Appendix B

Partial Listing of Generic Communications Involving Design-Basis Issues

Partial Listing of Generic Communications Involving Design-Basis Issues

YEAR	GENERIC COMMUNICATION	TITLE
1997	Generic Letter (GL) 96-06, Supplement 1	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
1997	GL 97-04	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps
1997	Information Notice (IN) 97-81	Deficiencies in Failure Modes and Effects Analyses for Instrumentation and Control System
1997	IN 97-79	Potential Inconsistency in Assessment of Radiological Consequences of Main Steam Line Break Associated with Implementation OD SG Tube Voltage Based Repair Criteria
1997	IN 97-76	Degraded Throttle Valves in Emergency Core Cooling System Resulting from Cavitation-induced Erosion During a Loss-Of-Coolant Accident
1997	IN 97-71	Inappropriate Use of 10 CFR 50.59 Regarding Reduced Seismic Criteria for Temporary Conditions
1997	IN 97-60	Incorrect Unreviewed Safety Question Determination Related to Emergency Core Cooling System Swapover from the Injection Mode to the Recirculation Mode
1997	IN 91-50, Supplement 1	Water Hammer Events since 1991
1997	IN 97-43	License Condition Compliance
1997	IN 97-41	Potentially Undersized Emergency Diesel Generator (EDG) Oil Coolers
1997	IN 97-33	Unanticipated Effect of Ventilation System on Tank Level Indications and Engineering Safety Features Actuation System Setpoint
1997	IN 97-27	Effect of Incorrect Strainer Pressure Drop on Available Net Positive Suction Head
1997	IN 87-10, Supplement 1	Potential for Water Hammer During Restart of Residual Heat Removal Pumps
1997	IN 97-25	Dynamic Range Uncertainties in the Reactor Vessel Level Instrumentation
1997	IN 97-21	Availability of Alternate AC Source Designed for Station Blackout Event
1997	IN 97-13	Deficient Conditions Associated with Protective Coatings at Nuclear Power Plants
1997	IN 97-07	Problems Identified During Generic Letter 89-10 Closeout Inspections
1996	GL 96-06	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions
1996	GL 96-05	Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1996	GL 96-01	Testing of Safety-Related Logic Circuits
1996	Bulletin (IEB) 96-03	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors
1996	IN 96-64	Modifications to Containment Blowout Panels Without Appropriate Design Controls
1996	IN 96-60	Potential Common-Mode Post-accident of Residual Heat Removal Heat Exchangers
1996	IN 96-55	Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps under Design Basis Accident Conditions
1996	IN 96-49	Thermally Induced Pressurization of Nuclear Power Facility Piping
1996	IN 96-45	Potential Common-Mode Post-accident Failure of Containment Coolers
1996	IN 96-41	Effects of a Decrease in Feedwater Temperature on Nuclear Instrumentation
1996	IN 96-39	Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly
1996	IN 96-36	Degradation of Cooling Water Systems Due to Icing
1996	IN 96-31	Cross-tied Safety Injection Accumulators
1996	IN 96-27	Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation
1996	IN 96-17	Reactor Operation Inconsistent with the Updated Final Safety Analysis Report
1996	IN 96-08	Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve
1996	IN 96-06	Design and Testing Deficiencies of Tornado Dampers at Nuclear Power Plants
1996	IN 96-01	Potential for High Post-Accident Closed-Cycle Cooling Water Temperatures to Disable Equipment Important to Safety
1995	GL 95-07	Pressure Locking and Thermal Binding of Safety-related Power-Operated Gate Valves
1995	GL 92-01, Revision 1, Supplement 1	Reactor Vessel Structural Integrity
1995		“Generic LTR 89-10 Design-Basis Closure Millstone Unit 3"
1995		“GL 89-10 Close-out CYAP Haddam Neck Plant”

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1995	IEB 95-02	Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode
1995	IN 91-29, Supplement 3	Deficiencies Identified During Electrical Distribution System Functional Inspections
1995	IN 95-47, Revision 1	Unexpected Opening of a Safety/relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage
1995	IN 95-47	Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage
1995	IN 95-37	Inadequate Offsite Power System Voltages During Design-Basis Events
1995	IN 95-30	Susceptibility of Low-pressure Coolant Injection and Core Spray Injection Valves to Pressure Locking
1995	IN 95-28	Emplacement of Support Pads for Spent Fuel Dry Storage Installations at Reactor Sites
1995	IN 95-18, Supplement 1	Potential Pressure-locking Safety-Related Power-operated Gate Valves
1995	IN 95-18	Potential Pressure-Locking of Safety-Related Power-Operated Gate Valves
1995	IN 95-16	Vibration Caused by Increased Recirculation Flow in a Boiling Water Reactor
1995	IN 95-14	Susceptibility of Containment Sump Recirculation Gate Valves to Pressure Locking
1995	IN 95-11	Failure of Condensate Piping Because of Erosion/Corrosion at a Flow-Straightening Device
1995	IN 95-10, Supplement 1	Potential for Loss of Automatic Engineered Safety Features Actuation
1995	IN 95-10	Potential for Loss of Automatic Engineered Safety Features Actuation
1995	IN 95-09	Use of Inappropriate Guidelines and Criteria for Nuclear Piping and Pipe Support Evaluation and Design
1995	IN 95-06	Potential Blockage of Safety-Related Strainers by Materials Brought Inside Containment
1994	GL 94-02	Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors
1994	IEB 93-02, Supplement 1	Debris Plugging of Emergency Core Cooling Suction Strainers
1994	IN 94-82	Concerns Regarding Essential Chiller Reliability During Periods of Low Cooling Water Temperature
1994	IN 94-76	Recent Failures of Charging/Safety Injection Pump Shafts
1994	IN 94-64	Reactivity Insertion Transient and Accident Limits for High Burnup Fuel

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1994	IN 94-60	Potential Overpressurization of Main Steam System
1994	IN 94-32	Revised Seismic Hazard Estimates
1994	IN 94-27	Facility Operating Concerns Resulting from Local Area Flooding
1994	IN 93-17, Revision 1	Safety Systems Response to Loss of Coolant and Loss of Offsite Power
1994	IN 94-20	Common-Cause Failures Due to Inadequate Design Control and Dedication
1994	IN 92-36, Supplement 1	Intersystem LOCA Outside Containment
1994	IN 94-03	Deficiencies Identified During Service Water System Operational Performance Inspections
1993	IEB 93-02	Debris Plugging of Emergency Core Cooling Suction Strainers
1993	IN 89-077, Supplement 1	Debris in Containment Emergency Sumps & Incorrect Screen Configurations
1993	IN 91-29, Supplement 2	Potential Deficiencies Found During Electrical Distribution System Functional Inspections
1993	IN 93-99	Undervoltage Relay and Thermal Overload Setpoint Problems
1993	IN 93-66	Switchover to Hot-Leg Injection Following a Loss-Of-Coolant Accident in Pressurized Water Reactors
1993	IN 93-55	Potential Problem with Main Steamline Break Analysis for Main Steam Vaults/Tunnels
1993	IN 93-46	Potential Problem with Westinghouse Rod Control System and Inadvertent Withdrawal of a Single Rod Control Cluster Assembly
1993	IN 93-34, Supplement 1	Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment
1993	IN 93-34	Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment
1993	IN 93-28	Failure to Consider Loss of DC Bus in the Emergency Core Cooling System Evaluation May Lead to Nonconservative Analysis
1993	IN 93-17	Safety Systems Response to Loss of Coolant and Loss Of Offsite Power
1993	IN 93-13	Undetected Modification of Flow Characteristics in the High Pressure Safety Injection System
1993	IN 93-11	Single Failure Vulnerability of Engineered Safety Features Actuation Systems
1992	GL 92-04	Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1992	GL 92-03	Compilation of the Current Licensing Basis: Request for Voluntary Participation in Pilot Program
1992	GL 92-01, Revision 1	Reactor Vessel Structural Integrity, 10 CFR 50.54(f)
1992	IN 92-74	Power Oscillations at Washington Nuclear Power Unit 2
1992	IN 92-71	Partial Plugging of Suppression Pool Strainers at a Foreign BWR
1992	IN 91-29, Supplement 1	Deficiencies Identified During Electrical Distribution System Functional Inspections
1992	IN 92-65	Safety System Problems Caused by Modifications That Were Not Adequately Reviewed and Tested
1992	IN 91-52, Supplement 1	Nonconservative Errors in Overtemperature Delta-temperature (OT Delta T) Setpoint Caused by Improper Gain Settings
1992	IN 92-41	Consideration of the Stem Rejection Load in Calculation of Required Valve Thrust
1992	IN 92-21	Spent Fuel Pool Reactivity Calculations
1992	IN 92-02, Supplement 1	Relap5/Mod3 Computer Code Error Associated with the Conservation of Energy Equation
1992	IN 92-02	Relap5/Mod3 Computer Code Error Associated with the Conservation of Energy Equation
1991	GL 91-13	Request for Information Related to the Resolution of Generic Issue 130, "Essential Service Water System Failures at Multi-unit Sites," Pursuant to 10 CFR 50.54(f)
1991	GL 91-06	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies," Pursuant to 10 CFR 50.54(f)
1991	IN 91-75	Static Head Corrections Mistakenly Not Included in Pressure Transmitter Calibration Procedures
1991	IN 91-69	Errors in Main Steam Line Break Analyses for Determining Containment Parameters
1991	IN 91-50	A Review of Water Hammer Events after 1985
1991	IN 91-29	Deficiencies Identified During Electrical Distribution Systems Functional Inspections
1991	IN 91-12	Potential Loss of Net Positive Suction Head of Standby Liquid Control Sys Pumps
1991	IN 91-11	Inadequate Physical Separation and Electrical Isolation of Non-Safety Related Circuits from Reactor Protection System Circuits

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1990	GL 90-06	Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," Pursuant to 10 CFR 50.54(f) (Generic Letter)
1990	GL 90-04	Request for Information on the Status of Licensee Implementation of Generic Safety Issues Resolved with Imposition of Requirements or Corrective Actions
1990	GL 89-13, Supplement 1	Service Water System Problems Affecting Safety-Related Equipment
1990	IN 90-78	Previously Unidentified Release Path from Boiling Water Reactor Control Rod Hydraulic Units
1990	IN 88-23, Supplement 3	Potential for Gas Binding of High-pressure Safety Injection Pumps During a Loss-Of-Coolant Accident
1990	IN 89-30, Supplement 1	High Temperature Environments at Nuclear Power Plants
1990	IN 90-64	Potential for Common-mode Failure of High Pressure Safety Injection Pumps or Release of Reactor Coolant Outside Containment During a Loss-Of-Coolant Accident
1990	IN 90-61	Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction
1990	IN 90-53	Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment
1990	IN 90-26	Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems
1989	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
1989	GL 89-21	Request for Information Concerning Status of Implementation of Unresolved Safety Issue (USI) Requirements (Generic Letter)
1989	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)
1989	GL 89-18	Resolution of Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants"
1989	GL 89-16	Installation of a Hardened Wetwell Vent
1989	GL 89-13	Service Water System Problems Affecting Safety-Related Equipment
1989	GL 89-11	Resolution of Generic Issue 101 "Boiling Water Reactor Water Level Redundancy"

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1989	IN 89-81	Inadequate Control of Temporary Modifications to Safety-Related Systems
1989	IN 89-77	Debris in Containment Emergency Sumps and Incorrect Screen Configurations
1989	IN 89-71	Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-Of-Coolant Accident
1989	IN 89-68	Evaluation of Instrument Setpoints During Modifications
1989	IN 89-63	Possible Submergence of Electrical Circuits Located above the Flood Level Because of Water Intrusion and Lack of Drainage
1989	IN 89-55:	Degradation of Containment Isolation Capability by a High-Energy Line Break
1989	IN 89-54	Potential Overpressurization of the Component Cooling Water System
1989	IN 89-50	Inadequate Emergency Diesel Generator Fuel Supply
1989	IN 89-48:	Design Deficiency in the Turbine-driven Auxiliary Feedwater Pump Cooling Water System
1989	IN 88-75, Supplement 1	Disabling of Generator Output Circuit Breakers by Anti-Pump Circuitry
1989	IN 89-36	Excessive Temperatures in Emergency Core Cooling System Piping Located Outside Containment
1989	IN 88-86, Supplement 1	Operating with Multiple Grounds in Direct Current Distribution Systems
1989	IN 89-30	High Temperature Environments at Nuclear Power Plants
1989	IN 89-29	Potential Failure of ASEA Brown Boveri Circuit Breakers During Seismic Event
1989	IN 89-28	Weight and Center of Gravity Discrepancies for Copes-Vulcan Air-Operated Valves
1989	IN 89-26	Instrument Air Supply to Safety-Related Equipment
1989	IN 89-16	Excessive Voltage Drop in DC Systems
1989	IN 89-11	Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cable Sizing
1989	IN 89-08	Pump Damage Caused by Low-flow Operation
1989	IN 88-23, Supplement 1	Potential for Gas Binding of High-Pressure Safety Injection Pumps During a Loss-Of-Coolant Accident
1988	GL 88-15	Electric Power Systems - Inadequate Control over Design Processes
1988	GL 88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1988	GL 88-11	NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations
1988	GL 88-03	Resolution of Generic Safety Issue 93, "Steam Binding of Auxiliary Feedwater Pumps"
1988	IEB 88-04	Potential Safety-Related Pump Loss
1988	IN 88-94	Potentially Undersized Valve Actuators
1988	IN 88-92	Potential for Spent Fuel Pool Draindown
1988	IN 88-86	Operating with Multiple Grounds in Direct Current Distribution Systems
1988	IN 88-80	Unexpected Piping Movement Attributed to Thermal Stratification
1988	IN 88-76	Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control
1988	IN 88-75	Disabling of Diesel Generator Output Circuit Breakers by Anti-Pump Circuitry
1988	IN 88-74	Potentially Inadequate Performance of ECCS in PWRs During Recirculation Operation Following a LOCA
1988	IN 88-72	Inadequacies in the Design of DC Motor-Operated Valves
1988	IN 88-61	Control Room Habitability - Recent Reviews of Operating Experience
1988	IN 88-60	Inadequate Design and Installation of Watertight Penetration Seals
1988	IN 88-55	Potential Problems Caused by Single Failure of an Engineered Safety Feature Swing Bus
1988	IN 88-50	Effect of Circuit Breaker Capacitance on Availability of Emergency Power
1988	IN 88-45	Problems in Protective Relay and Circuit Breaker Coordination
1988	IN 88-28	Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage
1988	IN 88-01	Safety Injection Pipe Failure
1987	GL 87-05	Request for Additional Information Assessment of Licensee Measures to Mitigate And/or Identify Potential Degradation of Mark I Drywells
1987	GL 87-03	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46
1987	GL 87-02	Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46
1987	IN 87-67	Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11

Partial Listing of Generic Communications Involving Design-Basis Issues (cont.)

YEAR	GENERIC COMMUNICATION	TITLE
1987	IN 87-28, Supplement 1	Air Systems Problems at U.S. Light Water Reactors
1987	IN 87-65	Plant Operation Beyond Analyzed Conditions
1987	IN 87-63	Inadequate Net Positive Suction Head in Low Pressure Safety Systems
1987	IN 87-59	Potential RHR Pump Loss
1987	IN 87-53	Auxiliary Feedwater Pump Trips Resulting from Low Suction Pressure
1987	IN 87-50	Potential LOCA at High- and Low-pressure Interfaces from Fire Damage
1987	IN 87-49	Deficiencies in Outside Containment Flooding Protection
1987	IN 87-34	Single Failures in Auxiliary Feedwater Systems
1987	IN 87-28	Air Systems Problems at U.S. Light Water Reactors
1987	IN 87-10	Potential for Water Hammer During Restart of Residual Heat Removal Pumps
1987	IN 87-09	Emergency Diesel Generator Room Cooling Design Deficiency
1987	IN 87-06	Loss of Suction to Low-Pressure Service Water System Pumps Resulting from Loss of Siphon
1987	IN 87-02	Inadequate Seismic Qualification of Diaphragm Valves by Mathematical Modeling and Analysis

Appendix C

NRC Guidance in Defining Operability and Functional Capability, Resolving Degraded or Nonconforming Conditions, and Using Risk Assessment Techniques in Assessing Design-Basis Issues

Operability and Functional Capability

The NRC Inspection Manual, Part 9900: Technical Guidance, "Operable/Operability: Ensuring the Functional Capability of a System or Component," defines several terms important to design-basis issues (DBIs).

(1) Current Licensing Basis (CLB). That set of NRC requirements applicable to a specific plant, and a licensee's written commitments for assuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.

(2) Nonconforming Condition. A condition of an structure, system or component (SSC) in which there is failure to meet requirements or licensee commitments.

(3) Operability. The Standard Technical Specifications (TS) define operable or operability as: "A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function, and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions(s) are also capable of performing their related support function(s)."

(4) Full Qualification. Full qualification constitutes conforming to all aspects of the CLB, including codes and standards, design criteria, and commitments.

(5) Consequential Failure. A consequential failure is a failure of an SSC caused by a postulated accident within the design basis.

Operability and qualification are closely related concepts. However, the fact that a system is not fully qualified does not, in all cases, render that system unable to perform its specified function if called upon. The prompt determination of operability will result in decisions or actions pertaining to continued plant operation, while qualification or requalification becomes a corrective action goal.

Operability determinations should be performed for those potential consequential failures [i.e., an SSC failure that would be a direct consequence of a design-basis event] for which the SSC in question needs to function. Where consequential failures would cause a loss of function needed for limiting or mitigating the effects of the event, the affected SSC is inoperable because it cannot perform all of its specified functions. Such situations are most likely discovered during design basis reconstitution studies, or when new credible failure modes are identified.

In probabilistic risk assessment (PRA), the parameter of interest is equipment availability, not operability. The two terms have a different basis. Operability is related to licensing specifications, while availability is related to operational performance requirements. If licensing assumptions are input into PRA, the results may significantly overestimate risk.

Resolution of Design-Basis Issues as Degraded and Nonconforming Conditions

The NRC Inspection Manual, Part 9900: Technical Guidance, “Resolution of Degraded and Nonconforming Conditions,” provides guidance to NRC inspectors on resolving degraded and nonconforming conditions affecting certain SSCs. This guidance indicates that upon discovery of an existing but previously unanalyzed condition that significantly compromises plant safety, the licensee shall report that condition in accordance with 10 CFR 50.72 and 50.73, and put the plant in a safe condition. Once a degraded or nonconforming condition has been identified, Part 9900 provides the following:

The license authorizes the licensee to operate the plant in accordance with the regulations, license conditions, and the TS. If an SSC is degraded or nonconforming but operable, the license establishes an acceptable basis to continue to operate and the licensee does not need to take any further actions. The licensee must, however, promptly identify and correct the condition adverse to safety or quality in accordance with 10 CFR 50, Appendix B, Criterion XVI (Corrective Actions).

For SSCs that are not expressly subject to TS and that are determined to be inoperable, the licensee should assess the reasonable assurance of safety. If the assessment is successful, then the facility may continue to operate while prompt corrective action is taken.

In its evaluation of the impact of a degraded or nonconforming condition on plant operation and on operability of SSCs, a licensee may decide to implement a compensatory measure as an interim step to restore operability or to otherwise enhance the capability of SSCs until the final corrective action is complete. Reliance on a compensatory measure for operability should be an important consideration in establishing the “reasonable time frame” to complete the corrective action process. NRC would normally expect that conditions that require interim compensatory measures to demonstrate operability would be resolved more promptly than conditions that are not dependent on compensatory measures to show operability, because such reliance suggests a greater degree of degradation. Similarly, if an operability determination is based upon operator action, NRC would expect the nonconforming condition to be resolved expeditiously.

The licensee may make mode changes, restart from outages, etc., provided that necessary equipment is operable and the degraded condition is not in conflict with the TS or the license.

The responsibility for corrective action rests on the licensee. A licensee’s range of corrective action could include (1) full restoration to the SAR-described condition, (2) NRC approval for a change to its licensing basis to accept the as-found condition as is, or (3) some modification of the facility other than restoration to the original FSAR condition.

Strengths and Limitations of Assessment Methodologies for Design-Basis Issues

Design-basis issues should be evaluated using analyses that are traditional (deterministic) and risk-based (PRA). Each of these analysis methodologies have flaws which may both underestimate or overestimate the end result. The confidence level in each method is highly dependent upon the data available, the scope and depth of the models, and the understanding of the users. The analyses results in both instances should be reviewed for accuracy and soundness, with a full understanding of their strengths and limitations.

In SECY-98-144, “White Paper on Risk-Informed and Performance-Based Regulation,” dated June 22, 1998, defines risk as a “risk triplet” composed of three questions, (1) What can go wrong?, (2) How likely is it?, and (3) What are the consequences? The traditional definition of risk, that is, probability times consequences, is fully embraced by the triplet definition of risk. The first question, “What can go wrong?” is usually answered in the form of a “scenario” (a combination of events and/or conditions that could occur) or a set of scenarios. The second question, “How likely is it?” can be answered in terms of the available evidence and the processing of that evidence to quantify the probability and the uncertainties involved. The third question, “What are the consequences?” can be answered for each scenario by assessing the probable range of outcomes given the uncertainties. The outcomes are the “end states” of the analyses.

The current body of regulations, guidance and license conditions is based largely on a “deterministic” approach. As described in the PRA Policy Statement, the deterministic approach to regulation establishes requirements for engineering margin and for quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist and establishes a specific set of design basis events (i.e., what can go wrong?). The deterministic approach involves implied, but unquantified, elements of probability in the selection of the specific accidents to be analyzed as design basis events. It then requires that the design include safety systems capable of preventing and/or mitigating the consequences (i.e., what are the consequences?) Of those design basis events in order to protect public health and safety. Thus, a deterministic approach explicitly addresses only two questions of the risk triplet.

A probabilistic approach to regulation considers risk in a more coherent, explicit, and quantitative manner. The probabilistic approach explicitly addresses a broad spectrum of initiating events and their event frequency. It then analyzes the consequences of those event scenarios and weights the consequences by the frequency, thus giving a measure of risk.

A “risk-informed” approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.

A “risk-based” approach to regulatory decision-making is one in which a safety decision is solely based on the numerical results of a risk assessment. This places heavier reliance on risk assessment results than may currently be practicable. Note that the Commission does not endorse an approach that is “risk-based”; however, this does not invalidate the use of calculations to demonstrate compliance with certain criteria, such as dose limits.

As stated in Part 9900, “Probabilistic risk assessment is a valuable tool for the relative evaluation of accident scenarios while considering, among other things, the probabilities of occurrence of accidents or external events. The definition of operability states, however, that the SSC must be capable of performing its specified function(s). The inherent assumption is that the occurrence conditions or event exists and that the safety function can be performed. The use of PRA or probabilities of the occurrence of accidents or external events is not acceptable for making operability decisions. However, PRA is a useful tool for determining the safety significance of SSCs. The safety significance, whether determined by PRA or other analyses, is a necessary factor in decisions on the appropriate ‘timeliness’ of operability determinations.”

Probabilistic risk assessment, like other disciplines, has a number of identifiable strengths and limitations. The strengths tend to be related to the fact that a PRA provides a rigorous, detailed means of addressing the complex issues of risk and reliability. The limitations are primarily related to the uncertainties which are inherent in many of the supporting disciplines. Utilization of PRA results can be effectively accomplished by application of PRA in those areas which most closely related to its strengths. However, useful information can also be gained in areas where PRA is limited, as long as those limitations are considered when interpreting the significance of that information. By fully recognizing the strengths and limitations, PRA analysts can attempt to capitalize on the strengths and address the limitations. In this respect it may be true that the nature of the limitations, in an absolute sense, is not as important as the recognition of those limitations.

Traditional PRAs are good at (1) identifying important accident sequences, and (2) identifying important equipment failures and human errors. Traditional PRAs are not so good at (1) absolute numbers, (2)human errors of commission, (3) design and construction errors, (4) low power or shutdown conditions, and (5) partial failures of SSCs.

A wide range of PRA capability exists in the industry. While some licensees have the capability to predict the risk involved with future plant maintenance activities and outages, and also the capability to evaluate the risk significance of past plant configurations (on-line risk monitors), others do not. The scope and quality of the engineering analyses (including traditional and probabilistic analyses) conducted should be based on the as-built and as-operated and maintained plant, including the reflection of operating experience at the plant to reduce uncertainties in the data and analyses results.

Appendix D

Significant 1997 Safety System Engineering Inspection Findings

Significant 1997 Safety System Engineering Inspection Findings

Plant	Design Findings Contained in Report Forwarding Letter
Arkansas Nuclear One Unit 1	The team identified an issue regarding excessive emergency feedwater (EFW) flowrates to a single steam generator. To reduce steam generator tube vibration crossflow velocity, Framatome Technologies, Inc. (FTI), in a 1991 report to ANO-1, recommended a maximum flow of 1500 gpm assuming both pumps are available and one steam generator is isolated. However, ANO-1 plant operating procedures have no provisions to monitor and preclude exceeding this limit. In May 1996, ANO-1 experienced a transient in which peak EFW flowrates of 1716 gpm were identified for a brief time. An analysis performed by FTI, together with the results of your staff's steam generator tube inspection performed during the last outage, suggest that there is no immediate operability concern. The NRR staff will review and evaluate the plant specific and potentially generic aspects of this issue.
	The team identified issues associated with an operability evaluation performed for the borated water storage tank flange removal which did not account for the installation of a foreign materials exclusion cover and did not adequately address radioactive releases rom the tank.
	Other findings included and inadequate evaluation for non-"Q" steam traps which, if failed, could significantly alter the EFW pump room environment; not periodically testing certain molded case circuit breakers; not establishing a basis for determining design requirements for the installation of instrument tubing and sensing lines that were found to be inadequately supported; inadequate control of some field-routed conduits; a vortexing calculation which did not account for instrument error, and discrepancies in the final safety analysis report.
D.C. Cook Units 1&2	Revisions made in 1992 to the emergency operating procedure for the manual swapover from the refueling water storage tank to the containment recirculation sump during a loss-of-coolant accident (LOCA) created a single failure vulnerability that potentially could have caused both trains of the centrifugal charging and safety injection pumps to be inoperable.
	Operational changes after 1988, permitted the plant to operate above the design basis ultimate heat sink temperature of 76 °F without your staff having performed a 10 CFR Part 50.59 evaluation, and without considering the impact this would have on overall plant operation. As a result, an apparent unreviewed safety question and unanalyzed condition was created in 1988, when the plant operated for 22 days with an averaged ultimate heat sink temperature of 81 °F, creating the potential for safety-related equipment in the control room to not perform its safety function under design basis assumptions.
	The licensee documented in a letter to the NRC, dated December 29, 1978, containment sump enhancement modifications that consisted of installing five 3/4-inch vent holes in the roof of the containment recirculation sump. However, the updated final safety analysis report (UFSAR) was not updated to reflect these changes, and the vent holes were in excess of the 1/4-inch sump particulate retention design basis value. In addition, these vents were sealed in 1996 and 1997 without performed a 10 CFR Part 50.59 evaluation, and without an adequate understanding of the commitment made to the NRC to maintain vents in the containment recirculation sump.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
D.C. Cook Units 1&2 (Continued)	During the Unit 2 1996 refueling outage, both component cooling water (CCW) and emergency service water (ESW) trains were removed from service contrary to the assumptions contained in Chapter 9 of the UFSAR, with the intention by your staff of performing a dual CCW/ESW train outage. Although the dual train outage was not fully sustained as originally planned by your staff, this operational condition would have placed the plant at increased risk, outside of its design basis, and in an unanalyzed condition.
	Although [most] items listed above had been known and documented by your staff, no apparent effective action was taken to correct the problems or their root causes. The team concluded that a contributing cause to these issues and others identified in the enclosed report was that prior to this inspection, your staff had an apparent lack of understanding of what constitutes the plant's design basis, the role of the UFSAR, and how each of these are affected by 10 CFR Part 50.59.
	The team also identified examples involving: (1) failure to account for instrument bias and establish the proper refueling water storage tank (RWST) and containment level setpoints necessary to preclude premature manual RWST switchover and subsequent potential vortexing in the containment , (2) failure to remove fibrous insulation materials from containment cable trays, that could potentially be swept into and block in excess of the design value of 50 percent of the containment recirculation sump screen area, and (3) the creation of a common-mode failure vulnerability that could potentially clog redundant trains of emergency core cooling system (ECCS) throttle injection valves and containment spray nozzles.
	On September 8, 1997, your staff initiated a dual unit shutdown, and issued a notification of an unusual event (NOUE), as a result of the inability to demonstrate to the team that the ECCS system would have performed its safety function during post-LOCA conditions under all postulated accident scenarios. On September 19, 1997, the NRC issued a confirmatory action letter listing many of the issues identified during this inspection.
Cooper	The team identified that the design change to the reactor equipment cooling (REC) system for the installation of the filter demineralizer in 1991, the associated safety analysis, and the operating procedure did not address the importance of maintaining water inventory in the closed REC system. The REC system would not have been able to support its long-term cooling functions in the event of a design basis accident, because the minimum available volume of water in the surge tank would have been depleted within a day through the sampling valves that were left open apparently since the modification was installed in 1991. Your staff isolated the sampling valves, notified the NRC of the condition, and issued LER 97-014 on December 12, 1997, which identified the cause as a failure to understand the design basis functions of the system.
	Although many calculations reviewed by the team were satisfactory, the team noted that nonconservative assumptions and design inputs were used in the calculations for estimating the residual heat removal (RHR) pump room temperature and for verifying the capability of the service water (SW) system to provide adequate back-up cooling for safety-related equipment in the REC system. A night order was issued to secure one of the RHR pumps if the fan coil unit in that room becomes inoperable, and SW back-up cooling calculation was revised.
Cooper (Continued)	The 10 CFR 50.59 safety evaluation that was performed for the updated safety analysis report (USAR) revision to increase the residual heat removal service water (RHRSW) booster pump room temperature limit to 131 °F did not address the consequences of operator actions required during post-accident conditions to prevent exceeding this temperature limit.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
	The effects of failure of air pressure regulators in the instrument air system on air operated valves had not been evaluated. At the exit meeting, we urged you to expedite this investigation and promptly perform operability evaluations as required.
	Previous NRC inspections had identified weaknesses in factoring instrument uncertainties into test acceptance criteria and operating procedures. The team noted that the procedure for monitoring SW temperature and the surveillance test procedure for RHR pumps did not consider applicable instrument uncertainties.
	The team also identified other issues, such as: weaknesses in performance monitoring of RHR and REC heat exchangers; and inadequate reportability review of a deficiency in the design of power sources to RHR heat exchanger vent valves; not including in operating procedures vendor recommended limitations on RHR pump operation at low flows; and not considering the potential for pumping post-accident leakage from ECCS to the radwaste system. In addition, the team has referred four issues identified in the report to the NRR staff for evaluation.
	The team noted several discrepancies in the USAR, technical specification, and system design criteria documents. The design criteria documents (DCD-13) for the RHR system contained several incorrect statements that were inconsistent with the current system design.
	Some of the deficiencies discussed above challenged the capability of the systems to perform their full design bases functions. The contributory causes for these deficiencies appear to be a lack of understanding of the design bases of the systems, use of nonconservative assumptions and design inputs in calculations, and not maintaining control over the configuration of the design bases reflected in various plant documents. Where appropriate, your staff took immediate corrective or compensatory actions to ensure system operability. For other issues, you have initiated problem identification reports to address required corrective actions. Taking into consideration your immediate actions, the team concluded at the end of the inspection, that both systems were capable of performing their safety functions.
Davis-Besse	The team identified some weaknesses in the design process and installation of the systems. For example: reverse flow testing of check valves DH81 and DH82 on the two low-pressure injection pump suction lines from the borated water storage tank (BWST) and seat leakage testing of stop check valves HP32 and HP 32 on the high-pressure injection pump recirculation lines were not done; and although your staff had identified the deterioration of the BWST level transmitter support hardware in 1994, no action was taken to remove, examine, and replace the hardware until the inspection team expressed its concern on the condition of the supports. The team referred to NRR staff for evaluation the issue of the acceptability of the use of normally open safety-related valves as safety class interface between the high-pressure injection system and local pressure gauges that were not seismically qualified.
Davis-Besse (Continued)	Other issues identified by the team included updated safety analysis report discrepancies, weaknesses in periodic testing of battery chargers, lack of testing of inverters, not including certain electrical components in the environmental qualification program, and not revising and installation detail drawing after modifying the actuators for decay heat removal cooler outlet and bypass valves.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
Diablo Canyon Units 1&2	Two issues identified may represent potential unreviewed safety questions and an additional NRC evaluation is ongoing. One issue involves the single failure design of the component cooling water, auxiliary salt water (ASW), and the residual heat removal (RHR) systems. Because of the design of the electrical distribution system, these systems are operated with both trains cross-tied. The resultant single train systems are vulnerable to passive failure when cross-tied and to active failures when the trains are split. The second issue involves the availability of the containment spray function during containment recirculation. Both issues were previously identified and evaluated by Pacific Gas and Electric Company (PG&E) staff. The evaluations resulted in compensatory administrative actions, which involved changing emergency operating procedures and assignment of manual functions to operating and emergency response staff.
	Issues were identified with the current ASW pump testing method that results in pump and heat exchanger unavailability. PG&E staff are pursuing changes to the current test method to improve system availability. Additionally, the ASW system supply path from the demusseling line is credited in the UFSAR since the single ASW intake bay screen is not seismically qualified. However, this alternate supply line is not being maintained or tested. PG&E's response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and its actions to implement the maintenance rule did not resolve this issue.
	Some design calculation weaknesses were identified, although they did not affect the overall results of the calculations. They involved updating and control of calculations, and the use of nonconservative assumptions. In addition, the team identified discrepancies and inconsistencies in the updated final safety analysis report, procedures, design criteria memorandum, calculations, drawings, and other documents.
Farley Units 1 and 2	The team had concerns with inadequate tornado missile protection of the turbine-driven auxiliary feedwater (TDAFW) pump vent stack and the exposed piping connections, level transmitter, electrical conduits and cables of the condensate storage tank (CST). The as-built plant configuration for these did not conform to the Farley design and licensing bases. In addition, the exhaust silencers for the diesel generators (including the station blackout diesels) were not protected against vertical and other non-horizontal missiles. The NRR staff will review this issue associated with the diesel generators to determine whether the tornado missile protection in the Farley Unit 1 and 2 design and licensing bases included missile spectra other than horizontal missiles.
Farley Units 1 and 2 (Continued)	Evaluations of plant modifications, conducted in accordance with 10 CFR 50.59, were generally adequate. However, the team identified certain examples of inadequate safety evaluations. For example, the 10 CFR 50.59 evaluation for a final safety analysis report change deleting the requirement for tornado missile protection for several CST piping connections did not identify a potential unreviewed safety question. Your staff evaluated this issue, notified the NRC in accordance with 10 CFR 50.72, on February 27, 1997, and implemented interim corrective actions to maintain the operability of the system until the issue can be resolved.
	The team noted design control issues for calculations, as well as nonconservative assumptions and inputs in calculations. In addition, the team identified discrepancies between the final safety analysis report and other documents, such as procedures, functional system descriptions, calculations, and drawings.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
Ginna	Some discrepancies were identified regarding adherence of the systems to their design and licensing bases. For example, the team found that: the updated final safety analysis report had not been updated to reflect changes in the peak clad temperature calculated to occur during a design basis accident; certain safety related valves were not being tested to ensure functionality; and instructions in the Emergency Operating Procedures were not clear regarding the sequence of steps necessary to insure a successful post-loss-of-coolant accident switchover from injection to recirculation. Also, the level of review of the loss-of-coolant accident analyses was found to be insufficient, as evidenced by several errors and inconsistencies identified by the team during the inspection.
Palisades	The team identified deficiencies in the control and performance of calculations. These deficiencies involved not updating the calculations when analytical inputs were changed; errors in some calculations; failure to specify uncertainty values in instrument setpoint calculations; a calculation which contained inadequate analysis to support the conclusion; and a dc short circuit calculation issued without verifying all input parameters or providing any conclusion on the acceptability of the dc system.
	The team identified many inconsistencies between the installed configurations of instrument tubing and the design basis in the component cooling water (CCW) and safety injection (SI) systems. As a result of these inconsistencies, the team had concerns with potential air entrapment into the instrument sensing lines for the high and low head SI flow transmitters.
	The team had questions on some calculations for which the adequacy of the design basis could not be verified. For example, no analysis was available to demonstrate that the dc loads would operate at the minimum battery voltage stated in the final safety analysis report; no analysis was available to demonstrate adequate ac voltage at the 120 volt safety-related loads; and no analysis was available to demonstrate that the battery could carry all required dc loads during a design-basis accident with the battery chargers cross-connected.
Perry Unit 1	The team identified three examples where the facility was being operated or maintained differently than described in the updated safety analysis report (USAR) or in vendor's design input information. Two of these examples were not supported by a written safety or engineering evaluation. The first example involved continuous operation of the suppression pool cleanup system and the second example related to the improper setting of the governor speed droop for the division III emergency diesel generator. These examples have a direct impact on the high-pressure core spray (HPCS) system performance. The third example, determined by the NRC to be a potential unreviewed safety question, resulted from an inadequate 10 CFR 50.59 safety evaluation for a change involving early use of operator action to fill the emergency core cooling system surge tank in a post loss-of-coolant accident environment.
	The team noted design program weaknesses, including updating and control of calculations, and nonconservative assumptions and inputs to calculations. In addition, the team identified discrepancies and inconsistencies in the USAR, procedures, system description manuals, calculations, drawings, and other documents.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
	The team identified concerns with tornado missile protection of the HPCS and reactor core isolation coolant suction piping from the condensate storage tank and protection of the condensate storage tank level instrumentation tubing. The existing plant configuration for this equipment did not conform to the licensing basis described in the USAR. The current plant design and initial evaluation of the team's findings used a probability approach, which differed from the licensing basis described in the USAR and the NRC's safety evaluation report (NUREG-0887). This is considered to constitute a potential unreviewed safety question.
H.B. Robinson Unit 2	During the inspection, the team identified a concern with the net positive suction head (NPSH) requirements within the safety injection (SI) and residual heat removal (RHR) pumps. A hydraulic analysis was performed subsequent to the inspection. You discovered a potential NPSH problem with SI pump C and reported it to the NRC in accordance with 10 CFR 50.72 on June 27, 1997. Subsequently you identified actual NPSH problems with SI pumps B and C for a large break loss-of- coolant accident (LBLOCA) as stated in LER 97-08. You have undertaken several immediate corrective actions, including raising the refueling water storage tank level. The hydraulic analysis is still ongoing for the RHR pumps.
	You discovered as the result of the team's inquiries that the redundant autostart cables for SI pumps A and C were routed in the same raceway in violation of your electrical separation criteria. You declared SI pump C inoperable, immediately placed the installed spare pump B in service, and implemented a modification to provide correct separation subsequent to this inspection. You advised the NRC of this discrepancy in accordance with 10 CFR 50.72 on May 21, 1997.
H.B. Robinson Unit 2 (Continued)	The team identified that you had not reported significant peak cladding temperature (PCT) changes as required by 10 CFR 50.46. Prior to the inspection you reported significant PCT changes for only the most limiting transient for all the evaluation models, whereas you should report them for the limiting transient of each evaluation model and its applications. The NRR technical staff is still evaluating a potential unreviewed safety question with regard to your commitments about the transfer to cold leg recirculation following a LBLOCA.
	The team found deficiencies with the improper slope of instrument sensing lines, the exclusion of the seismic uncertainty term in calculations for safe shutdown and accident mitigation instrumentation, and with verification of the closure capability of the accumulator isolation valves if a LOCA occurred while filling the accumulators.
	Weaknesses were also identified concerning updating and control of calculations, nonconservative design inputs and assumptions, and incorporating design bases into maintenance and test procedures. In addition, the team noted deficiencies and inconsistencies in the updated final safety analysis report, procedures, design basis documents, systems descriptions, calculations, drawings, and other documents.
St. Lucie Units 1 and 2	While none of the team's findings resulted in system inoperability, some errors made during the original plant design have reduced system operating margins. Of specific concern are the calculations which support operation of the component cooling water system. The current calculations for determining the temperature limit for the seawater intake to the component cooling water heat exchangers are nonconservative. Your interim actions to establish an 82 °F temperature limit on intake cooling water are adequate for the short term, but plant operation could be challenged by higher intake cooling water temperature that occur during the warmer months of the year.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
Three Mile Island Unit 1	Although many calculations reviewed by the team were satisfactory, the team identified design control weaknesses in the performance and control of calculations. In particular, the team noted the use of several nonconservative inputs and assumptions in the analysis for switchover of decay heat removal system (DHRS) pump suction from the borated water storage tank (BWST) to the reactor building sump under post-accident conditions. The plant was operated outside the design basis with potential for air entrainment in the emergency core cooling system pumps that could have rendered them inoperable. You evaluated this issue and concluded that the system was inoperable, notified the NRC in accordance with 10 CFR 50.72 on December 21, 1996, and revised operating procedures to resolve the problem. You also issued a licensee event report (LER 96-002 on January 20, 1997.
	The team identified that calculations were being performed in documents, such as memoranda, technical data reports, and plant engineering evaluation requests, that do not comply with your engineering procedures for calculations. For example, on the basis of a calculation in a memorandum, an incorrect decision was made not to test the check valves in the DHRS pump suction from the BWST to assure that the check valves are capable of preventing backflow from the reactor building sump.
Three Mile Island Unit 1 (Continued)	The team determined that the consequences of a letdown line break in the auxiliary building apparently had not been adequately evaluated. The team referred this issue to the technical review branch in the Office of Nuclear Reactor Regulation (NRR) staff for review regarding the extent to which TMI-1 was required to consider the effects of a letdown line break in the auxiliary building. The staff review concluded that the TMI-1 licensing basis for pipe breaks includes the postulation of full diameter breaks in the letdown line between the containment penetration and the breakdown orifice as described in Appendix 14A to the final safety analysis report (FSAR). Therefore, the design of safety-related equipment in the affected areas should consider the conditions resulting from these breaks.
	The team's other findings included: nonconservative assumptions and missing inputs in calculations for the makeup pumps and makeup tank; a potential unreviewed safety question in your evaluation of an FSAR change regarding the net position suction head for DHRS pumps; not periodically testing certain molded case circuit breakers; incorrect assignment of power supply to the makeup isolation valve; not initiating corrective actions in a timely manner for open items from your self-assessments of the two inspected systems; and discrepancies in the FSAR.
Vermont Yankee	First, the team identified several operability issues which required prompt corrective actions by your staff. For example, the team found that the nonsafety-related pressure regulator could result in loss of service water to the diesel generators. Also, the team questioned the operability of your residual heat removal (RHR) pumps with minimum pump flow considerably less than what the pump vendor recommended for continued operation. Additionally, the team raised concerns regarding the operability of the RHR pumps while in the torus cooling mode and the operability of the RHR heat exchangers on the basis of improperly performed tests. Your initial corrective actions to address these issues were acceptable.
	Secondly, the team had concerns with your past resolution to several engineering issues such as operation of the unit with the torus temperature above the analyzed region and various discrepancies in the plant's technical specifications. In particular, we were concerned with your long-term resolution to operation of your RHR pump with less than recommended minimum pump flow during a design basis event.

Significant 1997 Safety System Engineering Inspection Findings (cont.)

Plant	Design Findings Contained in Report Forwarding Letter
	Based on the understanding of your current design bases efforts, the team concluded that it was unlikely that you would have uncovered some of the issues identified in this report. Based on the conversation at the exit meeting, we understand that your staff will be re-examining your design bases program.
Washington Nuclear Project 2	One of the team's significant findings addresses a design deficiency that was introduced during the modification of the automatic depressurization system (ADS), inadvertently defeating the intended manual initiation of the system. Other significant issues involved the residual heat removal heat exchanger operability assessment based on data from faulty instruments, and the potential for exceeding the ADS activator design pressure by initiating containment spray during past loss-of-coolant accident elevated containment temperatures.
Washington Nuclear Project 2 (Continued)	Many of the team's findings relate to your failure to keep the final safety analysis report updated. Examples are, the ADS wiring modification and the removal of the service water system keepfull pumps from service. Your design review documents were of uneven quality. For example, the residual heat removal document lacked important instrumentation and control information.

Appendix E

Generic Plant System Groups

For the purposes of analysis, the design-basis issues (DBIs) and their affected systems was recorded using one or more of the 26 generic reactor plant system groupings listed below. Each generic system group represents one or more similar or related reactor plant systems. Since there is no standardization of system names amongst plants, a system group was chosen that best fit the actual plant system and its function. If the licensee did not specify the affected system, or the system specified did not fit into system groupings 1-25, the DBI-system was labeled as system 26 (Other).

Table 1 System groupings

System Number	System Group Title and Common System Titles
1	Accident monitoring instrumentation (Plant protection system, engineered safety features actuation system, post-accident monitoring system)
2	Auxiliary/emergency feedwater systems (Auxiliary/emergency feedwater system)
3	Combustible gas control systems (Containment combustible gas control system, Emergency/standby gas treatment system)
4	Component cooling water system (Closed/component cooling water system)
5	Containment and containment isolation (Containment isolation control system, containment leakage control system, containment vacuum relief system, reactor containment building, primary containment/undetermined system, reactor building (BWR)
6	Containment cooling systems (reactor building environmental control system, shield annulus return and exhaust system, containment ice condenser/refrigeration system, containment spray system, containment fan cooling system, containment fan cooling system)
7	Control room emergency ventilation system (control building/control complex environment control system)
8	Emergency AC/DC power systems (Diesel cooling water system, diesel generator starting air system, medium-Voltage power system - Class 1E, low-voltage power system - Class 1E, instrument and uninterruptable power system - Class 1E, DC power system - Class 1E, emergency onsite power supply system, emergency onsite power supply building environmental control system
9	Emergency core cooling systems (high pressure coolant injection system, high pressure core spray system, low pressure coolant injection system, low pressure core spray system, low pressure safety injection system, high pressure safety injection system, upper head injection, intermediate head injection)
10	Engineered safety features instrumentation (engineered safety features actuation system, radiation monitoring system, integrated control system, feedwater/steam generator water level control system, reactor power control system, solid state control system/auxiliary logic control system, containment environmental monitoring system, anticipated transient without scram system)
11	Essential compressed air system (essential air system)
12	Essential service water system (essential service water system)
13	Fire detection/suppression systems (fire detection system, fire protection system (water), fire protection system (chemical))
14	Isolation condenser system (isolation condenser system)
15	Low temperature/overpressure protection (low temperature/overpressure system)
16	Main steam isolation valves (main steam isolation valves)

System Number	System Group Title and Common System Titles
17	Primary reactor systems (control rod drive system, reactor coolant system, reactor recirculation system, reactor vessel system, pressurizer system, steam generating system)
18	Radiation monitoring instrumentation (radiation monitoring system, incore/excore neutron monitoring system, leak monitoring system, containment environmental monitoring system)
19	Reactor core isolation cooling systems (reactor core isolation cooling system)
20	Reactor trip instrumentation (plant protection system, feedwater/steam generator water level control system, reactor power control system, incore/excore neutron monitoring system)
21	Residual heat removal systems (residual heat removal system)
22	Safety and relief valves (reactor coolant system, main/reheat steam system, automatic depressurization system)
23	Spent fuel systems (fuel pool cooling and purification system, fuel building environmental control system)
24	Standby liquid control system (standby liquid control system)
25	Ultimate heat sink system (ultimate heat sink system)
26	Other