ESBWR Design Certification Application Acceptance Review Checklist

Technical Information

	The application for a design certification contains the following technical information required by 10 CFR Part 52.47:	Yes	No
I.	The application contains the technical information which is required of applicants for construction permits and operating licenses 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)]. (see Attachment 1)	9	9
II.	The application contains a demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth 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)].	9	9
III.	The application contains the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters [10 CFR 52.47(a)(1)(iii)].	9	9
IV.	The application contains proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application [NUREG-0933 Supplement 28, published August 2004] which are technically relevant to the design [10 CFR 52.47(a)(1)(iv)].	9	9
V.	The application contains a design-specific probabilistic risk assessment [10 CFR 52.47(a)(1)(v)].	9	9
VI.	The application contains proposed inspections, tests, analyses, and acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the design certification [10 CFR 52.47(a)(1)(vi)].	9	9
VII.	The application contains the interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessment required by paragraph V above [10 CFR 52.47(a)(1)(vii)].	9	9
VIII.	The application contains justification that compliance with the interface requirements of paragraph VII above is verifiable through inspection, testing (either in the plant or elsewhere), or analysis. The method to be used for verification of interface requirements must be included as part of the proposed inspections, tests, analyses, and acceptance criteria required by paragraph VI above [10 CFR 52.47(a)(1)(viii)].	9	9

		Yes	No
IX.	The application contains a representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the staff in its review of the final safety analysis and probabilistic risk assessment required by paragraph V above, and to permit assessment of the adequacy of the interface requirements called for by paragraph VII above [10 CFR 52.47(a)(1)(ix)].	9	9
X.	The application contains a level of design information sufficient to enable the Commission to judge the applicants' proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted includes performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination [10 CFR 52.47(a)(2)].	9	9
XI.	The application contains any information beyond that required by 10 CFR 52.47 which the staff advised the applicant to submit with the design certification application [10 CFR 52.47(a)(3)]. This includes addressing issues discussed in SECY papers and SRMs needed to support the review of the application. (see Attachment 2)	9	9
XII.	The application contains an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink [10 CFR 52.47(b)(1) and 10 CFR 52.47(b)(2)(i)(A)(4)]; or there has been acceptable testing of an appropriately sited, full size, prototype [10 CFR 52.47(b)(2)(i)(B)].	9	9
XIII.	The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof [10 CFR 52.47(b)(2)(i)(A)(1)].	9	9
XIV.	Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof [10 CFR 52.47(b)(2)(i)(A)(2)].	9	9
XV.	Sufficient data exists on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions [10 CFR 52.47(b)(2)(i)(A)(3)].	9	9
XVI.	The application proposes the specific testing necessary to support certification of the design [10 CFR 52.47(b)(2)(ii)].	9	9

Procedural Requirements

		Yes	No
	The design certification application meets the following procedural requirements:		
A.	The application follows the relevant sections of 10 CFR 50.4 [10 CFR 52.45(d)].		
a.	The application is addressed to the Document Control Desk and sent by mail, hand delivered, or sent by electronic submission in accordance with the requirements of 10 CFR 50.4(a).	9	9
b. `	If the application is on paper, the submission must be the signed original [10 CFR 50.4(b)].	9	9
c.	The form of the application meets the requirements of 10 CFR 50.4(c).	9	9
B.	The application is submitted under oath or affirmation [10 CFR 50.30(b), 10 CFR 52.45(d)].	9	9
C.	The application for design certification must include an application for a final design approval [10 CFR 52.45(c)].	9	9
D.	The application includes an agreement limiting access to Classified Information [10 CFR 50.37].	9	9
E.	The application meets the provisions of 10 CFR 2 related to public availability including the provisions of 10 CFR 2.390 concerning proprietary information [10 CFR 50.39].	9	9

Technical information included in ESBWR design certification application:

PART 20--STANDARDS FOR PROTECTION AGAINST RADIATION

	Yes	No
20.1406 Minimization of contamination.		

PART 50--DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

	Yes	No
50.12 Specific exemptions.		
50.34 Contents of applications; technical information.		
50.34(a) Preliminary safety analysis report.		
50.34(b) Final safety analysis report.		
50.34(c) Physical security plan.		
50.34(d) Safeguards contingency plan.		
50.34(e) Protection against unauthorized disclosure.		
50.34(f) Additional TMI-related requirements.		
50.34(g) Combustible gas control.		
50.34(h) Conformance with the Standard Review Plan (SRP).		
50.34a Design objectives for equipment to control releases of radioactive material in effluents--nuclear power reactors.		
50.36 Technical specifications.		
50.36a Technical specifications on effluents from nuclear power reactors.		
50.44 Combustible gas control for nuclear power reactors.		
50.46 Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.		
50.46a Acceptance criteria for reactor coolant system venting systems.		
50.47 Emergency plans.		
50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.		

Attachment 1

	Yes	No
50.55a Codes and standards.		
50.61 Fracture toughness requirements for protection against pressurized thermal shock events.		
50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.		
50.63 Loss of all alternating current power.		
50.69 Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.		
Appendix A to Part 50--General Design Criteria for Nuclear Power Plants		
I. Overall Requirements:		
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		
II. Protection by Multiple Fission Product Barriers:		
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		
III. Protection and Reactivity Control Systems:		
20 Protection System Functions		
21 Protection System Reliability and Testability		

	Yes	No
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		
IV. Fluid Systems:		
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		
V. Reactor Containment:		
50 Containment Design Basis		
51 Fracture Prevention of Containment Pressure Boundary		

	Yes	No
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		
VI. Fuel and Radioactivity Control:		
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		
Appendix B to Part 50--Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants		
Appendix E to Part 50--Emergency Planning and Preparedness for Production and Utilization Facilities		
Appendix G to Part 50--Fracture Toughness Requirements		
Appendix H to Part 50--Reactor Vessel Material Surveillance Program Requirements		
Appendix I to Part 50--Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents		
Appendix K to Part 50--ECCS Evaluation Models		
Appendix S to Part 50--Earthquake Engineering Criteria for Nuclear Power Plants		

PART 73--PHYSICAL PROTECTION OF PLANTS AND MATERIALS

	Yes	No
73.55 Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage.		

Additional issues to be addressed in ESBWR design certification application:

I. SECY-90-016 Issues:

A. Use of a Physically Based Source Term
B. Anticipated Transient Without Scram
D. Station Blackout
E. Fire Protection
F. Intersystem Loss-of-Coolant Accident
G. Hydrogen Control
H. Core Debris Coolability
I. High-Pressure Core Melt Ejection
J. Containment Performance
K. Dedicated Containment Vent Penetration
L. Equipment Survivability
M. Elimination of Operating-Basis Earthquake
N. Inservice Testing of Pumps and Valves

II. Other Evolutionary and Passive Design Issues (SECY-93-087):

A. Industry Codes and Standards
B. Electrical Distribution
C. Seismic Hazard Curves and Design Parameters
D. Leak-Before-Break
E. Classification of Main Steamlines in Boiling Water Reactors
F. Tornado Design Basis
G. Containment Bypass
H. Containment Leak Rate Testing
I. Post-Accident Sampling System
J. Level of Detail
K. Prototyping
L. ITAAC
M. Reliability Assurance Program
N. Site-Specific Probabilistic Risk Assessments and Analysis of External Events
O. Severe Accident Mitigation Design Alternatives
P. Generic Rulemaking Related to Design Certification
Q. Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems
S. PRA Beyond Design Certification
T. Control Room Annunciator (Alarm) Reliability

III. Issues Limited to Passive Designs (SECY-93-087):
A. Regulatory Treatment of Nonsafety Systems in Passive Designs
B. Definition of Passive Failure
C. SBWR Stability
D. Safe Shutdown Requirements
E. Control Room Habitability
F. Radionuclide Attenuation
G. Simplification of Offsite Emergency Planning
H. Role of the Passive Plant Control Room Operator

IV. Confirmatory Items from NRC safety evaluation report for General Electric topical report NEDC-33083P regarding the application of TRACG Code to ESBWR LOCA analyses:

The applicant is to address the confirmatory items contained in the NRC safety evaluation report contained in NRC letter dated October 28, 2004, ADAMS Accession Number ML043000285.

V. Incorporate Operating Experience Into Design:

Operating experience is to be addressed in the design as requested by SRMs dated July 31, 1989, February 15, 1991, and March 5, 1991. This includes the operating experience discussed in NRC Bulletins (BLs) and Generic Letters (GLs). At a minimum, an ESBWR design certification application should address the BLs and GLs listed below:

Bulletins:

BL 79-02r2*	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts
BL 79-08	Events Relevant to Boiling Water Reactors Identified During Three Mile Island Incident
BL 80-01	Operability of ADS Valve Pneumatic Supply
BL 80-03	Loss of Charcoal from Standard Type II, 2 Inch, Tray Absorber Cells
BL 80-05	Vacuum Condition Resulting in Damage to Chemical Volume Control System (CVCS) Holdup Tanks
BL 80-06	Engineered Safety Feature (ESF) Reset Controls
BL 80-08	Examination of Containment Liner Penetration Welds
BL 80-10	Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment
BL 80-12	Decay Heat Removal System Operability
BL 80-13	Cracking in Core Spray Spargers
BL 80-15	Possible Loss of Emergency Notification System with Loss of Offsite Power
BL 80-20	Failures of Westinghouse Type W-2 Spring Return To Neutral Control Switches
BL 80-21	Valve Yokes Supplied by Malcolm Foundry Company
BL 80-22	Automatic Industries, Model 200-500-008 Sealed Source Connectors
BL 80-24	Prevention of Damage Due To Water Leakage Inside Containment

BL 80-25 Operating Problems With Target Rock Safety-Relief Valves at BWRs

* r = Revision

BL 81-01 Surveillance of Mechanical Snubbers

BL 81-02 Failure of Gate Type Valves to Close Against Differential Pressure

BL 81-02s1* Failure of Gate Type Valves to Close Against Differential Pressure

BL 81-03 Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. (Asiatic Clam) and Mytilus Sp. (Mussel)

BL 82-04 Deficiencies in Primary Containment Electrical Penetration Assemblies

BL 83-06 Nonconforming Materials Supplied by Tube-Line Corporation Facilities

BL 84-01 Cracks in Boiling Water Reactor Mark 1 Containment Vent Headers

BL 84-03 Refueling Cavity Water Seal

BL 85-03 Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings

BL 85-03s1 Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings

BL 86-01 Minimum Flow Logic Problems That Could Disable RHR Pumps

BL 86-03 Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line

BL 87-01 Thinning of Pipe Walls in Nuclear Power Plants

BL 87-02 Fastener Testing to Determine Conformance with Applicable Material Specifications

BL 87-02s1 Fastener Testing to Determine Conformance with Applicable Material Specifications

BL 87-02s2 Fastener Testing to Determine Conformance with Applicable Material Specifications

BL 88-04 Potential Safety-Related Pump Loss

BL 88-07 Power Oscillations in Boiling Water Reactors

BL 88-07s1 Power Oscillations in Boiling Water Reactors

BL 90-01 Loss of Fill-Oil in Transmitters Manufactured by Rosemount

BL 90-01s1 Loss of Fill-Oil in Transmitters Manufactured by Rosemount

* s = Supplement

BL 90-02 Loss of Thermal Margin Caused by Channel Box Bow

BL 91-01 Reporting Loss of Criticality Safety Controls

BL 91-01s1 Reporting Loss of Criticality Safety Controls

BL 92-01 Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage

BL 92-01s1 Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function

BL 93-02 Debris Plugging of Emergency Core Cooling Suction Strainers

BL 93-02s1 Debris Plugging of Emergency Core Cooling Suction Strainers

BL 93-03 Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs

BL 94-01 Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1

BL 95-02 Unexpected Clogging of a Residual Heat Removal Pump Strainer While Operating in Suppression Pool Cooling Mode

BL 96-02 Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment

BL 96-03 Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors

BL 2005-02 Emergency Preparedness and Response Actions for Security-Based Events

Generic Letters:

GL 80-34 Clarification of NRC Requirements for Emergency Response Facilities at Each Site

GL 80-113 Control of Heavy Loads

GL 81-03 Implementation of NUREG-0313, Rev. 1, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping

GL 81-04 Emergency Procedures and Training for Station Blackout Events

GL 81-07 Control of Heavy Loads

GL 81-10 Post-TMI Requirements for the Emergency Operations Facility

GL 81-11 Comments on NUREG-0619

GL 81-20	Safety Concerns Associated with Pipe Breaks in the BWR Scram System
GL 81-37	ODYN Code Reanalysis Requirements
GL 81-38	Storage of Low-Level Radioactive Wastes at Power Reactor Sites
GL 82-09	Environmental Qualification of Safety-Related Electrical Equipment
GL 82-21	Technical Specifications for Fire Protection Audits
GL 82-23	Inconsistency Between Requirements of 10 CFR 73.40(d) and Standard Technical Specifications for Performing Audits of Safeguards Contingency Plans (Security Plan)
GL 82-27	Transmittal of NUREG-0763, “Guidelines for Confirmatory In-Plant Tests of Safety-elief Valve Discharges for BWR Plants,” and NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments”
GL 82-33	Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability
GL 82-39	Problems with the Submittals of 10 CFR 73.21 Safeguards Information for Licensing Review
GL 83-05	Safety Evaluation of “Emergency Procedure Guidelines, Revision 2,” NEDO-24934, June 1982
GL 83-07	The Nuclear Waste Policy Act of 1982
GL 83-13	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems
GL 83-28	Required Actions Based on Generic Implications of Salem ATWS Events
GL 83-33	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50
GL 84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability
GL 84-23	Reactor Vessel Water Level Instrumentation in BWRs
GL 86-10	Implementation of Fire Protection Requirements
GL 87-06	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves
GL 87-09	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements
GL 88-01	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping
GL 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment
GL 88-15	Electric Power Systems - Inadequate Control Over Design Processes

GL 88-16 Removal of Cycle-Specific Parameter Limits from Technical Specifications

GL 88-18 Plant Record Storage on Optical Disks

GL 88-20 Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)

GL 88-20s1 Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)

GL 88-20s2 Accident Management Strategies for Consideration in the Individual Plant Examination Process

GL 88-20s3 Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities

GL 88-20s4 Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)

GL 88-20s5 Individual Plant Examination of External Events for Severe Accident Vulnerabilities

GL 89-01 Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program

GL 89-02 Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products

GL 89-04 Guidance on Developing Acceptable Inservice Testing Programs

GL 89-04s1 Guidance on Developing Acceptable Inservice Testing Programs

GL 89-06 Task Action Plan Item I.D.2 - Safety Parameter Display System - 10 CFR 50.54(f)

GL 89-07 Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs

GL 89-07s1 Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs

GL 89-08 Erosion/Corrosion-Induced Pipe Wall Thinning

GL 89-10 Safety-Related Motor-Operated Valve Testing and Surveillance

GL 89-10s1 Results of the Public Workshops

GL 89-10s3 Consideration of the Results of NRC Sponsored Tests of Motor-Operated Valves

GL 89-10s4 Consideration of Valve Mispositioning in Boiling Water Reactors

GL 89-10s5 Inaccuracy of Motor-Operated Valve Diagnostic Equipment

GL 89-10s6 Information on Schedule and Grouping, and Staff Responses to Additional Public Questions

GL 89-13 Service Water System Problems Affecting Safety-Related Equipment

GL 89-13s1 Service Water System Problems Affecting Safety-Related Equipment

GL 89-14 Line Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals

GL 89-15 Emergency Response Data System

GL 89-16 Installation of a Hardened Wetwell Vent

GL 89-18 Resolution of Unresolved Safety Issue A-17, “Systems Interactions in Nuclear Power Plants”

GL 89-19 Request for Action Related to Resolution of Unresolved Safety Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants,” Pursuant to 10 CFR 50.54(f)

GL 89-22 Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed By the National Weather Service

GL 90-09 Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions

GL 91-03 Reporting of Safeguards Events

GL 91-04 Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle

GL 91-05 Licensee Commercial-Grade Procurement and Dedication Programs

GL 91-06 Resolution of Generic Issue A-30, “Adequacy of Safety-Related DC Power Supplies,” Pursuant to 10 CFR 50.54(f)

GL 91-10 Explosives Searches at Protected Area Portals

GL 91-11 Resolution of Generic Issues 48, “LCOs for Class 1E Vital Instrument Buses,” and 49, ”Interlocks and LCOs for Class 1E Tie Breakers” Pursuant to 10 CFR 50.54(f)

GL 91-14 Emergency Telecommunications

GL 91-16 Licensed Operators’ and Other Nuclear Facility Personnel Fitness for Duty

GL 91-17 Generic Safety Issue 29, “Bolting Degradation or Failure in Nuclear Power Plants”

GL 92-01r1 Reactor Vessel Structural Integrity

GL 92-04	Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)
GL 92-08	Thermo-Lag 330-1 Fire Barriers
GL 93-06	Research Results on Generic Safety Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas"
GL 94-02	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors
GL 94-03	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors
GL 95-07	Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves
GL 96-01	Testing of Safety-Related Logic Circuits
GL 96-04	Boraflex Degradation in Spent Fuel Pool Storage Racks
GL 96-05	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves
GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
GL 96-06s1	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps
GL 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
GL 99-02	Laboratory Testing of Nuclear- Grade Activated Charcoal
GL 2003-01	Control Room Habitability