Action Item II.K.3.5, "Automatic Trip Of Reactor Coolant Pumps	Addressed acceptability of manual RCP trip and W ECCS models.
85-16	High Boron Concentrations	NRC encouraged elimination of high concentration boron storage.
85-22	Potential For Loss Of Post-LOCA Recirculation Capability Due To Insulation Debris Blockage	Sump blockage potential concern.
86-07	Transmittal of NUREG-1190 Regarding the San Onofre Unit 1 Loss of Power And Water Hammer Event	Follow-up to San Onofre event
86-10	Implementation of Fire Protection Requirements	
86-10 S1	Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate Redundant Safe Shutdown Trains Within the Same Fire Area	
86-15	Info ... Compliance W 10CFR50.49 "EQ of Electric Equipment Important to Safety For Nuclear Power Plants"	EQ
87-02 87-02 S1	Verification of Seismic Adequacy of Mechanical and Electrical Equipment In Operating Reactors (USI A-46)	Review of seismic adequacy of older plants.
87-03	Verification of Seismic Adequacy of Mechanical and Electrical Equipment In Operating Reactors (USI A-46)	Review of seismic adequacy of older plants.
87-11	Relaxation in Arbitrary Intermediate Pipe Rupture Requirements	
87-12	Loss of Residual Heat Removal While The Reactor Coolant System is Partially Filled	Discussion of mid-loop operation.
88-03	Resolution of Generic Safety Issue 93, "Steam Binding of Auxiliary Feedwater Pumps"	The issue concerns the potential disabling of auxiliary feedwater pumps by steam binding that is caused by backleakage of main feedwater past the isolation check valves between the AFW and MFW systems. Most AFW systems are potentially vulnerable to common mode failure of the redundant AFW pumps as a result of steam binding. This vulnerability is inherent to the piping configurations used, which allow redundant trains of AFW to be cross-connected via common headers on the suction and discharge sides of the pumps.
88-07	Modified Enforcement Policy ... 10 CFR 50.49, "Environmental Qualification of Electrical Equipment Important to Safety"	EQ
88-11	NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations	Transmitted Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials,"
88-12	Removal of Fire Protection Requirements from Technical	Requested that licensees incorporate the NRC–approved Fire Protection Program into their FSAR.

Generic Letter	Title	Issue Addressed
	Specification	
88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment	Request that each licensee/applicant review NUREG-1275, Volume 2, and perform a design and operations verification of the instrument air system.
88-15	Electric Power Systems - Inadequate Control Over Design Process	Addressed a number of electrical system problems:(1) onsite distribution system voltages lower than required for proper operation of safety equipment, (2) diesel generator loads exceeding the diesel engine's load carrying capability, (3) diesel generator voltage regulating systems unable to maintain voltage at a sufficient level to permit continued operation of safety equipment, (4) overloading of 1E buses during a LOCA because of interaction of the fire suppression system and other safety-related systems, (5) lack of proper coordination of protective devices creating the potential for an unacceptable level of equipment loss during fault conditions, and (6) electrical distribution system components outside their design ratings for fault clearing capability creating the potential for an unacceptable level of equipment loss during fault conditions. These problems have occurred primarily as a result of inadequate control over the design process.
88-17	Loss of Decay Heat Removal	Requested responses to LDHR actions.
88-18	Plant Record Storage on Optical Disks	Guidance for storage of QA data on optical disk.
88-20	Individual Plant Examination for Severe Accident Vulnerabilities	Request to perform an Individual Plant Examination for severe accident vulnerabilities.
88-20 S1	Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54	
88-20 S2	Accident Management Stratagies for Consideration in the Individual Plant Examination Process	
88-20 S3	Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the IPE for SAV	
88-20 S4	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities	
88-20 S5	Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)	
89-04 89-04 S1	Guidance on Developing Acceptable Inservice Testing Programs	Approval of inservice pump and valve testing programs
89-06	Task Action Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54 (f)	
89-10	Safety-Related Motor-Operated Valve Testing and Surveillance	
89-19	Request for Actions Related to Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f)	The NRC concluded that all PWR plants should provide automatic steam generator overfill protection and that plant procedures and Technical Specifications should include provisions to verify periodically the operability of the overfill protection and to assure that automatic overfill protection is available to mitigate main feedwater overfeed events during reactor power operation. Also, the system design and setpoints should be selected with the objective of minimizing inadvertent trips of the main feedwater system during plant startup, normal operation, and protection system surveillance.

Generic Letter	Title	Issue Addressed
89-22	Potential for Increased Roof Loads...Due to Recent Change in Probable Maximum Precipitation Criteria...	Imposition of the latest NWS criteria call for higher rainfall intensities over shorter time intervals and smaller areas than have been previously considered. In some cases, such events could result in higher site flooding levels and greater roof ponding loads than have been used in previous design studies.
90-06	Resolution of Generic Issues 70, "PORV and Block Valve Reliability," and 94, "Additional LTOP Protection for PWRs"	
91-06	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies," Pursuant to 10 CFR 50.54(f)	Requested information on the DC power system
92-01 R1 92-01 R1 S1	Reactor Vessel Structural Integrity	Reactor Vessel material surveillance
95-07	Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves	
96-01	Testing of Safety-Related Circuits	Industry problems with safety related control logic testing
96-05	Periodic Verification of Design-Basis Capability of Safety-Related Power-Operated Valves	
96-06 96-06 S1	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	Potential for water hammer, two phase flow, and thermally induced overpressurization in cooling water systems serving containment coolers following a LOCA or MSLB.
97-01	Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations	
97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	NPSH analysis
98-02	Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition	NRC request that plants assess the susceptibility of their residual heat removal and emergency core cooling systems to common-cause failure as a result of reactor coolant system drain down while in a shutdown condition.
98-04	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	Sump blockage
03-01	Control Room Habitability	Control Room ventilation design and testing
04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors	Sump blockage
06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	Follow-up to Northeast Blackout

I. E. Circulars

Circular	Title	Issue Addressed
80-03	Protection from Toxic Gas Hazards	Toxic gas hazards
80-09	Problems with Plant Internal Communications Systems	1) Operability of internal communications systems during loss of offsite power or other foreseeable events and 2) Determine whether any plant electronic

Circular	Title	Issue Addressed
		equipment may be adversely affected by portable radio transmissions.
80-17	Fuel Pin Damage Due to Water Jet from Baffle Plate Corner	
81-09	Containment Effluent Water that Bypasses Radioactivity Monitor	All water system effluents that are not automatically isolated by a high-containment-pressure containment isolation signal and that flow directly to the environment from containment should be reviewed to determine whether or not a pathway exists for "significant" unmonitored discharge.
81-12	Inadequate Periodic Test Procedure of PWR Reactor Protection System	Adequacy of trip breaker testing.
81-14	Main Steam Isolation Valve Failures to Close	Numerous examples

Regulatory Issues Summaries

Regulatory Issue Summary	Title	Issue Addressed
00-24	Concerns about Offsite Power Voltage Inadequacies and Grid Reliability Challenges Due to Industry Deregulation	NRC concerns relating to the functional capability of the emergency core cooling system under degraded voltage conditions.
03-12	Clarification of NRC Guidance for Modifying Protective Actions	EQ qualification of low-voltage I&C cables.
04-03 04-03 R1	Risk-Informed Approach for Post-Fire Safe-Shutdown Associated Circuit Inspections	Associated circuits
04-04	Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits	RPV P-T limit curves
04-05	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	Loss of offsite power
05-05	Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations	Requirements for insuring sub-criticality in spent fuel pools and exemption to 10CFR70.24 requirement for criticality monitor in SFP.
05-17	Clarification of Requirements for Application of the ASME Code Symbol Stamp on Safety-Related Components	Application of N-Stamp to Class 1, 2, and 3 components
05-29	Anticipated Transients That Could Develop into More Serious Events	Inadvertent ECCS actuation resulting in Condition II or III event
05-30	Clarification of Post-Fire Safe-Shutdown Circuit Regulatory Requirements	

Information Notices

Information Notice	Title	Issue Addressed
80-01	Fuel Handling Events	
80-12	Instrument Failure Causes Opening of PORV and Block Valve	
80-21	Anchorage and Support of Safety-Related Electrical Equipment	
80-44	Actuation of ECCS in the Recirculation Mode while in Hot Shutdown	
81-09	Degradation of Residual Heat Removal (RHR) System	Loss of Shutdown Cooling event
81-10	Inadvertent Containment Spray Due to	

Information Notice	Title	Issue Addressed
	Personnel Error	
81-27	Flammable Gas Mixtures in the Waste Gas Decay Tanks in PWR Plants	
82-17	Overpressurization of Reactor Coolant System	Inadvertent overpressurization of RCS
82-19	Loss of High Head Safety Injection Emergency Boration and Reactor Coolant Makeup Capability	Hydrogen intrusion into CVCS
82-30	Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants	
82-45	PWR Low Temperature Overpressure Protection	
83-11	Possible Seismic Vulnerability of Old Lead Storage Batteries	
83-17	Electrical Control Logic Problem Resulting in Inoperable Auto-Start of Emergency Diesel Generator Units	
83-36	Impact of Security Practices on Safe Operations	Consideration of locked or security doors impeding access during abnormal operating events
83-41	Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment	
83-44	Potential Damage to Redundant Safety Equipment as a Result of Backflow Through The Equipment and Floor Drain System	
83-46	Common-Mode Valve Failures Degrade Surry's Recirculation Spray Subsystem	
83-49	Sampling and Prevention of Intrusion of Organic Chemicals Into Reactor Coolant Systems	Glycol was almost introduced into the RCS at Hatch. Is this a procedure issue or a flow path issue ?
83-58	Transamerica Delaval Diesel Generator Crankshaft Failure	
84-02	Operating a Nuclear Power Plant at Voltage Levels Lower Than Analyzed	Reduced 4160 V bus voltage
84-09	Lessons Learned From NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)	
84-13	Potential Deficiency In Motor-Operated Valve Control Circuits and Annunciation	MOV thermal overload (TOL) bypass condition may preclude timely detection of a failure of a safety-related motor. PP&L considered a bypass circuit design and possible continuous annunciation and indication of TOL trip conditions.
84-17	Problems With Liquid Nitrogen Cooling Components Below the Nil Ductility Temperature	Concern that liquid or cold gaseous nitrogen can potentially cool vital components of the plant below the nil ductility temperatures of susceptible associated materials.
84-18	Stress Corrosion Cracking in PWR Systems	Is this a design issue or an operational issue?
84-32	Auxiliary Feedwater Sparger and Pipe Hanger Damage	Water hammer in the Aux Feedwater System.
84-42	Equipment Availability For Conditions During Outages not Covered by Technical Specifications	Availability of electrical systems during outages.
84-47	Environmental Qualification Tests of Electrical Terminal Blocks	
84-57	Operating Experience Related to Moisture Intrusion in Safety-Related Electrical Equipment At Commercial Power Plants	
84-67	Recent Snubber Inservice Testing With	

Information Notice	Title	Issue Addressed
	High Failure Rates	
84-70 84-70 S1	Reliance On Water Level Instrumentation With a Common Reference Leg	
84-71	Graphitic Corrosion of Cast Iron in Salt Water	Salt water corrosion of the safety related service water system.
84-74	Isolation of Reactor Coolant System From Low-Pressure Systems Outside Containment	
84-86	Isolation Between Signals of The Protection System and Non-Safety-Related Equipment	
84-87	Piping Thermal Deflection Induced by Stratified Flow	
84-90	Main Steam Line Break Effect on Environmental Qualification of Equipment	W analysis did not consider the possibility of super-heated steam following MSLB and its impact on EQ.
84-93	Potential For Loss of Water From The Refueling Cavity	Refueling Cavity seal failure. Does U.S. EPR have an inflatable seal?
85-09	Isolation Transfer Switches and Post-Fire Shutdown Capability	Potential deficiencies in the electrical design of isolation transfer switches that provide electrical isolation of certain shutdown circuits from the control room and other essential fire areas during post-fire accident conditions.
85-14	Failure of Heavy Control Rod (B4C) Drive Assembly to Insert on a Trip Signal	Westinghouse CRDMs were found to have loose breech guide screws. Does U.S. EPR have similar design?
85-25	Consideration of Thermal Conditions of the Design and Installation for Diesel Generator Exhaust Silencers	
85-28	Partial Loss of AC Power and Diesel Generator Degradation	D/G voltage regulator setpoint issue.
85-30	Microbiologically Induced Corrosion of Containment Service Water System	Microbiologically induced corrosion (MIC) of essential service water.
85-34	Heat Tracing Contributes to Corrosion Failure of Stainless Steel Piping	
85-39	Auditability of Electrical Equipment Qualification Records at Licensees' Facilities	
85-40	Deficiencies in Equipment Qualification Testing and Certification Process	
85-71	Containment Integrated Leak Rate Tests	
85-73	Emergency Diesel Generator Control Circuit Logic Design Error	
85-76	Recent Water Hammer Events	
85-77	Possible Loss of Emergency Notification System Due to Loss of AC Power	
85-86	Lightning Strikes at Nuclear Power Generating Stations	Impact of lightning strikes on solid state protection equipment.
85-87	Hazards of Inerting Atmospheres	
85-89	Potential Loss of Solid-State Instrumentation Following Failure of Control Room Cooling	
85-91	Load Sequencers for Emergency Diesel Generators	The design of the electric power system at the Duane Arnold nuclear plant included features to sequence ESF loads onto the EDGs, but not to sequence loads onto offsite power.
85-94	Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA	It was discovered that minimum flow requirements might not or could not be met for some emergency core cooling system (ECCS) pumps under SBLOCA

Information Notice	Title	Issue Addressed
		conditions.
86-01	Failure of Main Feedwater Check Valves Causes Loss of Feedwater System Integrity and Water-Hammer Damage	Electric motor driven main feedwater pump failed to trip, check valve leakage caused feedwater heater tube breaks and MFW water hammer.
86-10	Safety Parameter Display System Malfunctions	SPDS survey results
86-11	Inadequate Service Water Protection Against Core Melt Frequency	Failure to consider service water as a potential core melt scenario.
86-38	Deficient Operator Actions Following Dual Function Valve Failures	Does U.S. EPR have any duel function valves?
86-40	Degraded Ability to Isolate the Reactor Coolant System from Low-Pressure Coolant Systems in BWRs	Over-pressurization of low-pressure systems
86-60	Unanalyzed Post-LOCA Release Paths	
86-63	Loss of Safety Injection Capability	Leakage of high concentration boric acid
86-64 86-64 S1	Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures	
86-70	Potential Failure of All Emergency Diesel Generators	Design deficiencies that could result in overloading one or both D/Gs.
86-73	Recent Emergency Diesel Generator Problems	
86-76	Problems Noted in Control Room Emergency Ventilation Systems	
86-77	Computer Program Error Report Handling	
86-83	Underground Pathways into Protected Areas, Vital Areas, Material Access Areas, and Controlled Access Areas	
86-91	Limiting Access Authorizations	
86-102	Repeated Multiple Failures of Steam Generator Hydraulic Snubbers Due to Control Valve Sensitivity	
86-105	Potential for Loss of Reactor Trip Capability at Intermediate Power Levels	
86-106 86-106 S1 86-106 S2 86-106 S3	Feedwater Line Break	Erosion-corrosion of secondary system piping
87-09	Emergency Diesel Generator Room Cooling Design Deficiency	
87-11	Enclosure of Vital Equipment within Designated Vital Areas	
87-14	Actuation of Fire Suppression System Causing Inoperability of Safety-Related Ventilation Equipment	Effects of inadvertent operation or leaks in moderate energy lines of the fire suppression system.
87-20	Hydrogen Leak in Auxiliary Building	Proper valve application for use with hydrogen and hydrogen line routing.
87-24	Operational Experience Involving Losses of Electrical Inverters	Inverter failures
87-34	Single Failures in Auxiliary Feedwater Systems	Potential single failures of auxiliary feedwater pump start and protective pump trip.
87-49	Deficiencies in Outside Containment Flooding Protection	Loss of safe shutdown capability as a consequence of potential flooding of safety-related equipment outside containment.
87-50	Potential LOCA at High- and Low-Pressure Interfaces from Fire Damage	Possible initiation of a LOCA as a result of fire damage in the control room or the cable spreading room.
87-53	Auxiliary Feedwater Pump Trips Resulting from Low Suction Pressure	AFW pump trips due to low pressure oscillations that occurred during pump start despite sufficient steady-state NPSH.
87-57	Loss of Emergency Boration Capability Due to Nitrogen Gas Intrusion	Illustrates potential for system inoperability resulting from air/gas intrusion.

Information Notice	Title	Issue Addressed
87-59	Potential RHR Pump Loss	Minimum flow recirculation line configuration for the residual heat removal (RHR) pumps.
87-63	Inadequate Net Positive Suction Head in Low Pressure Safety Systems	Excessive flow rates in low pressure safety systems that could occur following a LOCA.
88-08	Chemical Reactions with Radioactive Waste Solidification Agents	
88-23 88-23 S1 88-23 S2 88-23 S3 88-23 S4 88-23 S5	Potential for Gas Binding of High-Pressure Safety Injection Pumps During a Loss-of-Coolant Accident	
88-31	Steam Generator Tube Rupture Analysis Deficiency	Tube bundle uncovery during an SGTR event
88-41	Physical Protection Weaknesses Identified Through Regulatory Effectiveness Reviews (RERs)	
88-47	Slower-than-Expected Rod-Drop Times	Slow rod drop times due to test method
88-61	Control Room Habitability - Recent Reviews of Operating Experience	
88-74	Potentially Inadequate Performance of ECCS in PWRs During Recirculation Operation Following a LOCA	
88-75 88-75 S1	Disabling of Diesel Generator Output Circuit Breakers by Anti-Pump Circuitry	
88-80	Unexpected Piping Movement Attributed to Thermal Stratification	Pressurizer surge line movement
88-86 88-86 S1	Operating with Multiple Grounds in Direct Current Distribution Systems	
88-99	Detection and Monitoring of Sudden and/or Rapidly Increasing Primary-to-Secondary Leakage	
89-08	Pump Damage Caused by Low-Flow Operation	Operating pumps at flows significantly below their design flow rates resulted in slow deterioration of pump internals occurring over a long period.
89-30 89-30 S1	High Temperature Environments at Nuclear Power Plants	Higher than expected Reactor Building temperatures invalidated EQ calculations.
89-32 89-32 S1	Surveillance Testing of Low-Temperature Overpressure Protection Systems	
89-50	Inadequate Emergency Diesel Generator Fuel Supply	Non-conservative calculation of fuel oil supply.
89-54	Potential Overpressurization of the Component Cooling Water System	Potential for overpressurization of CCW due to underdesign in relief capacity of the CCW lines connected to the thermal barrier heat exchangers on the reactor coolant pumps.
89-71	Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a LOCA	Potential for a Component Cooling Water System single failure to cause a diversion of cooling water flow from ECCS pump cooling loads.
89-73	Potental Overpressurization of Low Pressure Systems	Potential for overpressurization of ECCS/RB spray sub-systems during surveillance testing.
89-79 89-79 S1	Degraded Coatings and Corrosion of Steel Containment Vessels	
89-87	Disabling of Emergency Diesel Generators by Their Neutral Ground-Fault Protection Circuitry	Neutral ground-fault relays potentially shut down the emergency diesel generators.
90-31	Update on Waste Form and High Integrity Container ... Review Status, Identification of Problems with Cement Solidification ... Waste Mishaps	
90-39	Recent Problems with Service Water	

Information Notice	Title	Issue Addressed
	Systems	
90-53	Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment	Potential for a seismic event to fail house heating steam system and result in the degradation of safety related equipment classified for mild environments.
90-69	Adequacy of Emergency and Essential Lighting	
91-06	Lock-Up of Emergency Diesel Generator and Load Sequencer Control Circuits Preventing Restart of Tripped EDG	
91-26	Potential Nonconservative Errors ... Working Format Hansen-Roach Cross-Section Set Provided with the KENO and SCALE Codes	
91-38	Thermal Stratification in Feedwater System Piping	Thermal stratification in horizontal feedwater piping inside containment.
91-40	Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to Environment	
91-43	Recent Incidents Involving Rapid Increases in Primary-to-Secondary Leak Rate	
91-50 91-50 S1	A Review of Water Hammer Events After 1985	
91-54	Foreign Experience Regarding Boron Dilution	
91-57	Operational Experience on Bus Transfers	
91-66	(1) Erroneous Data in ... NUREG/CR-0095 and (2) Thermal Scattering Data Limitation ... KENO and SCALE Codes	
91-69	Errors in Main Steam Line Break Analyses for Determining Containment Parameters	
91-80	Failure of Anchor Head Threads on Post-Tensioning System During Surveillance Inspection	Load applied to the anchor head threads during stressing operations exceeded the anchor head material shear yield stress resulting in anchor head failure.
92-02	RELAP5/MOD3 Computer Code Error Associated With the Conservation of Energy Equation	RELAP5/MOD3 used in Ch 15 ???
92-06 92-06 S1	Reliability of ATWS Mitigation System and Other NRC Required Equipment Not Controlled by Plant Technical Specifications	Operability control for beyond-design-basis systems
92-07	Rapid Flow-Induced Erosion/Corrosion of Feedwater Piping	
92-09	Overloading and Subsequent Lock Out of Electrical Buses During Accident Conditions	
92-12	Effects of Cable Leakage Currents on Instrument Settings and Indications	
92-20	Inadequate Local Leak Rate Testing	
92-35	Higher ... Erosion/Corrosion in Unisolable Reactor Coolant Pressure Boundary Piping Inside Containment at a BWR	Feedwater piping erosion/corrosion
92-36 95-36 S1	Intersystem LOCA Outside Containment	
92-39	Unplanned Return to Criticality During Reactor Shutdown	
92-49	Recent Loss or Severe Degradation of	

Information Notice	Title	Issue Addressed
	Service Water Systems	
92-54	Level Instrumentation Inaccuracies Caused by Rapid Depressurization	
92-61 92-61 S1	Loss of High Head Safety Injection	
92-69	Water Leakage From Yard Area Through Conduits Into Buildings	
92-86	Unexpected Restriction to Thermal Growth of Reactor Coolant Piping	
93-06	Potential Bypass Leakage Paths Around Filters Installed in Ventilation Systems	
93-11	Single Failure Vulnerability of Engineered Safety Features Actuation Systems	
93-13	Undetected Modification of Flow Characteristics in the High Pressure Safety Injection System	
93-17 93-17 R1	Safety Systems Response to Loss of Coolant and Loss of Offsite Power	Load sequencer response to LOCA with delayed loss of off-site power or LOOP with delayed LOCA.
93-20	Thermal Fatigue Cracking of Feedwater Piping to Steam Generators	
93-25	Electrical Penetration Assembly Degradation	
93-32	Nonconservative Inputs for Boron Dilution Event Analysis	
93-34 93-34 S1	Potential For Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment	
93-55	Potential Problem With Main Steamline Break Analysis for Main Steam Vaults/Tunnels	Mass and energy release for MSLB
93-57	Software Problems Involving Digital Control Console Systems at Non-Nuclear Reactors	
93-58	Nonconservatism in Low-Temperature Overpressure Protection For Pressurized-Water Reactors	
93-66	Switchover to Hot-Leg Injection Following a Loss-Of-Coolant Accident in Pressurized Water Reactors	
93-82	Recent Fuel and Core Performance Problems in Operating Reactors	
93-84	Determination of Westinghouse Reactor Coolant Pump Seal Failure	Response to failure of RCP #1 seal
93-92	Plant Improvements to Mitigate Common Dependencies in Component Cooling Water Systems	
93-99	Undervoltage Relay and Thermal Overload Setpoint Problems	
94-03	Deficiencies Identified During Service Water System Operational Performance Inspections	
94-23	Guidance to hazardous, Radioactive and Mixed Waste Generators on the Elements of a Waste Minimization Program	LLW and mixed waste minimization
94-27	Facility Operating Concerns Resulting From Local Area Flooding	Flooding via below grade walls
94-29	Charging Pump Trip During a Loss-of-Coolant Event Caused by Low Suction Pressure	Too many pumps aligned to a single suction path resulted in charging pump trip during accident mitigation.

Information Notice	Title	Issue Addressed
94-30 94-30 S1	Leaking Shutdown Cooling Isolation Valves at Cooper Nuclear Station	Over-pressurization of RHR
94-43	Determination of Primary-to-Secondary Steam Generator Leak Rate	
94-58	Reactor Coolant Pump Lube Oil Fire	
94-59	Accelerated Dealloying of Cast Aluminum-Bronze Valves Caused by Microbiologically Induced Corrosion	Microbiologically induced corrosion (MIC) of essential service water.
94-60	Potential Overpressurization of Main Stream System	
94-62	Operational Experience on Steam Generator Tube Leaks and Tube Ruptures	Steam generator tube leakage and leakage detection
94-64 94-64 S1	Reativity Insertion Transient and Accident Limits for High Burnup Fuel	
94-77	Malfunction in Main Generator Voltage Regulator Causing Overvoltage at Safety-related Electrical Equipment	
94-79	Microbiologically Influenced Corrosion of Emergency Diesel Generator Service Water Piping	Microbiologically induced corrosion (MIC) of essential service water.
94-80	Inadequate DC Ground Detection in Direct Current Distribution System	
94-82	Concerns Regarding Essential Chiller Reliability During Periods of Low Cooling Water Temperature	
95-10 95-10 S1 95-10 S2	Potential For Loss of Automatic Engineered Safety Features Actuation	Potential impact of a steam line break on protection system functions in the Turbine Building.
95-36 95-36 S1	Potential Problems with Post-Fire Emergency Lighting	
95-37	Inadequate Offsite Power Systems Voltages During Design-Basis Events	
95-54	Decay Heat Management Practices During Refueling Outages	Heat capacity of spent fuel cooling system.
96-01	Potential for High Post-Accident Closed-Cycle Cooling Water Temperatures to Disable Equipment Important to Safety	
96-09 96-09 S1	Damage in Foreign Steam Generator Internals	
96-27	Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation	
96-36	Degradation of Cooling Water Systems Due to Icing	Degradation of condenser circulating water, service water and fire protection due to ice build-up.
96-49	Thermally Induced Pressurization of Nuclear Power Facility Piping	
96-55	Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps under Design Basis Accident Conditions	
97-01	Improper Electrical Grounding Results in Simultaneous Fires in The Control Room And The Safe- Shutdown Equipment Room	Grounding error caused simultaneous fires in the control room and the Train B dc equipment room.
97-05	Offsite Notification Capabilities	
97-09	Inadequate Main Steam Safety Valve (MSSV) Set- Point and Performance Issues Associated With Long MSSV Inlet Piping	
97-10	Liner Plate Corrosion in Concrete Containments	

Information Notice	Title	Issue Addressed
97-13	Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants	
97-21	Availability of Alternate AC Power Source Designed for Station Blackout Event	
97-22	Failure of Welded-Steel Moment-Resisting Frames During The Nothridge Earthquake	
97-41	Potentially Undersized Emergency Diesel Generator (EDG) Oil Coolers	It was determined that the Limerick EDG lubricating oil coolers were undersized relative to the design conditions reported on the heat exchanger data sheet.
97-46	Unisolable Crack in High-Pressure Injection Piping	Crack in HPI/RCS nozzle
97-76	Degraded Throttle Valves in Emergency Core Cooling System Resulting From Cavitation-Induced Erosion During a Loss-of-coolant Accident	
97-85	Effects of Crud Buildup and Boron Deposition on Power Distribution and Shutdown Margin	
97-88	Experiences During Recent Steam Generator Inspections	
98-07	Offsite Power Reliability Challenges from Industry Deregulation	
98-11	Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants	
98-25	Loss of Inventory from Safety-Related, Closed-Loop Cooling Water Systems	
98-27	Steam Generator Tube End Cracking	
98-29	Predicted Increase in Fuel Rod Cladding Oxidation	
98-31	Fire Protection System Design Deficiencies and Common-Mode Flooding of Emergency Core Cooling System Rooms at Washington Nuclear Project Unit 2	
98-40	Design Deficiencies Can Lead to Reduced ECCS Pump Net Positive Suction Head During Design-Basis Accidents	
98-41	Spurious Shutdown of Emergency Diesel Generators from Design Oversight	Automatic shutdown of emergency diesel due to decrese in non-safety starting air pressure.
99-10 99-10 R1	Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments	
99-17	Problems Associated with Post-Fire Safe-Shutdown Circuit Analyses	
00-06	Offsite Power Voltage Inadequacies	
00-08	Inadequate Assessment of the Effect of Differential Temperatures on Safety-Related Pumps	Events highlight the importance of having test programs that include suitable qualification testing under the most adverse design conditions (e.g., temperature and differential temperature).
00-14	Non-Vital Bus Fault Leads to Fire and Loss of Offsite Power	
00-20	Potential Loss of Redundant Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers	
01-05	Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head	

Information Notice	Title	Issue Addressed
	Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3	
02-01	Metalclad Switchgear Failures and Consequent Losses of Offsite Power	
02-10 02-10 S1	Nonconservative Water Level Setpoints on Steam Generators	Errors in SG water level setpoint analysis.
02-11	Recent Experience with Degradation of Reactor Pressure Vessel Head	
02-13	Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation	
02-24	Potential Problems with Heat Collectors on Fire Protection Sprinklers	Use of metal plate to activate sprinklers.
02-27	Recent Fires at Commercial Nuclear Power Plants in the United States	
03-14	Potential Vulnerability of Plant Computer Network to Worm Infection	Computer worm disabled SPDS and plant computer.
03-19	Unanalyzed Condition of Reactor Coolant Pump Seal Leakoff Line During Postulated Fire Scenarios or Station Blackout	Millstone 3 identified the potential for overpressurization of the RCP seal lealoff lines.
04-07	Plugging of Safety Injection Pump Lubrication Oil Coolers With Lakeweed	Biofouling of service water coolers
04-19	Problems Associated with Back-up Power Supplies to Emergency Response Facilities and Equipment	Back-up power for the EOF
05-03	Inadequate Design and Installation of Seismic-Gap Fire Barriers	Original seismic-gap fire barrier design found to be deficient.
05-11	Internal Flooding/Spray-Down of Safety-Related Equipment Due to Unsealed Equipment Hatch Floor Plugs and/or Blocked Floor Drains	Auxiliary (Safeguards) Building flooding
05-13	Potential Non-Conservative Error in Modeling Geometric Regions in the Keno-V.a Criticality Code	
05-20	Electrical Distribution System Failures Affecting Security Equipment	Loss of power to security equipment
05-24	Nonconservatism in Leakage Detection Sensitivity	Low-leakage fuel results in non-conservative RCS leak detection capability.
05-26	Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment	
05-31	Potential Non-Conservative Error in Preparing Problem-Dependent Cross Sections for Use with the Keno V.A. or Keno-VI Criticality Code	

U.S. EPR Design Certification - Applicable Regulatory Requirements, Guidance and Issues

A. Regulations	
10 CFR 20	Standards for Protection Against Radiation
10 CFR 21	Reporting of Defects and Noncompliance
10 CFR 50.34 (f)	Additional TMI-related Requirements
10 CFR 50.34 (g)	Combustible Gas Control
10 CFR 50.34 (h)	Conformance with the Standard Review Plan
10 CFR 50.34a	Design Objectives for Equipment to Control Release of Radioactive Material in Effluents - Nuclear Power Reactor
10 CFR 50.36	Technical Specification
10 CFR 50.44	Standards for Combustible Gas Control System in LWR
10 CFR 50.46	Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors
10 CFR 50.48	Fire Protection
10 CFR 50.49	Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Reactors
10 CFR 50.55a	Codes and Standards
10 CFR 50.60	Acceptance Criteria for Fracture Prevention Measures for LWRs for Normal Operation
10 CFR 50.61	Pressurized Thermal Shock
10 CFR 50.62	ATWS
10 CFR 50.63	Station Blackout
10 CFR 50, App. A (General Design Criteria)	
1	Quality Standards and Records
2	Design Bases for Protection Against Natural Phenomena
3	Fire Protection
4	Environmental and Dynamic Effects Design Bases
5	Sharing of Structures, Systems, and Components
10	Reactor Design
11	Reactor inherent Protection
12	Suppression of Reactor Power Oscillations
13	Instrumentation and Control
14	Reactor Coolant Pressure Boundary
15	Reactor Coolant System Design
16	Containment Design
17	Electric Power Systems
18	Inspection and Testing of Electric Power Systems
19	Control Room
20	Protection System Functions
21	Protection System Reliability and Testability
22	Protection System Independence
23	Protection System Failure Modes
24	Separation of Protection and Control Systems
25	Protection System Requirements for Reactivity Control Malfunctions
26	Reactivity Control System Redundancy and Capability
27	Combined Reactivity Control Systems Capability
28	Reactivity Limits
29	Protection Against Anticipated Operational Occurrences
30	Quality of Reactor Coolant Pressure Boundary
31	Fracture Prevention of Reactor Coolant Pressure Boundary
32	Inspection of Reactor Coolant Pressure Boundary
33	Reactor Coolant Makeup
34	Residual Heat Removal
35	Emergency Core Cooling
36	Inspection of Emergency Core Cooling System
37	Testing of Emergency Core Cooling System
38	Containment Heat Removal
39	Inspection of Containment Heat Removal System
40	Testing of Containment Heat Removal System
41	Containment Atmosphere Cleanup
42	Inspection of Containment Atmosphere Cleanup Systems
43	Testing of Containment Atmosphere Cleanup Systems
44	Cooling Water

45	Inspection of Cooling Water System
46	Testing of Cooling Water System
50	Containment Design Basis
51	Fracture Prevention of Containment Pressure Boundary
52	Capability for Containment Leakage Rate Testing
53	Provisions for Containment Testing and Inspection
54	Systems Penetrating Containment
55	Reactor Coolant Pressure Boundary Penetrating Containment
56	Primary Containment Isolation
57	Closed Systems Isolation Valves
60	Control of Releases of Radioactive Materials to the Environment
61	Fuel Storage and Handling and Radioactivity Control
62	Prevention of Criticality in Fuel Storage and Handling
63	Monitoring Fuel and Waste Storage
64	Monitoring Radioactivity Releases
10 CFR 50, App. B	Quality Assurance Criteria
10 CFR 50, App. G	Fracture Toughness Requirements
10 CFR 50, App. H, I	Reactor Vessel Material Surveillance Program Requirement
10 CFR 50, App. J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactor
10 CFR 50, App. K	ECCS Evaluation Models
10 CFR 50, App. O	Standardization of Design
10 CFR 50, App. S	Earthquake Engineering Criteria for Nuclear Power Plants
10 CFR 52	Early Site Permits; Standard Plant Design Certification; and Combined License for Nuclear Power Plants
10 CFR 73	Physical Protection of Plants and Materials
B. USNRC Regulatory Guides	
RG 1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1) (ML003739925)
RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6) (ML003739924)
RG 1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (ML003739927)
RG 1.9	Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants (Draft RS 802-5, Proposed Revision 3, published 11/1988) (Draft DG-1021, Second Proposed Revision 3, published 04/1992) (Rev. 3, ML003739929)
RG 1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations (ML003739934)
RG 1.12	Nuclear Power Plant Instrumentation for Earthquakes (Draft MS 140-5, Proposed Revision 2, published 07/1981) (DG-1016, the Second Proposed Revision 2, published 11/1992) (DG-1033, the Second Proposed Revision 2, published 02/1995) (Rev. 1, ML003739947; Rev. 2, ML003739944)
RG 1.13	Spent Fuel Storage Facility Design Basis (for Comment) (Draft CE 913-5, Proposed Revision 2, published 12/1981) (Rev. 1, ML003739943)
RG 1.14	Reactor Coolant Pump Flywheel Integrity (for Comment) (Rev. 1, ML003739936)
RG 1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, ML003739957)
RG 1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, ML3739960)
RG 1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22)
RG 1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)
RG 1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)
RG 1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear
RG 1.27	Ultimate Heat Sink for Nuclear Power Plants (for Comment) (Rev. 2, ML003739969)
RG 1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, ML003739981) (Draft RS 002-5, Proposed Revision 3, published 03/1981) (Draft DG-1010, Proposed Revision 4, published 11/1992)
RG 1.29	Seismic Design Classification (Rev. 3, ML003739983)
RG 1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, ML003739986)
RG 1.32	Criteria for Power Systems for Nuclear Power Plants (Rev. 2, ML003739990) (DG-1079, Proposed Revision 3, issued 04/2003,
RG 1.34	Control of Electroslag Weld Properties (ML003739997)

RG 1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments (Rev. 2, ML003740001) (Draft SC 810-4, Proposed Revision 3, published 04/1979) (Rev. 3, ML003740007)
RG 1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (ML003740040) (Draft SC 807-4
RG 1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (ML003740046)
RG 1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power
RG 1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, ML003740057)
RG 1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, ML003740067)
RG 1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (ML003740083)
RG 1.41	Preoperational Testing of Redundant On-Site Electric Power Systems To Verify Proper Load Group Assignments (ML003740090)
RG 1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (ML003740095)
RG 1.44	Control of the Use of Sensitized Stainless Steel (ML003740109)
RG 1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (ML003740113)
RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (ML003740127)
RG 1.49	Power Levels of Nuclear Power Plants (Rev. 1, ML003740132)
RG 1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (ML003740136)
RG 1.52	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, ML003740139) (DG-1102, Proposed Revision 3, issued 10/00, ML003756180) (Rev. 3, ML011710176)
RG 1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (ML003740182) (Draft DG-1118, Proposed Revision 1, ML021260080, published 05/2002) (Rev.1, ML032670945) (Rev. 2, ML033220006)
RG 1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (ML003740187) (Draft DG-1976, Proposed
RG 1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (ML003740195)
RG 1.59	Design Basis Floods for Nuclear Power Plants (Errata published 07/30/1980) (Rev. 2, ML003740388)
RG 1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, ML003740207)
RG 1.61	Damping Values for Seismic Design of Nuclear Power Plants (ML003740213)
RG 1.62	Manual Initiation of Protective Actions (ML003740216)
RG 1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Draft EE 405-4, Proposed Revision 3,
RG 1.65	Materials and Inspections for Reactor Vessel Closure Studs (ML003740228)
RG 1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants
RG 1.68.2	Initial Startup Test Program To Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, ML003740258)
RG 1.68.3	Preoperational Testing of Instrument and Control Air Systems (Draft RS 709-4, a proposed revision to Regulatory Guide 1.80, published 10/1980) (ML003740231)
RG 1.69	Concrete Radiation Shields for Nuclear Power Plants (ML003740235)
R.G 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 2, ML01610289) (Rev. 3 in three parts, ML011340072, ML011340108, and ML011340116)
RG 1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (ML003740261)
RG 1.75	Physical Independence of Electric Systems (Rev. 2, ML003740265) (Rev. 3, ML043630448)
RG 1.76	Design Basis Tornado for Nuclear Power Plants (ML003740273)
RG 1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (ML003740298) (Proposed Revision 1, DG-1087, published 02/2001, ML010440064) (Revision 1 incorporates guidance from withdrawn Regulatory Guide 1.95) (Revision 1, ML013100014)
RG 1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, ML003740351)
RG 1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Draft MS 203-4, Proposed
RG 1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, ML003740256)
RG 1.84	Design and Fabrication and Materials Code Case Acceptability, ASME Section III
RG 1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (1974, ML012880422) (Draft EE 042-2, Proposed Revision 1, published 02/1982) (Rev. 1, ML003740271)
RG 1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (Rev. 1, ML003740281)
RG 1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, ML003740290) (Draft DG-1108 , Proposed Revision 2, published 08/01)
RG 1.93	Availability of Electric Power Sources (ML003740292)

RG 1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following
RG 1.99	Radiation Embrittlement of Reactor Vessel Materials (Draft ME 305-4, Proposed Revision 2, published 02/1986) (Rev. 2,
RG 1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Draft EE 108-5, Proposed Revision 2, published 08/1987) (Rev. 2, ML003740293)
RG 1.102	Flood Protection for Nuclear Power Plants (ML003740308)
RG 1.105	Setpoints for Safety-Related Instrumentation (Draft IC 010-5, Proposed Revision 2, published 12/81) (Draft DG-1045, Proposed Revision 3, ML003739248, published 10/96) (Rev. 2, ML003740318; Rev. 3, ML993560062)
RG 1.105	
RG 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, ML003740323)
RG 1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures (Rev. 1, ML003740374)
RG 1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Rev. 1, ML003740384)
RG 1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for Comment) (ML003740332)
RG 1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, ML003740354)
RG 1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Rev. 0-R, ML003740361)
RG 1.115	Protection Against Low-Trajectory Turbine Missiles (Rev. 1, ML003739456)
RG 1.117	Tornado Design Classification (Rev. 1, ML003739346)
RG 1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, ML003739468) (DG-1028, Proposed Revision 3, published
RG 1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, ML003739367)
RG 1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, ML003739380)
RG 1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, ML003739388)
RG 1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, ML003739385)
RG 1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, ML003739392)
RG 1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, ML003740099)
RG 1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, ML003740123)
RG 1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (for Comment) (ML003740128) (Draft RS 050-2, Proposed Revision 1, published 08/1979)
RG 1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, ML032800710) (DG-1101, Proposed Revision 2, issued 02/2001, ML010510162) (Rev. 1, ML003740350)
RG 1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, ML003740137)
RG 1.135	Normal Water Level and Discharge at Nuclear Power Plants (for Comment) (ML003740143)
RG 1.136	Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments") (Draft SC 814-5, Proposed Revision 2, published 11/1979) (Rev. 2, ML003740155)
RG 1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, ML003740180)
RG 1.139	Guidance for Residual Heat Removal (for Comment)
RG 1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in
RG 1.141	Containment Isolation Provisions for Fluid Systems (for Comment) (ML003740194)
RG 1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) (DG-1098, Proposed Revision 2, published 08/2000) (Rev. 1, ML003740197; Rev. 2, ML013100274)
RG 1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled
RG 1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Reissued 02/1983 to correct page 1.145-7) (Rev. 1, ML003740205)
RG 1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1 (Draft SC 721-4 published 08/1979) Previous revisions and their publication dates follow: 0, 03/1981; 2, 06/1983, 3, 07/1984; 4, 09/1985; 5, 08/1986; 6, 05/1988; 7, 07/1989 (ML003740209); 8, 11/1990; 9, 04/1992; 10, 07/1993; 11, 10/1994 (ML003739955); (DG-1050, proposed Revision 12, ML003739242, published 5/1997), (Revision 12, ML003671361, published 5/1999), (DG-1091, proposed Revision 13, ML013120019, published 12/2001), (Revision 13, ML030730423, published 6/2003), (Corrected reprint of Revision 13, ML040230509, published 1/2004)

RG 1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (ML003739979)
RG 1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations (Draft SC 705-4 published 05/1979) (Rev. 1, ML003739996)
RG 1.151	Instrument Sensing Lines (ML003740003) (Draft IC 126-5 published 03/1982)
RG 1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants (11/85) (ML003740088) (Draft IC 127-5 published 03/1983) (Draft DG-1039, Proposed Revision 1, published 05/1995) (Rev. 1, ML003740015)
RG 1.153	Criteria for Safety Systems (12/85) (Draft IC 609-5 published 12/1982) (Draft DG-1042, Proposed Revision 1, published 11/1995) (ML003740019) (Rev. 1, ML003740022)
RG 1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (ML003740028) (Draft SI 502-4 published 01/1986)
RG 1.155	Station Blackout (Issued June 1988, reissued August 1988 with corrected tables) (ML003740034) (Draft SI 501-4 published
RG 1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (ML003740042) (Draft EE 404-4 published 05/1987)
RG 1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (ML003739584) (Draft RS 701-4 published 03/1987)
RG 1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (ML003740047) (Draft EE 006-5 published 08/1987)
RG 1.163	Performance-Based Containment Leak-Test Program (ML003740058) (Draft DG-1037 published 02/1995) (Errata to NEI 94-01
RG 1.165	Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion (ML003740084) (Draft DG-1015 issued 11/1992, Draft DG-1032, issued 02/1995)
RG 1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants
RG 1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants
RG 1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740105) (Draft DG-1056, ML003739146, issued 08/1996)
RG 1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740108) (Draft DG-1057, ML003739141, issued 08/1996)
RG 1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740094) (Draft DG-1058, ML003739228, issued 08/1996)
RG 1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (ML003740101) (Draft DG-1059, ML003740101, issued 08/1996)
RG 1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control
RG 1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (ML003716792) (Draft DG-1081, ML003739148, issued 12/1999)
RG 1.193	ASME Code Cases Not Approved for Use (ML030730440) (Draft guide was issued as DG-1112, 12/01, ML013120071)
RG 1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (ML031530505) (Draft guide was issued as DG-1111, 12/01, ML013130132)
RG 1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (ML031490640) (Draft guide was issued as DG-1113, 01/02, ML020160023)
RG 1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (ML031490611) (Draft guide was issued as DG-1114, 03/02, ML020790125)
RG 1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (ML031490664) (Draft guide was issued as DG-1115, 03/02, ML020790191)
RG 1.199	Anchoring Components and Structural Supports in Concrete (ML033360660) (Draft was issued as DG-1099, 07/02, ML021910490)
RG 1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (ML040630078) (Issued with SRP Chapter 19.1 , 02/2004, ML040630300) (Draft guide was issued as DG-1122, 11/02, ML023360076)
RG 8.5	Criticality and Other Interior Evacuation Signals (Rev. 1, ML003739454)
RG 8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is
RG 8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants -- Design Stage Man-Rem Estimates (Rev. 1, ML003739550)
C. Standard Review Plan	
3.2.1	Seismic Classification (04/96)
3.2.2	System Quality Group Classification (04/96)
3.3.1	Wind Loadings (04/96)
3.3.2	Tornado Loadings (04/96)
3.4.1	Flood Protection, Revision 2, July 1981
3.5.1.1	Internally Generated Missiles (Outside Containment), Revision 2, July 1981

3.5.1.2	Internally Generated Missiles (Inside Containment), Revision 2, July 1981
3.5.1.3	Turbine Missiles (04/96) (04/96)
3.5.1.4	Missiles Generated by Natural Phenomena (04/96)
3.5.1.5	Site Proximity Missiles (Except Aircraft) (04/96)
3.5.1.6	Aircraft Hazards (04/96)
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles (04/96)
3.5.3	Barrier Design Procedures (04/96)
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, Revision 1, July 1981
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Draft Revision 2, April 1996
3.6.3	Leak-Before-Break Evaluation Procedures, Draft, March 1987
3.7.1	Seismic Design Parameters (04/96)
3.7.2	Seismic System Analysis (04/96)
3.7.3	Seismic Subsystem Analysis (04/96)
3.7.4	Seismic Instrumentation (04/96)
3.8.1	Concrete Containment (04/96)
3.8.2	Steel Containment (04/96)
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments (04/96)
3.8.4	Other Seismic Category I Structures (04/96)
3.8.5	Foundations (04/96)
3.9.1	Special Topics for Mechanical Components, Draft Revision 3, April 1996
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment, Draft Revision 3, April 1996
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures, Draft Revision 2, April 1996
3.9.4	Control Rod Drive Systems (04/96)
3.9.5	Reactor Pressure Vessel Internals, Draft Revision 3, April 1996
3.9.6	Inservice Testing of Pumps and Valves, Draft Revision 3, April 1996
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, Draft Revision 3, April 1996
3.11	Environmental Qualification of Mechanical and Electrical Equipment, Draft Revision 3, April 1996
4.2	Fuel System Design, Draft Revision 3, April 1996
4.3	Nuclear Design, Draft Revision 3, April 1996
4.4	Thermal and Hydraulic Design, Draft Revision 2, April 1996
4.5.1	Control Rod Drive Structural Materials, Draft Revision 3, April 1996
4.5.2	Reactor Internal and Core Support Materials, Draft Revision 3, April 1996
4.6	Functional Design of Control Rod Drive System, Draft Revision 2, April 1996
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a, Draft Revision 3, April 1996
5.2.2	Overpressure Protection, Draft Revision 3, April 1996
5.2.3	Reactor Coolant Pressure Boundary Materials, Draft Revision 3, April 1996
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing, Draft Revision 2, April 1996
5.3.1	Reactor Vessel Materials, Draft Revision 2, April 1996
5.3.2	Pressure-Temperature Limits and Pressurized Thermal Shock, Draft Revision, Draft Revision 2, April 1996
5.3.3	Reactor Vessel Integrity, Draft Revision 2, April 1996
5.4.1.1	Pump Flywheel Integrity (PWR) (04/96)
5.4.2.1	Steam Generator Materials (04/96)
5.4.7	Residual Heat Removal (RHR) System, Draft Revision 4, April 1996
5.4.11	Pressurizer Relief Tank, Revision 2, July 1981
5.4.12	Reactor Coolant System High Point Vents (04/96)
6.1.1	Engineered Safety Features Materials, Draft Revision 2, April 1996
6.1.2	Protective Coating Systems (Paints) - Organic Materials, Draft Revision 3, April 1996
6.2.1	Containment Functional Design, Revision 2, July 1981
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments, Revision 2, July 1981
6.2.1.2	Subcompartment Analysis, Revision 2, July 1981
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant, Revision 1, July 1981
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, Revision 1, July 1981
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies, Revision 2, July 1981
6.2.2	Containment Heat Removal Systems, Revision 4, October 1985
6.2.3	Secondary Containment Functional Design, Revision 2, July 1981

6.2.4	Containment Isolation System (04/96)
6.2.5	Combustible Gas Control in Containment, Revision 2, July 1981
6.2.6	Containment Leakage testing (04/96)
6.2.7	Fracture Prevention of Containment Pressure Boundary (04/96)
6.3	Emergency Core Cooling System, Draft Revision 3, April 1996
6.4	Control Room Habitability System, Draft Revision 3, April 1996
6.5.1	ESF Atmosphere Cleanup Systems, Revision 2, July 1981
6.5.2	Containment Spray as Fission Product Cleanup System (04/96)
6.5.3	Fission Product Control Systems and Structures, Revision 2, July 1981
6.6	Inservice Inspection of Class 2 and 3 Components (04/96)
7.0 -7.8	Instrumentation and Controls, Revision 4, June 1997
8.1	Electric Power - Introduction, Draft Revision 3, April 1996
8.2	Offsite Power System, Draft Revision 4, April 1996
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Appendix 8-B	Offsite Power System, Draft Revision 4, April 1996
8.3.1	A-C Power Systems (Onsite), Draft Revision 3, April 1996
8.3.2	D-C Power Systems (Onsite), Draft Revision 3, April 1996
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9.1.3	Spent Fuel Pool Cooling and Cleanup System, Revision 1, July 1981
9.1.4	Light Load Handling System (Related to Refueling), Revision 2, July 1981
9.1.5	Overhead Heavy Load Handling Systems (04/96)
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9.4.2	Spent Fuel Pool Area Ventilation System, Revision 2, July 1981
9.4.3	Auxiliary and Radwaste Area Ventilation System, Revision 2, July 1981
9.4.4	Turbine Area Ventilation System, Revision 2, July 1981
9.4.5	Engineered Safety Feature Ventilation System, Revision 2, July 1981
9.5.1	Fire Protection Program, Revision 4, October 2003
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10.2	Turbine Generator, Revision 2, July 1981
10.2.3	Turbine Rotor Integrity (04/96)
10.3	Main Steam Supply System, Revision 3, April 1984
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10.4.8	Steam Generator Blowdown System (PWR), Draft Revision 3, April 1996
10.4.9	Auxiliary Feedwater System (PWR), Revision 2, July 1981
11.1	Source Terms, Draft Revision 3, April 1996
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11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems (04/96)
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15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR), Draft Revision 3, April 1996
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR, Draft Revision 3, April 1996
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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), Draft Revision 2, April 1996
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries, Draft Revision 2, April 1996
15.2.7	Loss of Normal Feedwater Flow, Draft Revision 2, April 1996
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment, Draft Revision 2, April 1996
15.3.1 - 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions, Draft Revision 2, April 1996
15.3.3 - 15.3.4	Reactor Coolant Pump Motor Seizure and Reactor Coolant Pump Shaft Break, Draft Revision 3, April 1996
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, Draft Revision 3, April 1996
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15.4.3	Control Rod Misoperation (System Malfunction or Operator), Draft Revision 3, April 1996
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR), Draft Revision 2, April 1996
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position, Draft Revision 2, April 1996
15.4.8	Spectrum of Rod Ejection Accidents (PWR), Draft Revision 3, April 1996
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR), Draft Revision 2, April 1996
15.5.1 - 15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory, Draft Revision 2, April 1996
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15.6.5	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary, Draft Revision 3, April 1996
15.6.5.A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident including Containment Leakage Contribution, Draft Revision 2, April 1996
15.6.5.B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution, Draft Revision 2, April 1996
15.6.5.D	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR), , Draft Revision 2, April 1996
15.7.4	Radiological Consequences of Fuel Handling Accidents, Draft Revision 2, April 1996
15.7.5	Spent Fuel Cask Drop Accidents, Draft Revision 3, April 1996
17.1	Quality Assurance During the Design and Construction Phases (04/96)
17.3	Quality Assurance Program Description, Revision 0, July 1990
18	Human Factors Engineering, Revision 1, February 2004

19	Use of Probabilistic Risk Assessment in Plant-specific, Risk-informed Decisionmaking: General Guidance
19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities
D. TMI Items	
1. NUREG 0660, Item II.B.8	Perform Plant/Site Specific TMI Related Risk Assessment
2. NUREG 0737, Item II.E.1	Auxiliary Feedwater System Evaluation
3. NUREG 0737, Items II.K.2.16 and II.K.3.25	Reactor Coolant Pump Seals
4. NUREG 0737, Item II.K.3.2	Automatic PORV Isolation
12. NUREG 0660, Item II.B.8	Hydrogen Control System Evaluation
15. NUREG 0737, Item I.D.1	Control Room Design
16. NUREG 0737, Item I.D.2	Safety Parameter Display System
17. NUREG 0737, Item I.D.3	Safety System Status Indication
18. NUREG 0737, Item II.B.1	RCS High Point Vents
19. NUREG 0737, Item II.B.2	Plant Radiation Shielding
20. NUREG 0737, Item II.B.3	Post Accident Sampling
21. NUREG 0660, Item II.B.8	Hydrogen Control
22. NUREG 0737, Item II.D.1	RCS Valve Testing
23. NUREG 0737, Item II.D.3	Valve Position Indication
24. NUREG 0737, Item II.E.1.2	AFW System Initiation and Indication
25. NUREG 0737, Item II.E.3.1	Pressurizer Heater Power Supplies
26. NUREG 0737, Item II.E.4.2	Containment Isolation System
27. NUREG 0933, Item II.E.4.4	Containment Purging/Venting
29. NUREG 0737, Item II.F.1	Specific Accident Monitoring Instrumentation
30. NUREG 0737, Item II.F.2	Inadequate Core Cooling Instrumentation
31. NUREG 0933, Item II.F.3	Post Accident Monitoring Instrumentation
32. NUREG 0737, Item II.G.1	Power Supplies for Pressurizer PORVs, Block Valves, and Level Indication
37. NUREG 0737, Item II.A.1.2	Emergency Response Facilities
38. NUREG 0737, Item III.D.1.1	Leakage Control Outside Containment
39. NUREG 0737, Item III.D.3.3	In-Plant Monitoring
40. NUREG 0737, Item III.D.3.4	Control Room Habitability
41. NUREG 0737, Item I.C.5	Industry Experience
42. NUREG 0933, Item I.F.1	Quality Assurance List

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43. NUREG 0737, Item I.F.2	Quality Assurance Program
44. NUREG 0660, Item II.B.8	Dedicated Containment Penetrations
45. NUREG 0660, Item II.B.8	Containment Design
46. NUREG 0737, Item II.E.4.1	Hydrogen Recombiners
47. NUREG 0933, Item II.J.3.1	Management Plan
E. Policy Concerns	
[51 FR 24643 (1986)]	Advanced Reactor Policy
[51 FR 30028 (1986)]	Safety Goal Policy
[50 FR 32138 (1985)]	Severe Accident Policy
[52 FR 34884 (1987)]	Standardization Policy
F. SECY Compliance Issues	
Evolutionary LWR Public Safety Goals	Position on acceptable probabilistic targets to conform to the Commission's public safety goal. This issue was originally identified in SECY 90-016 but was not included as a licensing issue in SECY 93-087.
Source Term	Position on the use of a realistic source term in design that deviates from the siting requirements in 10CFR100
Anticipated Transient Without SCRAM (ATWS)	Position on the current practices and design features to achieve a high degree of protection against an ATWS.
Mid-Loop Operation	Position on design features necessary to ensure a high degree of reliability of RHR systems in PWR.
Station Blackout (SBO)	Position on methods to mitigate the effects of a loss-of-offsite-power concurrent with failure of the emergency diesel generators.
Fire Protection	Positions on design configuration and features the fire protection system and other management schemes to ensure safe shutdown of the reactor.
Intersystem LOCA	Position on acceptable design practices and preventative measures to minimize the probability of an ISLOCA
Hydrogen Generation and Control	Position on acceptable requirements to measure and to mitigate against the effects of hydrogen produced due to a water reaction with zirconium fuel cladding.
Core-Concrete Interaction (Ability to Cool Core Debris)	Acceptability criteria for cooling area and quenching ability regarding corium interaction with concrete.
High Pressure Core Melt Ejection	Position on acceptable design features to prevent the event of a high-pressure core melt ejection
Containment Performance	Position on acceptable conditional containment failure probabilities or other analyses to ensure a high degree of protection from the containment.
Equipment Survivability	Position on the applicability of environmental qualification and quality assurance requirements related to plant features provided only for severe-accident protection.
Operating Basis Earthquake / Safe Shutdown Earthquake	Position on the applicability of the OBE in design and the possibility of decoupling the OBE and SSE in the design of safety systems.
Industry Codes and Standards	Position on use of recently developed or modified design codes and industry standards in ALWR designs that have not been reviewed for acceptability by the NRC.
Electrical Distribution	Position on acceptable practices relating to the electrical distribution of safety- and non-safety loads.
Seismic Hazard Curves and Design Parameters	Position on use of proposed generic bounding seismic hazard curves and performance of seismic PRA.
Leak-Before-Break	Position on use of leak-before-break concept.
Tornado Design Basis	Position on the maximum tornado wind speed to be used for a design basis tornado.
Containment Bypass	Position on ALWR design against containment bypass. Specifically, failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling heat exchanger tubes in large pools of water outside containment.
Post Accident Sampling System	Position on the required capability to analyze dissolved hydrogen, oxygen, and chloride in accordance with applicable regulations.
Site-Specific PRA and Analysis of External Events	Position on the inclusion of external event analysis beyond the design basis, that needs to be addressed as part of the plant PRA during the design certification review.
Severe Accident Mitigation Design Alternatives (SAMDAs)	Position on the consideration of SAMDA as part of the final design approval/design certification of an advanced reactor.

Definition of Passive Failure	Position on the staff redefining some passive failures of components as active failures (i.e., check valves) to cause valves to be evaluated in a much more stringent manner than in previous licensing review
Safe Shutdown Requirements	Position on using non-safety grade active cooling systems to bring a reactor to cold shutdown since non-safety RHR systems to not comply with the guidance of RG 1.139 or branch technical position 5-1
Control Room Habitability	Position on appropriate analytical methods (i.e., dose limits and accident duration) to be used in determining the acceptability criteria for control room habitably in accordance with regulatory standards.
Radionuclide Attenuation	Position on fission product removal processes inside containment by natural effects and holdup by the secondary building and piping systems in addition to commission position on containment spray systems for passive ALWRs.
G. Generic Safety Issues/Unresolved Safety Issues (GSIs/USIs)	
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I.D.2	Plant Safety Parameter Display Console
I.D.5(2)	Plant Status and Post-Accident Monitoring
I.F.1	Expand QA List
I.F.2(2)	Include QA Personnel in Review and Approval of Plant
I.F.2(3)	Include QA Personnel in All Design, Construction,
I.G.1	Training Requirements
I.G.2	Scope of Test Program
II.B.1	Reactor Coolant System Vents
II.B.2	Plant Shielding to Provide Access to Vital Areas and
II.B.3	Post-Accident Sampling
II.B.8	Rulemaking Proceeding on Degraded Core Accidents
II.D.1	Testing Requirements
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II.E.1.1	Auxiliary Feedwater System Evaluation
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and
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II.K.1(10)	Review and Modify Procedures for Removing Safety-
II.K.1(13)	Propose Technical Specification Changes Reflecting
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break
II.K.3(1)	Install Automatic PORV Isolation System and Perform
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps
II.K.3(25)	Effect of Loss of AC Power on Pump Seals
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance
III.A.1.2	Upgrade Licensee Emergency Support Facilities
III.A.1.2(1)	Technical Support Center
III.A.1.2(2)	On-Site Operational Support Center
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining
III.D.3.3	In-plant Radiation Monitoring
III.D.3.4	Control Room Habitability
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Issue 163	Multiple Steam Generator Tube Leakage
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