Introduction Contents

Introduction Tables

Material. As provided by 10 CFR 52.47(a)(1)(ix), these conceptual designs are not a part of the design certification for the System 80+ Standard Plant Design, and do not impose requirements applicable to a COL, nor to an application for a COL, that references the design certification rule. Textual material comprising Conceptual Design information is denoted by brackets surrounding such material; a listing of this information is provided in Table 1-1.

3.5 Plant-Specific Changes to Designated Material in the Approved Design Material

Certain information [Tier 2*] within sections of the Approved Design Material, summarized in Table 1-2, is designated with italicized text in the ADM. Plant-specific changes to any of this italicized design information shall require prior NRC Staff approval. The requirement for prior NRC Staff approval will expire for some of the designated information, as indicated in Table 1-2, when the COL holder first achieves 100% power operation.

3.6 Treatment of Probabilistic Risk Assessment Information

A design-specific Probabilistic Risk Assessment [PRA] for the System 80+ Standard Plant Design was submitted as part of the application for design certification, as required by 10 CFR 52.47. One purpose of the PRA was to develop insights for the design and its features. Significant insights that resulted from the PRA are identified in ADM Section 19.15. However, the detailed methodology and quantitative portions of the design-specific PRA were not included in the DCD because it is anticipated that this material will be subject to modifications and refinements as the detailed design is completed and the as-built plant parameters and new methodology become available.

3.7 Treatment of Severe Accident Evaluations

Chapter 19.11 of the ADM contains various deterministic evaluations of severe accidents for the System 80+ Standard Plant Design. With respect to these evaluations only; a proposed change in the facility or procedures described in the ADM, or a proposed on-site test or on-site experiment, shall be deemed to involve an unreviewed safety question if, as a result of the proposed change, test or experiment:

- The probability of a severe accident previously evaluated in Chapter 19.11 and deemed to be not credible, increases to the extent that the severe accident is ~~deemed~~ could become credible; or determined
- The postulated consequences to the public of a severe accident previously evaluated in Chapter 19.11 substantially increase.

typical: all chgs

Table 1-2 Index of ADM Items Requiring NRC Approval for Change

Item	~~Duration~~ Expiration	Reference
ASME Boiler & Pressure Vessel Code, Section III	First Full Power	Table 1-3
AISC-N690 and ACI-349 Industrial Codes	First Full Power	Tables 1-4, 1-5
Design, Qualification and Preoperational Testing for Motor-Operated Valves	First Full Power	Table 1-6
Equipment Seismic Qualification Methods	First Full Power	Table 1-7
Piping Design Acceptance Criteria	None	Table 1-8
First Cycle Fuel and Control Rod Design	First Full Power	Table 1-9
Instrumentation & Controls Setpoint Methodology	First Full Power	Table 1-10
Instrumentation & Controls Hardware and Software Changes	First Full Power	Table 1-11
Instrumentation & Controls Environmental Qualification	First Full Power	Table 1-12
Control Room Human Factors Engineering	None	Table 1-13

Note: The applicable portion of the designated Tier 2 reference material specified in Tables 1-3 through 1-13 is shown italicized within the identified Approved Design Material [ADM] text or table.

Table 1-3 ASME Boiler & Pressure Vessel Code, Section III

Commitment	ADM Reference
ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division ~~I~~ 1, ~~Division II~~	Table 1.8-6
ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components"	3.8.2.2

Table 1-4 AISC-N690 Industrial Code

Commitment	ADM Reference
AISC-N690, Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities	Table 1.8-6
Analysis and Design of Seismic Category I Steel Structures	3.8.4.5.2

Table 1-8 Piping Design Acceptance Criteria

Tbl 1.8-6

Commitment	ADM Reference
ASME Code and code cases for System 80+ piping and pipe support design	1.8 3.9A (1.1)
Analysis methods; experimental stress analysis, independent support motion, inelastic analysis, small-bore piping, non-seismic/seismic interaction, buried piping	3.7.3.2, 3.7.3.8, 3.7.3.9, 3.7.3.12, 3.7.3.13, 3.9.1.3, 3.9A (1.1)
Piping modeling; piping benchmark program, decoupling criteria	3.6.2.1.4.1, 3.9.1.2.1, 3.9A (1.5.2.2)
Pipe stress analysis criteria; loading and load combinations, damping values, combination of modal responses, high frequency modes, thermal oscillations in piping connected to the reactor coolant system, thermal stratification, safety-related valve design, installation and testing, functional capability, combination of inertial and seismic motion effects, welded attachments, modal damping for composite structures, minimum temperature for thermal analyses	3.6.2.2.2, 3.6.3.8, 3.7.2.15, Tbl 3.7-1, 3.9.3.1, 3.9.3.1.4.3, 3.9.3.3, Tbls 3.9-10 & 11, 3.9A (1.4.2, 1.4.3.2.1, 1.4.7, 1.5.2.2, 1.6.5)
Pipe support criteria; applicable codes, jurisdictional boundaries, pipe support baseplate and anchor bolt design, use of energy absorbers and limit stops, pipe support stiffness, seismic self-weight excitation, design of supplementary steel, consideration of friction forces, pipe support gaps and clearances, instrumentation line support criteria	3.9.3.4, 3.9A (1.10.1, 1.10.2, 1.7.2.3, 1.7.2.8, 1.7.2.9, 1.7.2.10, 1.7.4, 1.7.5)

Table 1-9 First Cycle Fuel and Control Rod Design

Commitment	ADM Reference
Fuel and initial core design description and permissible changes	4.1.1
Design features and acceptance criteria for fuel and initial core design	Tables 4.1-1, 4.1-2

Table 1-10 Instrumentation & Controls Setpoint Methodology

Commitment	ADM Reference
Generation of safety system setpoints	7.1.2.27

Table 1-11 Instrumentation & Controls Hardware and Software Changes

Commitment	ADM Reference
Design, verification, implementation and validation of computer systems software changes in safety-related systems	7.1.2.32

Table 1-12 Instrumentation & Controls Environmental Qualification

Commitment	ADM Reference
Environmental qualification of electrical equipment	3.11.1

2

Table 1-13 Control Room Human Factors Engineering

Commitment	ADM Reference
Human Factors program plan	18.4.2
Human Factors Engineering verification and validation plan	18.4.9
Functional task analysis, workload & environmental assumptions and bases	18.5.1.1
Task decomposition and data framework	18.5.1.3, 18.5.1.3.2, 18.5.1.3.3
Workload loading criteria	18.5.1.4
Nuplex 80+ control room functional task analysis, scope, PRA and critical tasks, information and control requirements, time profile/workload evaluation, link analysis, identification of overload situations	18.5.1.5.1, 18.5.1.5.2, 18.5.1.5.3, 18.5.1.5.4, 18.5.1.5.5, 18.5.1.5.6
Main control room annunciator, display and control inventory	18.5.4
Control room staffing assumptions	18.6.2.2
Control room console panel profiles	18.6.5.7
Nuplex 80+ information presentation, standard features	18.7.1
Nuplex 80+ safety-related information	18.7.1.8.1
Remote shutdown panel safety-grade instrumentation and controls	18.8.1.1

ABBREVIATION LIST (Continued)

Abbreviation	Meaning
CGS	Compressed Gas Systems
CH	Channel
CHRS	Containment Hydrogen Recombiner System
CIAS	Containment Isolation Actuation Signal
CIS	Containment Isolation System
CIV	Containment Isolation Valve
COL	Combined Operating License
CONT	Containment
CPC	Core Protection Calculator
CPVS	Containment Purge Ventilation System
CRS	Control Room Supervisor
CSAS	Containment Spray Actuation Signal
CSB	Core Support Barrel
CSS	Containment Spray System
CST	Chemical Sample Tank
CT	Combustion Turbine/Generator
CVAP	Comprehensive Vibration Assessment Program
CVCS	Chemical and Volume Control System
CWT	Chemical Waste Tank
DBVS	Diesel Building Ventilation System
DEMIN	Demineralized
DFSS	Diesel Fuel Storage Structure
DIAS	Discrete Indication and Alarm System
DIAS-N	Discrete Indication and Alarm System - Channel N
DIAS-P	Discrete Indication and Alarm System - Channel P
DNBR	Departure From Nucleate Boiling Ratio
DPS	Data Processing System
D-RAP	Design Reliability Assurance Program
DVI	Direct Vessel Injection
DWMS	Demineralized Water Makeup System

ABBREVIATION LIST (Continued)

Abbreviation	Meaning
HSI	Human-System/Interface
HVAC	Heating, Ventilating, Air Conditioning
HVT	Holdup Volume Tank
HX	Heat Exchanger
HZ	Hertz
IAS	Instrument Air System
ICI	In-Core Instrument
ILRT	Integrated Leak Rate Test
INIT	Initiation
INJ	Injection
INST	Instrumentation
IPSO	Integrated Process Status Overview
IRWST	In-containment Refueling Water Storage Tank
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
ITP	Interface and Test Processor
IVMS	Internals Vibration Monitoring System
IWSS	In-containment Water Storage System
IX	Ion Exchanger
LA	Low Activity
LBB	Leak-Before-Break
LOCA	Loss-of-coolant Accident
LOOP	Loss-of-Offsite-Power
LPMS	Loose Parts Monitoring System
LPZ	Low Population Zone
LS	Liquid Sample
LTOP	Low Temperature Overpressure Protection
LWMS	Liquid Waste Management System
MCC	Motor Control Center
MCR	Main Control Room
MCRACS	Main Control Room Air Conditioning System

- ~~a.~~ Requirements that the design basis analytical limits, data, assumptions, and methods used as the bases for selection of trip setpoints are specified and documented.

- ~~b.~~ Instrumentation accuracies, drift and the effects of design basis transients are accounted for in the determination of setpoints.

- ~~c.~~ The method utilized for combining the various uncertainty values is specified.

- ~~d.~~ Identifies required pre-operational and surveillance testing.

- ~~e.~~ Identifies performance requirements for replacement of setpoint related instrumentation.

- ~~f.~~ The setpoint calculations are consistent with the physical configuration of the instrumentation.

Reactor Trip Initiation Function

Process instrumentation, the Plant Protection Calculators (PPCs), the Core Protection Calculators (CPCs), and the reactor trip switchgear function to initiate an automatic reactor trip. The process instrumentation provides sensor data input to the PPS which monitors the following plant conditions to provide a reactor trip:

Reactor Power - High
Reactor Coolant System Pressure - Low or High
Steam Generator Water Level - Low or High
Steam Generator Pressure - Low
Containment Pressure - High
Reactor Coolant Flow - Low
Departure from Nucleate Boiling Ratio - Low
Linear Heat Generation Rate - High

Setpoints for initiation of a reactor trip are installed for each monitored condition to provide for initiation of a reactor trip prior to exceeding reactor fuel thermal limits and the Reactor Coolant System pressure boundary limits for anticipated operational occurrences. If a monitored condition reaches its setpoint, the PPS automatically actuates the reactor trip switchgear.

Engineered Safety Features Initiation Function

Process instrumentation, the PPCs, the ESF-CCS, motor starters, and other actuated devices function to initiate the engineered safety feature systems. The process instrumentation provides sensor data input to the PPCs, which monitor the following plant conditions to initiate the engineered safety features systems.

Pressurizer Pressure - Low
Steam Generator Water Level - Low or High
Steam Generator Pressure - Low
Containment Pressure - High

If a monitored condition reaches its setpoint, the PPCs automatically generate one or more of the following Engineered Safety Feature Actuation Signals (ESFAS).

Typical - Revised Pg N°'s

Table 2.5.1-1 Plant Protection System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. The PPS initiates reactor trip and ESF system actuations within allocated response times.	13. Testing and analysis will be performed to measure PPS equipment response times.	13. Measured response times are less than or equal to the response time values required for reactor trip and ESF actuations.
14. Setpoints for initiation of PPS safety-related functions are determined using methodologies which have the following characteristics:	14. Inspection will be performed on the setpoint calculations.	14. The inspection of the setpoint calculation confirms the use of setpoint methodologies that require:
• a) Requirements that the design basis analytical limits, data, assumptions, and methods used as the bases for selection of trip setpoints are specified and documented.		• a) Documentation of data, assumptions, and methods used in the bases for selection of trip setpoints is performed.
• b) Instrumentation accuracies, drift, and the effects of design basis transients are accounted for in the determination of setpoints.		• b) Consideration of instrument calibration uncertainties and uncertainties due to environmental conditions, instrument drift, power supply variation, and the effect of design basis event transients is included in determining the margin between the trip setpoint and the safety limit.
• c) The method utilized for combining the various uncertainty values is specified.		• c) The methods used for combining uncertainties is consistent with those specified in the methodology plan.
• d) Identifies required preoperational and surveillance testing.		• d) The use of written procedures for required preoperational and surveillance testing.

Table 2.5.1-1 Plant Protection System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
• ~~e)~~ Identifies performance requirements for replacement of setpoint related instrumentation.		• ~~e)~~ Evaluation for equivalent or better performance of replacement instrumentation which is not identical to original equipment is documented.
• ~~f)~~ The setpoint calculations are consistent with the physical configuration of the instrumentation.		• ~~f)~~ The configuration of the as-built instrumentation is consistent with the attributes used in the setpoint calculations for location of taps and sensing lines.

Table 2.4.4-1 Safety Injection System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
2. (Continued) Redline.	2.d) Inspection of construction records for SIS piping will be conducted.	2.d) The volume in each direct vessel injection line, from the connection for the SIT return header to the piping-to-DVI nozzle weld, is no greater than 27.8 cubic feet.
3. The safety injection tanks can be depressurized by venting for entry into shutdown cooling.	3. Testing will be performed with the SITs pressurized and the associated SIT isolation valve shut. Each SIT vent valve will be opened from the MCR.	3. The SIT vent valves can be opened from the MCR and the SIT pressure decreases while the SIT is being vented.
4. A flow recirculation line from each SIS pump discharge to the IRWST provides a minimum flow recirculation path.	4. Testing of SIS will be performed by manually aligning SI flow to the IRWST through the minimum flow recirculation line and manually starting each SIS pump.	4. Minimum flow recirculation rate meets or exceeds the pump vendor's minimum flow requirements.
5. The SIS pumps can be tested at full flow during plant operation.	5. Testing of the SIS will be performed by manually aligning SIS flow to the IRWST and manually starting each SIS pump.	5. Each SIS pump has a flow capacity of at least 980 gpm to the IRWST through the test line.
6. The ASME Code Section III SIS components shown on Figure 2.4.4-1 retain their pressure boundary integrity under internal pressures that will be experienced under service.	6. A pressure test will be conducted on those components of the SIS required to be pressure tested by ASME Code Section III.	6. The results of the pressure test of ASME Code Section III components of the SIS conform with the pressure testing acceptance criteria in ASME Code Section III.
7.a) Displays of the SIS instrumentation shown on Figure 2.4.4-1 exist in the MCR or can be retrieved there.	7.a) Inspection for the existence or retrievability in the MCR of instrumentation displays will be performed.	7.a) Displays of the instrumentation shown on Figure 2.4.4-1 exist in the MCR or can be retrieved there.

Table 2.5.1-1 Plant Protection System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. PPS software is designed, tested, installed and maintained using a process which: Red line 15.a) Defines the organization, responsibilities, and software quality assurance activities for the software engineering life cycle that provides for: • establishment of plans and methodologies • specification of functional, system and software requirements and standards, identification of safety critical requirements • design and development of software • software module, unit, and system testing practices • installation and checkout practices • reporting and correction of software defects during operation	15. Inspection will be performed of the process used to design, test, install, and maintain the PPS safety related software.	15.a) The process defines the organization, responsibilities and activities for the following phases of the software engineering life cycle: • Establishment of plans and methodologies for all software to be developed. • Specification of functional, system and software requirements, and identification of safety critical requirements. • Design of the software architecture, program structure, and definition of the software modules. • Development of the software code and testing of the software modules. • Interpretation of software and hardware and performance of unit and system tests. • Software installation and checkout testing. • Reporting and correction of software defects during operation.

Table 2.5.1-1 Plant Protection System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
15. (Continued)	15. (Continued)	15. (Continued)
Redline 15b) Specifies requirements for: • software management, documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action • software configuration management, historical records of software, and control of software changes • verification & validation, and requirements for reviewer independence		Redline 15.b) The process has requirements for the following software development functions: • Software management, which defines organization responsibilities, documentation requirements, standards for software coding and testing, review requirements, and procedures for problem reporting and corrective actions. • Software configuration management, which establishes methods for maintaining historical records of software as it is developed, controlling software changes and for recording and reporting software changes. • Verification and validation, which specifies the requirements for the verification review process, the validation testing process, review and test activity documentation, and reviewer independence.
Redline 15c) Incorporates a graded approach according to the software's relative importance to safety.		Redline 15.c) The process establishes the method for classifying PPS software elements according to their relative importance to safety. The process defines the tasks to be performed for software assigned to each safety classification.

Table 2.5.2-1 Engineered Safety Features Component Control System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. Setpoints for interlocks and actuation of ESF-CCS safety-related functions are determined using methodologies which have the following characteristics: • ~~a)~~ Requirements that the design basis analytical limits, data, assumptions, and methods used as the bases for selection of trip setpoints are specified and documented. • ~~b)~~ Instrumentation accuracies, drift and the effects of design basis transients are accounted for in the determination of setpoints. • ~~c)~~ The method utilized for combining the various uncertainty values is specified. • ~~d)~~ Identifies of required preoperational and surveillance testing. • ~~e)~~ Identifies performance requirements for replacement of setpoint related instrumentation. • ~~f)~~ The setpoint calculations are consistent with the physical configuration of the instrumentation.	19.a) Inspection will be performed on the setpoint calculations.	19.a) The inspection of the setpoint calculation confirms the use of setpoint methodologies that require: • ~~a)~~ Documentation of data, assumptions, and methods used in the bases for selection of trip setpoints is performed. • ~~b)~~ Consideration of instrument calibration uncertainties and uncertainties due to environmental conditions, instrument drift, power supply variation, and the effect of design basis event transients is included in determining the margin between the trip setpoint and the safety limit. • ~~c)~~ The methods used for combining uncertainties is consistent with those specified in the methodology plan. • ~~d)~~ The use of written procedures for required preoperational and surveillance testing. • ~~e)~~ Evaluation for equivalent or better performance of replacement instrumentation which is not identical to original equipment is documented. • ~~f)~~ The configuration of the as-built instrumentation is consistent with the attributes used in the setpoint calculations for location of taps and sensing lines.

Table 2.5.2-1 Engineered Safety Features Component Control System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19. (Continued)	19.b) Testing will be performed to verify interlock and actuation responses to simulated input signals.	19.b) 1) The correct ESF-CCS response occurs when an input signal crosses the setpoint threshold. 2) Changing of a setpoint does not also change the setpoints of other trips or interlocks.
20. ESF-CCS software is designed, tested, installed, and maintained using a process which: Redline 20.a) Defines the organization, responsibilities, and software quality assurance activities for the software engineering life cycle that provides for: • establishment of plans and methodologies • specification of functional, system and software requirements and standards, identification of safety critical requirements • design and development of software • software module, unit and system testing practices	20. Inspection will be performed of the process used to design, test, install, and maintain the ESF-CCS software.	20.a) The process defines the organization, responsibilities and activities for the following phases of the software engineering life cycle: • Establishment of plans and methodologies for all software to be developed; • Specification of functional, system, and software requirements and identification of safety critical requirements; • Design of the software architecture, program structure, and definition of the software modules; • Development of the software code and testing of the software modules; • Interpretation of software and hardware and performance of unit and system tests; • Software installation and checkout testing; and • Reporting and correction of software defects during operation.

Table 2.5.2-1 Engineered Safety Features Component Control System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
20. (Continued) • installation and checkout practices • reporting and correction of software defects during operation Redline 20.b) Specifies requirements for: • software management, documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action • software configuration management, historical records of software, and control of software changes • verification & validation, and requirements for reviewer independence Redline 20.c) Incorporates a graded approach according to the software's relative importance to safety.	20. (Continued)	Red 20.b) The process has requirements for the following software development functions: • Software management, which defines organization responsibilities, documentation requirements, standards for software coding and testing, review requirements, and procedures for problem reporting and corrective actions; • Software configuration management, which establishes methods for maintaining historical records of software as it is developed, controlling software changes and for recording and reporting software changes; and • Verification and validation, which specifies the requirements for the verification review process, review and test activity documentation, and reviewer independence. Redline 20.c) The process establishes the method for classifying ESF-CCS software elements according to their relative importance to safety. The process defines the tasks to be performed for software assigned to each safety classification.

Table 2.5.2-1 Engineered Safety Features Component Control System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
20. (Continued)	20. (Continued)	20. b) The process has requirements for the following software development functions:
20. b) Specifies requirements for:		
• software management, documentation requirements, standards, review requirements, and procedures for problem reporting and corrective action		• Software management, which defines organization responsibilities, documentation requirements, standards for software coding and testing, review requirements, and procedures for problem reporting and corrective actions;
• software configuration management, historical records of software, and control of software changes		• Software configuration management, which establishes methods for maintaining historical records of software as it is developed, controlling software changes and for recording and reporting software changes; and
• verification & validation, and requirements for reviewer independence		• Verification and validation, which specifies the requirements for the verification review process, review and test activity documentation, and reviewer independence.
20. c) Incorporates a graded approach according to the software's relative importance to safety.		20. c) The process establishes the method for classifying ESF-CCS software elements according to their relative importance to safety. The process defines the tasks to be performed for software assigned to each safety classification.

Table 2.6.3-1 AC Instrumentation and Control Power System and DC Power System

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Basic Configuration of the AC Instrumentation and Control Power System and the DC Power System is as described in the Design Description (Section 2.6.3).	1. Inspection of the as-built AC Instrumentation and Control Power System and the DC Power System configuration will be conducted.	1. The as-built AC Instrumentation and Control Power System and the as-built DC Power System conforms with the Basic Configuration as described in the Design Description (Section 2.6.3).
2. Each Class 1E constant voltage, constant frequency inverter power supply unit in normal operating mode receives Class 1E direct current (DC) power from its respective DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center normal power source to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads. This alternate power source is a voltage regulating device which is supplied power from the same AC power source as the battery charger associated with the Class 1E DC distribution center servicing the inverter power supply unit.	2. Inspection of the as-built Class 1E constant voltage, constant frequency inverter power supply unit will be conducted. {automatically and manually}	2. Each Class 1E constant voltage, constant frequency inverter power supply unit in normal operating mode receives Class 1E direct current (DC) power from its respective DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center normal power source to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads. This alternate power source is a voltage regulating device which is supplied power from the same AC power source as the battery charger associated with the Class 1E DC distribution center servicing the inverter power supply unit.

Table 2.7.6-1 Equipment Receiving Component Cooling Water Flow

Plant Mode/ Components	Normal Operation	Shutdown Cooling	Refueling	Design Basis Accident
SAFETY RELATED		(Note ~~a~~ 1)		
Shutdown cooling heat exchanger	-	X	X	-
Containment spray heat exchanger	-	-	-	X
Spent fuel pool cooling heat exchanger	X	X	X	X (Note ~~b~~ 2)
Diesel Generator	X	X	X	X
Pump Motor Coolers, Miniflow Heat Exchangers, and Essential Chilled Water Condensers	X	X	X	X
NON-SAFETY RELATED		(Note ~~a~~ 1)		
Reactor coolant pumps ~~pumps~~ and pump motors	X	X	X	X
Charging pump motor coolers	X	X	X	X
Charging pump miniflow heat exchanger	X	X	X	X
Instrument Air Compressors	X	X	X	X
Normal Chilled Water Condensers (Note ~~c~~ 3)	X	X	X	-
Letdown Heat Exchanger, Sample Heat Exchangers, Gas Stripper, and Boric Acid Concentrator (Note ~~c~~ 3)	X	X	X	-

NOTES FOR ~~TABLES~~ TABLE 2.7.6-1

1 ~~a.~~ (X) = Equipment can receive component cooling water flow in this mode.
(-) = Equipment does not receive component cooling water flow in this mode.

2. ~~b.~~ Will require operator action to restore.

3 ~~c.~~ Assignment of the non-safety-related CCWS heat removal loads to the respective CCWS Division is dependent upon the location of the components associated with those loads.

Table 2.9.4-1 Process and Effluent Radiological Monitoring and Sampling System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Each safety-related area radiation monitor channel monitors the radiation level in its assigned area, and indicates its respective MCR alarm and local audible and visual alarm (if provided) when the radiation level reaches a preset level.	5. Testing of each channel of the safety-related area radiation monitors will be conducted using simulated input signals.	5. MCR and local alarms are initiated when the simulated radiation level reaches a preset limit.
6. The following PERMSS safety-related instrumentation shall be provided: a) MCR intake radiation monitor (2/intake), b) high range containment area radiation monitor (2), c) containment atmosphere radiation monitor (particulate channel only), d) primary coolant loop radiation monitors (2).	6. Inspection of the as-built system will be conducted.	6. The as-built PERMSS conforms with the design description.

these should be bullets

Table 2.9.4-1 Process and Effluent Radiological Monitoring and Sampling System (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. Each safety-related area radiation monitor channel monitors the radiation level in its assigned area, and indicates its respective MCR alarm and local audible and visual alarm (if provided) when the radiation level reaches a preset level.	5. Testing of each channel of the safety-related area radiation monitors will be conducted using simulated input signals.	5. MCR and local alarms are initiated when the simulated radiation level reaches a preset limit.
6. The following PERMSS safety-related instrumentation shall be provided: • MCR intake radiation monitor (2/intake), • high range containment area radiation monitor (2), • containment atmosphere radiation monitor (particulate channel only), • primary coolant loop radiation monitors (2).	6. Inspection of the as-built system will be conducted.	6. The as-built PERMSS conforms with the design description.
7. The PERMSS safety-related instrumentation (the MCR intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.	7. Seismic analyses of the as-built PERMSS safety-related instrumentation will be performed.	7. An analysis report exists which concludes that the PERMSS safety-related instrumentation (the MCR intake radiation monitors, high range containment area radiation monitors, containment atmosphere radiation monitor (particulate channel), and the primary coolant loop radiation monitors) are classified Seismic Category I.

Table 3.2-1 Radiation Protection

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The Radiation Protection design includes shielding or provisions for temporary shielding of rooms, corridors, cubicles, labyrinth access, and operating areas commensurate with their expected occupancy and use.	1. An analysis of the expected radiation levels in each plant area will be performed to verify the adequacy of the shielding design. This analysis shall consider the following: a) Confirmatory calculations shall consider significant radiation sources (greater than 5% contribution) for an area. Radiation source strength in plant systems and components will be determined based on an assumed source term of 0.25% fuel cladding defects and a core inventory commensurate with a 3914 MWt equilibrium core. Source terms shall be adjusted for radiological decay and buildup of activated corrosion and wear products. b) Commonly accepted shielding codes, using nuclear properties derived from well known references shall be used to model and evaluate plant radiation environments. i ~~1)~~ For non-complex geometries, point kernel shielding codes may be used. ii ~~2)~~ For complex geometries, more sophisticated two or three dimensional transport codes shall be used.	1. Maximum radiation levels are less than or equal to the radiation levels in the radiation zones specified in Table 3.2-2. Plant layout is such that access to higher zones (areas with higher dose rates) is from lower zoned areas. Corridors and normal traffic areas are Zone 3 or less. Control Rooms are Zone 2 or less. Radiation zone designations for components during normal operating conditions are listed in Table 3.2-3.
2. The plant design shall have provisions to reduce radiation exposure from adjacent sources of radiation.	2. Using the methods identified in 1. above, radiation levels in zones shall be evaluated for contribution from adjacent sources of radiation.	2. Shielding design of a zone is such that radiation from adjacent sources of radiation shall contribute no more than a small fraction of the dose rate in the zone.

Table 3.2-1 Radiation Protection (Continued)

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
3. The Radiation Protection design includes plant shielding to permit operators to perform actions that may require operator access during and following a design basis accident.	3. An analysis will be performed using design basis accident source terms and calculational methods consistent with 1.b. above to determine the expected radiation levels in areas of the plant that may require operator access during and following a design basis accident.	3.a) The predicted individual personnel occupational does are less than or equal to 5 rem to the whole body total, or its equivalent, over the entire time period(s) during which operator access is required. 3.b) For areas requiring continuous occupancy, the predicted individual personnel occupational dose rates do not exceed 15 mrem/hr, averaged over 30 days.
4. The Radiation Protection design includes provisions for ventilation to limit airborne radioactivity to levels that permit personnel access.	4. An analysis will be performed to predict airborne radioactivity concentrations in rooms, corridors, cubicles, and operating areas during normal operations. Total ventilation flow rates and equipment leakages will be considered in the analysis, which will be based on source terms consistent with item 1.a).	4. The analysis concludes that airflows are from areas of lower potential airborne contamination to those of higher potential airborne contamination, prior to removal by the filters or vent system, and,

Chapter 1 Tables

1.0 Introduction and General Description of Plant

1.1 Introduction

This section of the Design Control Document describes the System 80+™ (†) Standard Plant Design. Information is provided, to the extent applicable to the System 80+ Standard Design, for the final safety analysis report required under 10 CFR 50.34(b). Other information relevant to the System 80+ Standard Design, such as Three Mile Island requirements, technical resolution of Unresolved Safety Issues and medium and high priority Generic Safety Issues, a complete set of interface requirements and site parameters, and important design features identified in risk assessments, is provided as required for design certification under 10 CFR 52.47(a). This descriptive information for the System 80+ Standard Plant Design has been evaluated and accepted by the Nuclear Regulatory Commission (NRC) as documented in NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design." For purposes of the Design Control Document, this material on the description and analysis of the System 80+ Standard Design is termed "Approved Design Material."

The System 80+ Standard Design is an evolutionary development of the proven System 80 design constructed and operated at the Palo Verde Nuclear Generating Station, and currently (1994) under construction at the Yonggwang and Ulchin sites in South Korea. System 80+ incorporates a variety of engineering and operational improvements[1] designed to provide additional reliability and safety margins when compared to the System 80 design. Further, design features to address the NRC's Severe Accident and Safety Goal Policy Statements are incorporated into the System 80+ Standard Design.

A summary of the System 80+ Standard Design is presented in Section 1.2. Detailed information on Systems, Structures, and Components that comprise the System 80+ standard design is provided in the following chapters and sections of this Approved Design Material.

1.1.1 System 80+ Standard Design

The scope of the System 80+ Standard Design covers an essentially complete nuclear power plant and includes all structures, systems, and components that can significantly affect safe operation. All major structures within the scope of the certified design are identified with a "cross-hatch" marking on the site arrangement layout (Figure 1.2-1). Site-specific structures are shown on that arrangement layout with "slash" markings. Structures, systems and components not in or partially within the scope of the System 80+ design are listed in Section 1.9.

1.1.2 Power Levels

The System 80+ Standard Design, described herein, includes a reactor core designed to operate at a maximum core power level of 3914 MWt. While the System 80+ design is independent of power level, this core power level was selected for the analysis described herein to provide limiting design and safety analysis parameters. At this core power level, the total thermal output is 3931 MWt.

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† System 80+ is a trademark of Combustion Engineering, Inc.

1 Specifically, for the System 80+ Standard Design, the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document has been used as a guide for utility requirements regarding plant design.

1.2 General Plant Description

1.2.1 Principal Site Characteristics

1.2.1.1 Site Location

The System 80+ Standard Design is designed for use at multiple sites as described in Chapter 2. The site-specific SAR will identify the specific site for that unit.

1.2.1.2 Plant Surroundings

The System 80+ Standard Design is designed for use at multiple sites. The site-specific SAR will identify the specific surroundings for that unit.

1.2.1.2.1 Meteorology

Section 2.3 lists, for plant radiological evaluation purposes, the short-term (accident) and long-term (routine) diffusion estimates (χ/Q). Other meteorological design bases are listed in Table 2.0-1. Section 2.3 of the site-specific SAR will include data to show compliance with the design bases.

1.2.1.2.2 Hydrology

Hydrological design bases are listed in Table 2.0-1. Section 2.4 of the site-specific SAR will include data to show compliance with the design bases.

1.2.1.2.3 Geology and Seismology

The design of safety-related structures, systems, and components of the System 80+ Standard Design is consistent with the seismic envelope given in Section 2.5. Section 2.5 of the site-specific SAR will include data to show compliance with the seismic envelope.

1.2.1.3 Plant Independence

The System 80+ Standard Design can be used at either single-plant or multiple-plant sites. At multiple-plant sites, the independence of all safety-related systems and their support systems will be maintained between (or among) the individual plants.

1.2.1.4 Site Building Arrangement

A typical layout of the System 80+ Standard Design buildings is shown in Figure 1.2-1. Sufficient open space is shown so that a facility for dry storage of spent fuel casks can be added on a site-specific basis.

1.2.1.4.1 Site-Specific Structures Description and Interface Requirements

Some structures which house non-safety related and certain safety-related systems and components are supplied by the licensee and are not included in the System 80+ design certification. To ensure that the design of such structures is compatible with the System 80+ Standard Design, certain interface requirements must be met by the applicant (owner/operator). The following sections present the interface requirements and conceptual descriptions for the Administration Building, Personnel Access Portal and Warehouse. In addition to lists of interface requirements, the word "shall" is used to identify interface

requirements in the descriptive text. The remainder of the description is conceptual and it is not intended to be binding on the COL holder.

Interface requirements for structures which are related to a specific mechanical or electrical system are covered in the appropriate chapter, e.g. the Station Service Water Pump Structure is covered in Section 9.2.1, Station Service Water System. Section 1.9 contains an index of all System 80+ interface requirements.

1.2.1.4.1.1 Administration Building

An Administration Building shall be provided by the licensee. [[This building provides office and support space for station administration and management personnel who have no need to be located within the Protected Area.

A typical Administration Building is designed as non-safety related, non-seismic structure with the following conceptual features. The building is a steel-framed structure with a steel deck roof covered by non-combustible roofing. Walls are insulated metal siding or masonry. Roof drainage and clean floor drainage are discharged to the storm and waste water system. The building is located immediately outside the Protected Area fence at the entrance to the plant, near the Personnel Access Portal building. Air conditioning and heating is provided to meet normal office environment conditions.]][1]

1.2.1.4.1.2 Personnel Access Portal

The Personnel Access Portal (PAP) shall be provided by the licensee, and shall be designed to provide the following functions:

- Serve as access point through the Protected Area Boundary.
- Provide facilities to search, badge, and permit access to the Protected Area.
- Provide the Secondary Alarm Stations.
- Provide the required bullet-resistant features to support security force functions.

[[A typical PAP building is a masonry building with non-combustible roofing on a metal deck. The PAP building is located along the Protected Area fence at the entrance to the plant, near the Administration Building. A PAP building ventilation system is provided to maintain the building within design temperature limits.]][1]

1.2.1.4.1.3 Warehouse

The licensee shall provide a warehouse to accommodate the following:

- Material access to the Protected Area incorporating the Vehicle (and Cargo) Access Portal (VAP).
- Loading docks, search areas, QA inspection and QA Hold Areas, along with systems and fixtures to provide bulk storage of QA and non QA parts and supplies.

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1 Conceptual Design information; see DCD Introduction Section 3.4

reactor vessel, and flows through the tube side of the two vertical U-tube steam generators (with an integral economizer) where heat is transferred to the secondary system. Reactor coolant pumps return the reactor coolant to the reactor vessel.

Two steam generators using heat generated by the reactor core and carried by the primary coolant to each steam generator, produce steam for driving the plant turbine-generator. Each steam generator is a vertical U-tube heat exchanger with an integral economizer which operates with the reactor coolant on the tube side and secondary coolant on the shell side. Each unit is designed to transfer heat from the Reactor Coolant System to the secondary system to produce saturated steam when provided with the proper input feedwater. Moisture separators and steam dryers on the shell side of the steam generator limit the moisture content of the steam during normal operation at full power. An integral flow restrictor has been designed into each steam generator steam nozzle to restrict flow in the event of a steam line break.

The System 80+ steam generator incorporates several design enhancements including better steam dryers, increased overall heat transfer area and slightly reduced full power steam pressure. Lyle ✓ Reviewing 1-16

The System 80+ steam generator also has a larger secondary feedwater inventory which extends the "boil dry" time, thus enhancing the NSSS's capability to tolerate upset conditions and improving operational flexibility. Finally, the System 80+ steam generator design has a greater tube plugging allowance, thus; permitting the NSSS to maintain rated output with a significant number of tubes plugged.

The RCS is further discussed in Chapter 5.

1.2.4 Engineered Safety Features

Engineered safety features function in the highly unlikely event of an accidental release of radioactive fission products from the reactor coolant system, particularly as the result of loss-of-coolant-accidents. These safeguards function to localize, control, mitigate, or terminate such accidents to hold exposure levels below the limits of 10 CFR 100.

1.2.4.1 Containment Structure

General arrangements for the reactor building, including the containment vessel, are shown in Figures 1.2-2 through 1.2-12. The containment vessel is a 200-foot diameter spherical shell with a wall thickness of approximately one and three-quarter inches. This containment shell is supported by a spherical support pedestal which is part of the reactor building. The reactor building is a reinforced concrete cylindrical building with a hemispherical dome which totally encloses the containment, internal structure and subsphere. The exterior walls of the reactor building, including the dome, are referred to as the shield building. Space below the containment and inside the shield building is referred to as the subsphere and is occupied by Engineered Safety Features equipment, e.g., emergency core cooling system equipment, containment spray system equipment, shutdown cooling system equipment, and emergency feedwater equipment.

A more detailed physical description of the containment and the design criteria relating to the construction techniques, static loads, and seismic loads are provided or referenced in Section 3.8.

The containment design basis is to provide an essentially leak-tight barrier against the release of radioactive materials subsequent to postulated accidents. In order to meet this requirement, a maximum

containment leakage rate is defined in conjunction with performance requirements placed on the Engineered Safety Features (ESF) systems.

The capability of the containment structure to maintain design leaktight integrity and to provide a predictable environment for operation of ESF systems is ensured by a comprehensive design, analysis, and testing program that includes consideration of:

- The peak containment pressure and temperature associated with the most severe postulated accident coincident with the Safe Shutdown Earthquake.

- The maximum external pressure loading condition to which the containment may be subjected as a result of inadvertent containment systems operations that potentially reduce containment internal pressure below outside atmospheric pressure, coincident with the Safe Shutdown Earthquake.

1.2.4.2 Safety Injection System

The Safety Injection System (SIS) is designed to satisfy NRC regulatory requirements. These requirements are specified as the Licensing Design Basis for the System 80+ design. In addition, the EPRI ALWR Requirements Document has been used to define a Safety Margin Design Basis for the SIS design. The Safety Margin Design Basis contains requirements which go beyond the minimum required by the Code of Federal Regulations, thereby providing additional safety assurance in the SIS design.

In the highly unlikely event of a loss-of-coolant-accident, the SIS injects borated water into the Reactor Coolant System. The System 80+ SIS incorporates a four-train safety injection configuration and an In-Containment Refueling Water Storage Tank (IRWST).

The System 80+ SIS utilizes four safety injection pumps to inject borated water directly into the Reactor Vessel. In addition, four safety injection tanks are provided. The SI pumps are normally aligned to the IRWST and a realignment for recirculation following a LOCA is not required. This system provides cooling to limit core damage and fission product release and ensures adequate shutdown margin.

The SIS also provides continuous long-term, post-accident cooling of the core by recirculation of borated water from the IRWST. Water, drawn from the IRWST by the SI pumps and the containment spray (CS) pumps, is injected into the reactor vessel and the containment. The SI water then enters the containment through the primary pipe break. This water and the CS water return through floor drains and the holdup volume tank to the IRWST. During this process, heat is removed from the IRWST water by the containment spray heat exchanger. The SIS and the IRWST are discussed further in Sections 6.3 and 6.8, respectively.

The SIS is capable of providing an alternate means of decay heat removal for those events beyond the licensing design basis in which the steam generators are not available. Decay heat removal, via feeding and bleeding of the RCS, would be accomplished using the SIS to feed, the Safety Depressurization System (SDS) to bleed, and the Shutdown Cooling System (SCS) for cooling of the IRWST water.

1.2.4.3 Emergency Feedwater System

The Emergency Feedwater System (EFWS) for the System 80+ Standard Design is a dedicated safety system that is designed to perform the following functions:

- Supply feedwater to the steam generators for the removal of heat from the RCS in the event the main feedwater system is unavailable following a transient or accident.

- Supply feedwater to the steam generators for the removal of heat from the RCS in the event of a total loss of AC power (station blackout).

The EFWS consists of two storage tanks, four pumps, and associated piping and valves. Two pumps are motor-driven and two are steam-driven. The EFWS is designed to be automatically or manually initiated.

The EFWS is discussed further in Section 10.4.9.

1.2.4.4 Safety Depressurization System

The Safety Depressurization System (SDS) is a dedicated safety system designed to perform the following functions:

- Provide a safety grade means to depressurize the RCS in the event that pressurizer spray is unavailable during plant cooldown to cold shutdown.

- Provide a capability to rapidly depressurize the RCS to initiate the feed and bleed method of plant cooldown subsequent to a total loss of feedwater.

The system includes the valves and piping which establishes a flow path from the pressurizer steam space to the In-Containment Refueling Water Storage Tank (IRWST). It is manually actuated and controlled.

The SDS is discussed further in Section 6.7.

1.2.4.5 Containment Spray System

The Containment Spray System (CSS) for System 80+ is designed to maintain containment pressure and temperature within design limits in the unlikely event of design basis mass-energy releases to the containment atmosphere.

The CSS is a fully redundant two-train system. Two containment spray pumps supply water through two heat exchangers to the upper region of the containment. Spray headers are used to provide a relatively uniform distribution of spray over the cross sectional area of the containment. The In-Containment Refueling Water Storage Tank (IRWST) is used as the water source for the system. The Containment Spray Pumps can be manually aligned and used as residual heat removal pumps during Shutdown Cooling System (SCS) operation. Likewise, the SCS pumps can be manually aligned to perform the containment spray function.

The Containment Spray System also provides a containment air cleanup function to reduce the concentration of fission products in the containment atmosphere after an accident. No spray additives are required. |

The CS pumps and CS heat exchangers can also be used as a backup to the SCS pumps and heat exchangers to provide cooling of the IRWST water during post-accident feed and bleed operations when the steam generators are not available to cool the RCS. |

The CSS is discussed further in Sections 6.2.2 and 6.5.

The ARTS will initiate a reactor trip when the Pressurizer Pressure exceeds a predetermined value. Its sensors and circuitry including the final actuation devices are diverse from that of the RPS. The ARTS design uses a two-out-of-two logic to open the CEDM motor generator output contactors.

The AFAS will initiate emergency feedwater when the level in either Steam Generator decreases below a predetermined value. Its sensors and circuitry are independent and diverse from that of the PPS Emergency Feedwater Actuation System and the Reactor Protective System.

1.2.5.1.3 Engineered Safety Features Actuation System

The Engineered Safety Features Actuation System (ESFAS) operates in a manner similar to the RPS to automatically actuate the Engineered Safety Feature (ESF) systems. Again, it has a selective two-out-of-four actuation logic that can be converted to a selective two-out-of-three logic. The ESFAS is independent of the control systems.

1.2.5.1.4 Reactor Control Systems

The reactor control systems are used for startup and shutdown of the reactor, and for adjustment of the reactor power in response to turbine load demand. Reactor control functions are performed by the Power Control System (PCS) and the Process-Component Control System (Process-CCS) as described in Section 7.7.1.1. The PCS performs CEDMCS, MDS, RPCS and RRS functions. The Process-CCS performs SBCS, FWCS and pressurizer control functions. Reactor power control is normally accomplished by automatic movement of CEAs in response to a change in reactor coolant temperature, with manual control capable of overriding the automatic signal at any time. If the reactor coolant temperature is different from a programmed value, the CEAs are adjusted until the difference is within the prescribed control band. Regulation of the reactor coolant temperature in accordance with this program, maintains the secondary steam pressure within operating limits and matches reactor power to load demand.

The reactor is controlled by a combination of CEA motion and dissolved boric acid in the reactor coolant. Boric acid is used for reactivity changes associated with large but gradual changes in water temperature, xenon concentration, and fuel burnup. Addition of boric acid also provides an increased shutdown margin during the initial fuel loading and subsequent refuelings. The boric acid solution is prepared and stored at a temperature sufficient to prevent precipitation (maximum boric acid concentration in any storage tank is 2.5 weight percent).

CEA movement provides changes in reactivity for shutdown or power changes. The CEAs are moved by CEDMs mounted on the reactor vessel head. The CEDMs are designed to permit rapid insertion of the CEAs into the reactor core by gravity. CEA motion can be initiated manually or automatically. In addition to full strength CEAs, the System 80+ design provides reduced strength CEAs which can be used for reactivity control during maneuvers, thus minimizing the need for changes in RCS boron concentration during intended maneuvers and operational transients.

The pressure in the Reactor Coolant System is controlled by regulating the temperature of the coolant in the pressurizer where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce variations caused by expansion and contraction of the reactor coolant due to system temperature changes.

The Megawatt Demand Setter (MDS) is a Nuplex 80+ system that automatically controls the response of the station's main turbine to changes in power demand relayed from the utility's grid by the automatic dispatch system. A Steam Bypass Control System (SBCS) is used to dump steam in case of a large

- Vital Instrumentation and Equipment Rooms - four redundant separated rooms which contain their respective channels (A, B, C or D) of safety-related equipment.

- Non-Essential Electrical Equipment Rooms - two separated rooms which contain non-safety related instrumentation, controls and electrical equipment.

The Advanced Control Complex design includes the following major interdependent systems:

- Main Control Panels
- Remote Shutdown Panel
- Discrete Indication and Alarm System
- Data Processing System
- Component Control System
- Plant Protection System
- Power Control System

1.2.6.1 Main Control Panels

The main control panels are designed to permit command by a single individual during normal power operation. However, the main control room design accommodates two control room operators and a supervisor for all normal modes of plant operation and up to the full operating crew during emergencies.

Each main control panel section integrates miniaturized back lighted component control switches, meters, alarms, indicators and Video Display Units (VDUs) (e.g., CRTs, Plasma Displays) such that both safety-related Class 1E and non-Class 1E instrumentation are routinely used by the operator.

Discrete alarms and indicators are provided to allow accident and technical specification monitoring, safe shutdown and other licensing requirements for which the Data Processing System VDUs described in Section 1.2.6.4 cannot be credited. The discrete alarms and indicators are also designed to permit continued plant operation for unlikely instances when the Data Processing System is unavailable.

The panel arrangements and layouts for all controls and indicators on the main control panels are designed, verified and validated in accordance with human factors design guidelines and requirements specified in NUREG-0700. Refer to Chapter 18, Human Factors Engineering, for further information.

A Control Room Supervisor's Monitoring Console, including a DPS driven VDU and sufficient desk space, is provided to support the plant monitoring and daily operational needs of the Control Room Supervisor.

1.2.6.2 Remote Shutdown Panels

The Remote Shutdown Panel (RSP) design includes a minimum of two isolated redundant channels of the safety-related instrumentation and controls necessary to achieve hot standby (mode 3 plant conditions) if the main control room must be evacuated.

In addition to the above, atmospheric steam dump valves are connected to the main steam lines upstream of the main steam line isolation valves to provide the capability to hold the plant at hot standby or, in the event of loss of power to the condenser circulating water pumps, cool the plant down to the point at which the shutdown cooling system may be utilized. These valves are not part of the Turbine Bypass System; no credit for their use is assumed in obtaining the 55% capacity of the Turbine Bypass System.

Overpressure protection for the shell side of the steam generators and the main steam line piping up to the inlet of the turbine stop valve is provided by spring-loaded safety valves. Modulation of the turbine bypass valves discussed earlier would normally prevent the safety valves from opening. The steam bypass system, coupled with the Reactor Power Cutback System, would prevent opening of the safety valves following a turbine and/or reactor trip.

Each steam generator has two steam discharge lines. Each line is provided with a flow measuring device, five spring-loaded safety relief valves, a main steam isolation valve, a power operated atmospheric dump valve and a bypass line and valve around each main steam isolation valve. Each main steam line is provided with a turbine stop valve and a control valve just upstream of the high pressure turbine.

The Steam and Power Conversion System is described further in Chapter 10.

General arrangements for the turbine building are shown in Figures 1.2-13 through 1.2-19.

1.2.9 Heating, Ventilating and Air Conditioning Systems

The Heating, Ventilating and Air Conditioning (HVAC) Systems for all plant buildings are designed for personnel comfort and/or equipment operation with the exception of Annulus Ventilation System. In addition, the following systems have been provided with protection features described below.

- Control Room HVAC Subsystem is designed for uninterrupted safe occupancy of the control room during normal operation and post-accident shutdown.

- Fuel Building Ventilation System is a once through ventilation system designed to limit the radiation release following a fuel handling accident to meet the 10 CFR 100 guidelines. It maintains the building under negative pressure and directs the air flow from less-contaminated to more-contaminated areas before exiting.

- Nuclear Annex and Radwaste Building Ventilation Systems are once through ventilation systems with filtered exhausts. They maintain negative building pressures and direct the air flow from less-contaminated to more-contaminated areas before exiting.

- The Annulus Ventilation System (AVS) serves the space between the primary containment and the secondary containment. The system does not perform any normal ventilation function. However, it does provide additional assurance against the release of radioactivity to the environment; therefore, it is designed as an engineered safety feature and should be capable of operating and performing its function during startup, power operation, hot standby and hot shutdown.

- Subsphere Building Ventilation System is a once through ventilation system designed to filter the post-accident contaminated leakages before exiting to meet the 10 CFR 100 guidelines. It maintains building under negative pressure and directs air flow from cleaner to dirtier areas before exiting.

- The Containment Cooling and Ventilation System is provided with post-accident containment isolation features and filtration units for air cleanup during normal and refueling operations. It limits the radiation release to meet the 10 CFR 100 guidelines in case of a fuel handling accident inside containment.

1.2.10 Fuel Handling and Storage

1.2.10.1 Fuel Handling

Fuel handling equipment provides for the safe handling of fuel assemblies and CEAs under all specified conditions and for the required assembly, disassembly, and storage of reactor vessel head and internals during refueling.

The major components of the system are the refueling machine, the CEA change platform, the fuel transfer system, the spent fuel handling machine, and the new fuel and CEA elevators. This equipment is provided to transfer new and spent fuel between the fuel storage facility, the containment building, and the fuel shipping and receiving areas during core loading and refueling operations. Fuel is inserted and removed from the core using the refueling machine. During normal operations, irradiated fuel and CEAs are always maintained in a water environment.

The principal design criteria specify the following:

- Fuel is inserted, removed, and transported in a safe manner.
- Subcriticality is maintained in all operations.

Fuel handling is further discussed in Section 9.1.4.

1.2.10.2 Fuel Storage

The new fuel and spent fuel storage facilities are described in Sections 9.1.1 and 9.1.2, respectively. Also included in those sections are summaries of the criticality and safety analysis.

1.2.11 Auxiliary Systems

1.2.11.1 Shutdown Cooling System

The Shutdown Cooling System (SCS) is used to reduce the temperature of the reactor coolant, at a controlled rate, from 350°F to a refueling temperature of 120°F and to maintain the proper reactor coolant temperature during refueling. This system utilizes the shutdown cooling pumps to circulate the reactor coolant through two shutdown heat exchangers, returning it to the reactor coolant system. The component cooling water system supplies cooling water for the shutdown cooling heat exchangers.

The SCS for System 80+ has a design pressure of 900 psig. This higher system pressure provides for greater operational flexibility and simplifies concerns regarding system overpressurization. The SCS pumps do not share functions with the SIS.

The SCS is further discussed in Section 5.4.7.

1.2.11.2 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) controls the purity, volume, and boric acid content of the reactor coolant. The CVCS is not required for any safe shutdown or accident mitigation function.

The coolant purity level in the Reactor Coolant System (RCS) is controlled by continuous purification of a bypass stream of reactor coolant. Water removed from the RCS is cooled in the regenerative heat exchanger. From there, the coolant flows to the letdown heat exchanger and then through a filter and a demineralizer where corrosion and fission products are removed. It is then sprayed into the volume control tank and returned by the charging pumps to the regenerative heat exchanger where it is heated prior to return to the RCS loops and reactor coolant pump seal injection.

The CVCS automatically adjusts the amount of reactor coolant in order to maintain a programmed level in the pressurizer. The level program partially compensates for changes in specific volume due to coolant temperature changes and reactor coolant pump controlled seal leakage. (See Section 9.3.4.2 for details.)

The CVCS controls the boric acid concentration in the coolant by a "feed and bleed" method where the purified letdown stream is diverted to a boron recovery subsystem and either concentrated boric acid or demineralized water is sent to the charging pumps. The diverted coolant stream is processed by ion exchange and degasification and flows to a concentrator. The concentrator bottoms are sent to the boric acid storage tank for reuse as boric acid solution and the distillate is first passed through an ion exchanger and then stored for reuse as demineralized water in the reactor makeup water tank.

A description of the CVCS system Boric Acid Storage, Tank Structure, and the Holdup, and Reactor Makeup Water Tanks and Dike is provided in Section 1.2.16.5.

The CVCS for System 80+ incorporates several significant improvements and simplifications including the following:

- Reclassification as a non-safety related system by transferring of previously credited accident mitigation and safe shutdown functions to other dedicated safety systems.
- Improved letdown configuration.
- Improved charging configuration.

Transferring of accident mitigation and safe shutdown functions to other dedicated safety systems has permitted an overall simplification of plant systems. Although not a safety related system, the System 80+ CVCS provides reliable makeup and depressurization capabilities for defense in depth and ease of operation.

System 80+ employs an improved letdown configuration, of which key elements are the following:

- A full pressure letdown heat exchanger.
- Pressure reduction to CVCS operating pressures downstream of the letdown heat exchanger by use of a letdown orifice in series with a letdown flow control valve.

described in Sections 11.2 and 11.4. The gaseous waste system which is located in the nuclear annex is described in Section 11.3.

1.2.16.5 Boric Acid Storage, Holdup, and Reactor Makeup Water Storage Tanks and Dike ~~Structures/Dikes~~

The Boric Acid Storage, Holdup, and Reactor Makeup Water Storage ~~three~~ tanks are in the Chemical and Volume Control System (CVCS), and are of vertical, cylindrical, single-wall stainless steel construction. contained within a common Seismic Category II reinforced concrete dike structure which provides protection against

The Boric Acid Storage Tank is a requirements Seismic Category I, NNS ~~Safety Class 3~~ tank located in the yard. The tank is designed to ASME III, Class 3 and is ~~protected by a concrete structure designed to withstand~~ natural phenomena such as earthquakes, tornadoes, hurricanes and floods. The structure also serves as a containment barrier to limit ~~preclude~~ the release of water due to leaks or spills. The tank is equipped with an overflow line which is of sufficient size to handle a potential storage tank overflow. The Boric Acid Storage Tank ~~Structure~~ is described in Section 3.8.4.1.6. The location is shown on Figure 1.2-1.

The Holdup and Reactor Makeup Water Tanks are non-seismic and non-safety related. Both tanks are designed to API-650 and are located outdoors within the common Seismic Category II reinforced concrete ~~a single~~ dike structure of sufficient height/size to retain potential leakage, up to the complete contents of ~~both~~ tanks in the event of tank ruptures. The dike for these tanks is described in Section 3.8.4.1.11. The dike all location is shown on Figure 1.2-1.

A description of the CVCS system is presented in Section 9.3.4.

1.2.16.6 Condensate Storage Tank/Dike

The Condensate Storage Tank is designed to API-650 and is non-safety related. It is surrounded by a Seismic Category II reinforced concrete dike of sufficient height/size to retain the entire contents of the tank in the event of a tank overflow or tank failure. The Condensate Storage Tank and dike are described in Section 3.8.4.11. The location is shown on Figure 1.2-1.

The Condensate Storage System is described in Section 9.2.6.

1.2.16.7 Diesel Fuel Storage Structure

There are two Diesel Fuel Storage Structures, one on each side of the Nuclear Annex. The structures are Seismic Category I and are designed to withstand fire, sabotage, internally and externally generated missiles, floods, tornados, hurricanes and the Safe Shutdown Earthquake. Each structure contains two, one-half capacity, steel storage tanks, separated by a 2-hour-rated fire barrier. An adjacent steel framed, non-nuclear safety, Seismic Category II equipment room houses auxiliary equipment.

A description of the Diesel Fuel Storage Structure is given in Section 3.8.4.1.4. The building arrangement is shown on Figure 1.2-24.

A more detailed description of the Diesel Fuel Tank Structure~~s~~ is provided in Section 9.5.4.

1.2.16.8 Component Cooling Water Heat Exchanger Structure(s)

The Component Cooling Water (CCW) heat exchanger structure(s) houses the four CCW heat exchangers, associated piping, valves, auxiliaries and sump pumps described in Section 9.2.2. The

Table 1.2-1 System 80+ Improvements Based on Operating Experience (Cont'd.)

4. Human factors (i.e., the man-machine interface) are considered throughout the plant and especially in the control room (Chapter 18).

5. ALARA considerations affect the selection of materials and location of piping and equipment that carry radioactive coolant. For example, specifications for the reactor coolant system materials have been tightened to minimize transport of contamination. Improvements in the steam generator tubing material and access openings greatly reduce radiation exposures for maintenance, testing, and inspection. The overall goal is to maintain personnel exposure to less than 100 man-rems per year for each reactor (Chapter 12).

6. Plant security (i.e., sabotage protection) and fire protection concerns have been directly addressed in determining layouts for plant safety systems (Section 13.6).

Increased RCS Design Margins and Improvements

1. Reactor: The core operating margin has been increased by reducing the normal operating hot leg temperature and revising core parameter monitoring methods. The ability to change operating power level (i.e., maneuver) using control rods only (without adjusting boron concentration in the coolant system) has been provided, simplifying reactivity control during plant load changes and reducing liquid waste processing requirements (Sections 4.3 and 4.4).

2. Reactor Pressure Vessel: The reactor vessel is ring-forged with material specifications that result in a sixty year end-of-life RT_{NDT} well below the current NRC screening criteria. This results in a significant reduction in the number of welds (with resulting reduction in inservice inspection) and eliminates concern for pressurized thermal shock (Section 5.3).

3. Pressurizer: The pressurizer volume is increased to enhance the transient response of the RCS and to reduce unnecessary challenges to safety systems (Section 5.4.10).

4. Steam Generators: The steam generators include ~~Inconel~~ alloy 690 tubes, improved steam dryers, a seventeen percent increase in overall heat transfer area, which includes a ten percent margin and for potential tube plugging. The steam generators have a twenty-five percent larger secondary feedwater inventory to extend the "boil dry" time and improve response to upset conditions. Steam generator improvements also have been added to facilitate maintenance and long term integrity. These include larger and repositioned manways, a standby recirculation nozzle, and a redesigned flow distribution plate (Section 5.4.2).

5. Mechanical improvements based on System 80 startup and operating experience include strengthened reactor coolant pump impellers, redesigned reactor coolant temperature detector thermowells, strengthened reactor vessel upper guide structure, specification of antimony-free reactor coolant pump bearings, strengthened reactor coolant pump shafts, and redesigned steam generator economizer internals.

Table 1.3-1 Comparison of Reactor Characteristics

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Nuclear Design Data						
Structural Characteristics						
Core Diameter, in. (Equivalent)	136.0	136.0	123.0	143.6	143.6	4.2
Core Height, in. (Active Fuel)	136.7	150.0	150.0	150.0	150.0	
H/U, Unlimited Assembly (Hot)	4.34	4.34	4.07	4.23	4.12	4.3
Number of Fuel Assemblies	217	217	177	241	241	
UO_2 Fuel Rod Locations Per Assembly	176 (Batch A)	236 [1]	236 [1]	236 [1]	236 [1]	4.5 (Center in cell)
	164 (Batch B)					
	176/164/164 (Batch C)					
Performance Characteristics						
Loading Technique	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	4 Batch Mixed Central Zone	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	4.3
Fuel Discharge Burnup, MWD/MTU						
Average First Cycle	13,775	12,373	13,600	13,740	15,300	
First Core Average	22,550	21,700	28,124	24,144	31,700	
Fuel Enrichments W/O U-235						
Region 1	2.05	1.27	1.25	1.83	1.8	
Region 2	2.45	2.38	2.06	2.49	2.9	
Region 3	2.99	2.33	2.75	2.95	3.7	
Region 4			3.30			
Control Characteristics						
Critical Boron Concentrations, PPM (Beginning of Life, Rods out)						4.3
Cold, Zero Power, Clean	1120	899	1127	902	1431	
Hot, Zero Power, Clean	1095	832	1061	882	1414	
Hot, Equilibrium Xe, Full Power	725	719	700	516	1006	
Hot, Full Power, Clean	960	952	954	764	1270	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Nuclear Design Data (Cont'd.)						
Control Element Assemblies						
Material (Full/Part Length)	B_4C/SS-Ag-In-Cd	B_4C/Ag-In-Cd	B_4C/Inconel	B_4C/B_4C Inconel	B_4C or Ag-In-Cd/Inconel[2]	4.2
Number of Control Assemblies (Full/Part Length/Part Strength)	77/8/0	83/8/0	65/0/8[3]	76/13/0[3]	68/0/25[3]	4.2
Number of Absorber Rods Per CEA (or RCC) Assembly	5	4,5/5	4 or 12	4 or 12	4 or 12	
Total Rod Worth (Hot) %	≥ 9.60	11.35	16.00	16.76	16.4 (typical)	4.3
Kinetic Characteristics Range Over First Cycle						
Moderator Temperature Coefficient						
$\Delta\rho$/°F (Hot, Full Power), BOL/EOL)	-0.20×10^{-4}/ -1.96×10^{-4}	-0.5×10^{-4}/ -2.3×10^{-4}	-0.4×10^{-4}/ -2.6×10^{-4}	-0.7×10^{-4}/ -2.5×10^{-4}	-0.7×10^{-4}/ -2.6×10^{-4}	4.3
Moderator Pressure Coefficient (BOL) $\Delta\rho$/psi (Hot, Operating),	$+0.30 \times 10^{-6}$	$+0.9 \times 10^{-6}$	$+1.0 \times 10^{-6}$	$+0.69 \times 10^{-5}$	$+0.4 \times 10^{-5}$	
Moderator Void Coefficient (BOL) $\Delta\rho$/% Void (Hot, Operating),	-0.10×10^{-3}	-0.36×10^{-3}	-0.46×10^{-3}	-0.24×10^{-3}	-0.22×10^{-3}	
Doppler Coefficient $\Delta\rho$/°F (Hot, Operating Range)	-1.06×10^{-5}, to -1.46×10^{-5}	-1.13×10^{-5} to -1.87×10^{-5}	-1.34×10^{-5} to -1.52×10^{-5}	-1.18×10^{-5} to -1.66×10^{-5}	-1.52×10^{-5} to -1.63×10^{-5}	
Hydraulic and Thermal Design Parameters						
Total Core Heat Output, MWt	2560	3390	2815	3800	3914	4.4
Total Core Heat Output, Btu/hr.	8737×10^6	1157×10^7	9608×10^6	1297×10^7	1336×10^7	
Heat Generated in Fuel, %	97.5	97.4	97.4	97.4	97.4	
System Pressure, Nominal, psia	2250	2250	2250	2250	2250	
System Pressure, Minimum Steady State, psia	2200	2200	2200	2200	2200	
Hot Channel Factors, Overall						
Heat Flux, F_q	3.00	2.34	2.34	2.34	2.34	~~4.3~~ 4.4
Enthalpy Rise, F_H,	1.65	1.55	1.55	1.55	1.55	
DNB Ratio at Nominal Conditions	2.18 (W-3)	2.07 (CE-1)[4]	2.06 (CE-1)[4]	1.98 (CE-1)[4]	2.00 (CE-1)[4]	4.4

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Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Nuclear Design Data (Cont'd.)						
Coolant Flow						
Total Flow Rate, lb/hr.	139.5×10^6	148.0×10^6	121.5×10^6	164.0×10^6	165.8×10^6	4.4
Effective Flow Rate for Heat Transfer, lb/hr.	134.3×10^6	144.2×10^6	117.9×10^6	159.1×10^6	160.8×10^6	
Effective Flow Area for Heat Transfer, ft^2	53.5	54.7	44.8	60.8	60.8	
Average Velocity Along Fuel Rods, ft/sec.	15.4	16.5	16.8	16.8	16.7	
Average Mass Velocity, lb/hr-ft^2	2.51×10^6	2.64×10^6	2.63×10^6	2.62×10^6	2.65×10^6	
Coolant — Reactor Temperatures[10], °F						
Nominal Inlet, °F [10]	548	553	565	565	556	4.4
Average Rise in Vessel, °F [10]	48	58	56	56	59	
Average Rise in Core, °F [10]	50	59	58	58	61	
Average Temperature in Core, °F [10]	573	583	594	594	587	
Average Temperature in Vessel, °F [10]	572	582	593	593	586	
Hot Channel Outlet, °F [10]	643	642	646	646	644	
Average Film Coefficient, Btu/hr-ft^2-°F	5820	6270	6290	6290	6300	
Heat Transfer at 100% Power						
Active Heat Transfer Area, ft^2	48,416	61,860	52,100	68,320	70,960	4.4
Average Heat Flux, Btu/hr-ft^2	176,000	182,100	179,550	184,800	183,300	
Maximum Heat Flux, Btu/hr-ft^2	527,900	426,300	420,250	432,700	429,100	
Average Thermal Output, kW/ft	5.94	5.33	5.26	5.41	5.36	
Maximum Thermal Output, kW/ft	17.5	12.5	12.4	12.7	12.6	
Maximum Clad Surface Temperature[10] at Nominal Pressure, °F [10]	657	65[illegible]	657	657	657	
Fuel Centerline Temperature[10], °F Maximum at 100% Power	3,780	3180	3180	3,205	3179	
Engineering Heat Flux Factor	1.03	1.03	1.03	1.03	1.03	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Core Mechanical Design Parameters						
Fuel Assemblies						
Rod Bundle Arrangement	14 x 14	16 x 16	16 x 16	16 x 16	16 x 16	4.2
Design	CEA	CEA	CEA	CEA	CEA	↓
Rod Pitch, in.	0.580	0.506	0.506	0.506	0.506	
Cross Section Dimensions, in.	7.972 x 7.972	7.972 x 7.972	7.972 x 7.972	7.972 x 7.972	7.972 x 7.972	
Fuel Weight (as UO_2), lb.	207,269	223,900	188,609	257,245	264,300	
Number of Grids Per Assembly	8	12	11	11	11	
Fuel Rods						
Number of Locations	36,896[5]	49,500[5]	41,772[5]	56,876[5]	56,876[5]	4.2
Outside Diameter, in.	0.440	0.322	0.382	0.382	0.382	↓
Diametral Gap, in.	0.0085	0.007	0.007	0.007	0.0065	
Clad Thickness, in.	0.026	0.025	0.025	0.025	0.025	
Clad Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	
Fuel Pellets						
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	4.2
Diameter, in.	0.3795	0.325	0.325	0.325	0.3255	↓
Length, in.	0.650	0.390	0.390	0.390	0.390	
Control Assemblies						
Cladding Material	NiCrFe Alloy	NiCrFe Alloy	NiCrFe Alloy	NiCrFe Alloy	NiCrFe Alloy	4.2
Clad Thickness, in.	0.040	0.035	0.035	0.035	0.035	
Core Structure						
Core Barrel ID/OD, in.	148/152	148/153	138/143	157/162	157/162	4.2
Reactor Coolant System Code Requirements						
Component						
Reactor Vessel	ASME III, Class A	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	5.2, 5.4

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Reactor Coolant System Code Requirements (Cont'd.)						
Steam Generator						
Tube Side	ASME III, Class A	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	5.2, 5.4
Shell Side	ASME III, Class A	ASME III, Class 1	ASME III, Class 1	ASME III, Class 2	ASME III, Class 2	
Pressurizer	ASME III, Class A	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Pressurizer Safety Valves	ASME III	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Reactor Coolant Piping	ANSI B31.7	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	ASME III, Class 1	
Principal Design Parameters of the Reactor Vessel						
Material	Low alloy steel internally clad with austenitic SS.	Low alloy steel clad with austenitic SS.	Low alloy steel clad with austenitic SS.	Low alloy steel clad with austenitic SS.	Low alloy steel clad with austenitic SS.	5.3
Design Pressure, psia	2500	2500	2500	2500	2500	
Design Temperature, °F	650	650	650	650	650	
Operating Pressure, psia	2250	2250	2250	2250	2250	
Inside Diameter at Shell, in.	172	172	162	182-1/4	182-1/4	
Outside Diameter Across Nozzles, in.	253 1/16	253 1/16	250 3/4	271	271	
Overall Height of Vessel and Enclosure Head, ft-in. to top	41 11-3/4	43 4-5/8	48 0-1/2	48 7-7/8 (including bottom instrumentation nozzles)	48 7-7/8 (including bottom instrumentation nozzles)	
Minimum Clad Thickness, in.	1/8	1/8	1/8	1/8	1/8	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Principal Design Parameters of the Reactor Coolant Piping						
Material	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	Carbon steel internally clad with stainless steel	5.4.3
Hot Leg - I.D., in.	42	42	42	42	42	
Cold Leg - I.D., in.	30	30	30	30	30	
Between Pump and Steam Generator - I.D., in.	30	30	30	30	30	
Design Pressure, psia	2500	2500	2500	2500	2500	
Principal Design Parameters of the Reactor Coolant System						
Operating Pressure, psia	2250	2250	2250	2250	2250	5.1, 5.4
Reactor Inlet Temperature, °F [10]	548	553	565	565	556	
Reactor Outlet Temperature, °F [10]	596	611	622	621	615	
Number of Loops	2	2	2	2	2	
Design Pressure, psia	2500	2500	2500	2500	2500	
Design Temperature, °F	650	650	650	650	650	
Hydrostatic Test Pressure (cold), psia	3125	3125	3125	3125	3125	
Total Coolant Volume, ft^3	11,101	10,300	11,315	13,125	15,943	
Total Reactor Flow, gal/min [8]	370,000	396,000	330,000	445,600	444,650	
Principal Design Parameters of the Reactor Coolant Pumps						
Number of Units	4	4	4	4	4	5.4.1
Type	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	
Design Pressure, psia	2500	2500	2500	2500	2500	
Design Temperature, °F	650	650	650	650	650	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Principal Design Parameters of the Reactor Coolant Pumps (Cont'd.)						
Operating Pressure, nominal psia	2235	2250	2250	2250	2250	5.4.1
Suction Temperature, °F [1D]	548	553	565	565	556	
Design Capacity, gal/min.	92,500	99,000	82,500	111,400	111,160	
Design Head, ft	300	310	340	365	374	
Hydrostatic Test Pressure, (cold) psia	3125	3125	3125	3125	3125	
Motor Type	AC Induction Single Speed	AC Induction Single Speed	AC Induction Single Speed	AC Induction Single Speed	AC Induction Single Speed	
Motor Rating, hp. (cold)	7,200	9,700	8,500	12,000	12,000	
Principal Design Parameters of the Steam Generators						
Number of Units	2	2	2	2	2	5.4.2
Type	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator	Vertical U-tube with integral moisture separator and economizer	Vertical U-tube with integral moisture separator and economizer	
Tube Material	Inconel	SB-163 NiCrFe alloy	SB-163 NiCrFe alloy	SB-163 NiCrFe alloy	SB-163 NiCrFe alloy 690	
Shell Material	Primary side - low alloy steel clad with austenitic stainless steel Secondary side - carbon steel	Primary side - low alloy steel clad with austenitic stainless steel Secondary side - carbon steel	Primary side - low alloy steel clad with austenitic stainless steel Secondary side - carbon steel	Primary side - low alloy steel clad with austenitic stainless steel Secondary side - carbon steel	Primary side - low alloy steel clad with austenitic stainless steel Secondary side - alloy steel except top head is carbon steel	
Tube Side Design Pressure, psia	2500	2500	2500	2500	2500	
Tube Side Design Temperature, °F	650	650	650	650	650	
Tube Side Design Flow, lb/hr per steam generator	61×10^6	74×10^6	61×10^6	82×10^6	82.9×10^6	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Principal Design Parameters of the Steam Generators (Cont.d) (Cont'd.)						
Shell Side Design Pressure, psia	1000	1100	1270	1270	1200	5.4.2
Shell Side Design Temperature, °F	550	560	575	575	570	
Operating Pressure, Tube Side, Nominal, psia	2235	2250	2250	2250	2250	
Operating Pressure, Shell Side, Maximum, psia	900	1000	1170	1170	1100	
Maximum Moisture at Outlet at Full Load, %	0.20	0.20	0.25	0.25	0.25	
Hydrostatic Test Pressure, Tube Side (cold), psia	3110	3125	3125	3125	3125	
Steam Pressure, psia, at Full Power	850	900	1070	1070	1000	
Steam Temperature, °F at Full Power	525	532	553	553	545	
Steam Flow, at Full Power, lb/hr per steam generator	5.630×10^6	7.565×10^6	6.364×10^6	8.590×10^6	8.82×10^6	
Containment System Parameters						
Type	Steel-lined prestressed post-tensional concrete cylinder, curved dome roof.	Steel-lined prestressed post-tensional concrete cylinder, curved dome roof.	Note [7]	Note [7]	Steel spherical containment shell, surrounded by reinforced concrete shield building.	3.8
Design Parameters - Containment						
Inside Diameter, ft	130	150	Note [7]	Note [7]	200	3.8
Height, ft	181-2/3	240	Note [7]	Note [7]	N/A	
Free Volume, ft^3	2.000×10^6	2.335×10^6	Note [7]	Note [7]	3.4×10^6 [8]	
Design ~~Reference Incident~~ Pressure, psia	65	75	Note [7]	Note [7]	68	
Steel Thickness, in. (approx.)						
Vertical Wall	1/4	1/4	Note [7]	Note [7]	1-3/4	3.8
Hemispherical Head	1/4	1/4	Note [7]	Note [7]	N/A	

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Containment System Parameters (Cont'd.)						
Concrete Thickness, ft						
Vertical Wall	3-3/4	4-1/3	Note [7]	Note [7]	N/A	3.8
Dome	3-1/4	3-3/4	Note [7]	Note [7]	N/A	
Containment Leak Prevention and Mitigation Systems	Leak tight penetration and continuous steel liner. Automatic isolation where required.	Leak tight penetration and continuous steel liner. Automatic isolation where required. Continuous steel liner exhaust from penetration room to vent stack.	Note [7]	Note [7]	Leak tight penetrations. Spherical steel containment with concrete shield building. Automatic isolation where required. Annulus ventilation system maintains negative pressure between containment and shield building.	
Engineered Safety Features						
Safety Injection System						
No. of High Head Pumps, No.	3	3	2	2	4	6.3
No. of Low Head Pumps, No.	2	2	2	2	0	
Safety Injection Tanks, No.	4	4	4	4	4	
Emergency Power						
Standby Generator Units	3 total for 2 units	4 total for 2 units	Note [7]	Note [7]	3[9]	8.3
Instrumentation and Control Systems						
Reactor Protective System	Sec. 7.2	Sec. 7.2	Sec. 7.2	Sec. 7.2	Sec. 7.2	7.2
Initiating Reactor Trip						
Number of Manual Switches	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each in both MCR and at RSP	7.2

Table 1.3-1 Comparison of Reactor Characteristics (Cont'd.)

	2570 MWt	3410 MWt	System 80 2825 MWt	System 80 3817 MWt	System 80+ 3931 MWt	Reference Section
Instrumentation and Control Systems (Cont'd.)						
Automatic Initiation Parameter Channels/Logic	2 of 4 Logic for each trip	2 of 4 Logic for each trip	4 channels provided, coincidence of 2 required for trip	4 channels provided, coincidence of 2 required for trip	4 channels provided, coincidence of 2 required for trip	7.2
ESFAS						
Initiating ESFAS						
Number of Manual Switches	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each	2 Sets of 2 each	7.3
Automatic Initiation Parameter Channels/Logic	2 of 4 Logic for each function	2 of 4 Logic for each function	4 Channels provided, coincidence of 2 required for each function	4 Channels provided, coincidence of 2 required for each function	4 Channels provided, coincidence of 2 required for each function	

[1] In the first core, some UO_2 rods maybe replaced by burnable absorber rods.
[2] Inconel part strength CEAs in System 80+
[3] Locations are provided for 8 additional CEAs.
[4] Minimum DNBR at nominal conditions.
[5] Some of the rod locations are occupied by burnable absorber rods.
[6] Design minimum.
[7] See Site-Specific SAR.
[8] Nominal
[9] 2 Class 1E and 1 Non-Class 1E
[10] Temperatures are given to nearest degree.

1.4 Identification of Agents and Contractors

1.4.1 Applicant's Qualifications and Experience

(Presented in site-specific SAR.)

1.4.2 Architect-Engineer's Qualifications and Experience

(Presented in site-specific SAR.)

1.4.3 Combustion Engineering's Qualifications and Experience

Combustion Engineering, Inc. (hereafter referred to as C-E, Combustion, ABB Combustion Engineering Nuclear Power or ABB-CE) nuclear power activities are of three general types: design, development, construction and operation of reactor and auxiliary systems; design and fabrication of nuclear components; and, support of design, development and analytical projects.

A summary of the company's efforts, accomplishments, and operating experience in the light water reactor field is provided below.

1.4.3.1 Pre-Commercial Reactor Programs

1.4.3.1.1 Naval Propulsion Program

During the period 1955 through 1960, Combustion Engineering was a major contributor to the U.S. Naval Reactors program. The Company designed and built, at its Windsor, Connecticut site, the prototype of a small attack submarine power plant. This prototype, S1C, went into operation in 1959 as a naval training facility. A second plant of this type, also designed and built by Combustion Engineering, was installed in the USS Tullibee (SSN-597) and operated as a part of the United States nuclear submarine fleet.

In the design, development, construction and operation of the prototype system and the submarine power plant, Combustion's responsibilities included all safety aspects of the reactor systems.

1.4.3.1.2 Boiling Nuclear Superheat (BONUS) Plant

Combustion Engineering was responsible for the nuclear design and for the direction of startup and initial operation of the BONUS plant in Puerto Rico.

The design of this reactor system presented a number of unique problems, e.g., control and safety analysis of a two-region core, design of a superheater fuel element, design of a steam control system to assure adequate cooling of superheater fuel under all credible conditions, and design of a containment building of the "total containment" type to house the entire power generating installation.

The BONUS plant achieved full power operation in September 1965, and was the first nuclear power plant under U.S. AEC control operating with an integral superheating core.

1.4.3.2 Development and Design of Commercial PWR Systems

The development and design ~~by Combustion~~ of a pressurized water reactor by Combustion Engineering for utility service dates back to 1958. At that time, the Company was selected by the AEC to undertake the design, analysis and economic evaluation of a 250 MWe PWR plant, in conjunction with an architect-engineer. This effort provided initial technical and economic guidelines for Combustion's commercial development of the PWR.

With a subsequent decision by the Company to concentrate on the development of the PWR for large nuclear power stations, a program was initiated to guide required design and development work along appropriate lines. The following is representative of the types of PWR-oriented work which have been performed:

- Evaluation of overall plant and systems to establish optimum physical arrangement and design criteria from the standpoint of economics and safety. Much of this work has been performed in conjunction with qualified architect-engineering organizations;

- Design and development of nuclear components such as control element assemblies, control element assembly drive mechanisms, and auxiliary systems equipment.

- Extensive testing of PWR nuclear components, such as fuel assemblies and reactor control components, under actual service pressure, temperature and flow conditions.

Combustion Engineering's Nuclear Laboratories have been engaged in the development and testing of fuels, fuel elements, control assemblies, reactor components and materials for reactor application. Particular emphasis has been given to UO_2 and Zircaloy cladding technology, involving both in-pile and out-of-pile investigations. The initial efforts in the laboratories were associated with submarine reactor programs. Beginning in 1960, nuclear laboratory personnel actively participated in the joint U.S. AEC - Euratom research and development program for fuels development. In addition to these programs, laboratory personnel were responsible for materials design activities for the heavy water organically-cooled reactor study and for pressurized water, boiling water, nuclear superheat, and fast breeder reactor systems.

1.4.3.3 Major Component Design and Fabrication

Between 1955 and 1961, C-E was a major supplier of nuclear cores for naval propulsion service.

C-E fabricated the boiling and the superheating fuel for the BONUS reactor. The boiling section of the BONUS core contained Zircaloy-clad, rod type, UO_2 fuel elements fundamentally similar to those utilized in the C-E Standard Plant fuel design. The superheater fuel utilized Inconel-clad, rod type, UO_2 fuel elements; the superheater cladding was designed for an operating temperature of 1250°F.

Combustion Engineering has performed the design engineering and fabrication of control rod drive mechanisms and fuel rods for all reactors listed in Table 1.4-1.

Many reactor vessels designed for utility plant or for naval service have been fabricated by Combustion Engineering. Reactor vessels for plant sizes up through 1300 MWe are now in service.

Table 1.8-4 Deviations from the U.S. NRC Standard Review Plan (Cont'd.)

SRP Section/Title		Comment or Summary Description of Deviation	Section
11.1	Source Terms - Rev. 2, July 1981	Cost-benefit analysis for radioactive waste management systems is defered to the site-specific application.	11.1
		Cost-benefit analysis for radwaste augments used in the calculation of effluent releases to the environment is defered to the site-specific application.	
11.2	Liquid Waste Management Systems - Rev. 2, July 1981	Cost-benefit analysis for liquid waste management systems is deferred to the site-specific application due to the site-specific nature of population dose analyses.	11.2.6.4
		The plant transients which might occur less frequently than once per fuel cycle are not taken into account for the design of waste collection tanks and waste sample tanks.	11.2.2
11.3	Gaseous Waste Management Systems - Rev. 2, July 1981	Cost-benefit analyses for gaseous waste management systems is deferred to the site-specific application.	11.3.6.5
12.2	Radiation Sources - Rev. 2, July 1981	The shielding analysis will be performed subsequent to component procurement and detailed piping design (layout).	
12.3; 12.4	Radiation Protection Design Features - Rev. 2, July 1981	The shielding analysis will be performed subsequent to component procurement and detailed piping layouts.	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR) - Rev. 2, July 1981	Fuel rod failures are assumed based on the DNB convolution method.	4.4.4.1
		Loss of offsite power subsequent to turbine trip is assumed to occur three seconds after turbine trip.	15.1.5.1; 15.1.5.3
		Leak-Before-Break analysis and criteria are applied to the Main Steam Line.	3.6.2.2.1
15.3.3; 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break - Rev. 2, July 1981	The assumption of coincident turbine trip, loss of offsite power, and coastdown of damaged pumps is not made. Loss of offsite power after turbine trip is assumed to occur 3 seconds after turbine trip.	15.3.3.2 15.3.3.3

Table 1.8-6 System 80+ Industrial Codes and Standards

Code	Edition	Title
ANSI/American Concrete Institute [ACI]		
318	1989	Building Code Requirements for Reinforced Concrete, 1991 Printing
[[*349*	*1985*	*Code Requirements for Nuclear Safety-Related Concrete Structures*]][1]
ANSI/American Institute of Steel Construction [AISC]		
[[*N690*	*1984*	*Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*]][1]
	1989	Manual of Steel Construction, Allowable Stress Design, Ninth Edition
ANSI/American Nuclear Society [ANS]		
2.8	1992	Determining Design Basis Flooding of Power Reactor Sites
13.1	1993	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
51.1	1983	Nuclear Safety Criteria for the Design of Stationary PWR Plants
55.4	1993	Gaseous Radioactive Waste Processing Systems for Light Water Reactor Plants
56.2	1989	Containment Isolation Provisions for Fluid Systems after a LOCA
57.2	1976	Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations
58.1	1982	Plant Design Against Missiles
58.2	1988	Design Basis for Protection of LWRs against Effects of Pipe Rupture
58.8	1984	Time Response Design Criteria for Safety-Related Operator Action
58.9	1987	Single Failure Criteria for LWR Safety Related Fluid Systems
ANSI/American Petroleum Institute [API]		
650	1988	Welded Steel Tanks for Oil Storage
ANSI/American Society of Civil Engineers		
7	1990	Minimum Design Loads for Building and Other Structures [ANSI A58.1]
ANSI/American Society of Mechanical Engineers [ASME]		
BPVC	1989	Section II; Materials Specifications
[[*BPVC*	*1989*	*Section III; Rules for Construction of Nuclear Power Plant Components; Division I*]][1], Division II
BPVC	1989	Section V, Non-Destructive Examination
BPVC	1989	Section VIII; Rules for Construction of Pressure Vessels
BPVC	1989	Section IX; Qualification Standard for Welding and Brazing
BPVC	1989	Section XI; Rules for Inservice Inspection of Nuclear Power Plant Components; Editions and Addenda As Applicable
AG-1	1991	Code on Nuclear Air and Gas Treatment
B31.1	1992	Power Piping
OM-S/G	1990	Standards and Guides for Operation and Maintenance of Nuclear Power Plants; through 1992 Addenda.

[1] NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

Table 1.8-8 Cross-Reference for Unresolved and Generic Safety Issues (Cont'd.)

No.	Title	Section[1]
83	Control Room Habitability	6.4; 9.4.1; 20.2.23
87	Failure of HPCI Steam Line Without Isolation (in BWRs)	3.9.6; 20.2.24
93	Steam Binding of Auxiliary Feedwater Pumps	10.4.9.5.2; 20.2.25
94	Additional LTOP for Light Water Reactors	5.2.2.10; 5.2.3; 5.3; 20.2.26
99	Loss of RHR Capability in PWRs	5.4.7; App 19.8A; 20.2.27
103	Design for Probably Maximum Precipitation	2.0; 3.1.2; 20.2.28
105	Interfacing Systems LOCA at LWRs	App 5E; 20.2.29
106	Piping and Use of Highly Combustible Gases in Vital Areas; Fire Protection	9.5.10; 20.2.30
113	Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers	3.9.3.4; 20.2.31
118	Tendon Anchorage Failure	3.8; 20.2.32
119.1	Pipe Rupture Requirements	3.6.2.1; 3.9.2.5; 3.9.3.1; 20.2.33
119.2	Pipe Damping Values	3.7.1.3; 20.2.34
119.3	Decoupling OBE from SSE	2.5; 3.7; 20.2.35
119.5	Leak Detection Requirements	3.6.3.3; 5.2.5; 7.7.1.6; 20.2.36
120	On-Line Testability of Protection Systems	Ch 16; 20.2.37
121	Hydrogen Control for Large, Dry PWR Containments	3.8; 6.2.5; 19.11; 20.2.38
122.2	Initiating Feed and Bleed	7.5.1.1.5; 10.4.9; 20.2.39
124	Auxiliary Feedwater System Reliability	10.4.9; 20.2.40
125.I.3	SPDS Availability	7.5; 7.7.1; 18.7.1; 20.2.41
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During Line Break	10.4.9; 20.2.42
128	Electrical Power Reliability	8.3; 20.2.43
130	Essential Service Water Pump Failure at Multiplant Sites	1.2.1.3; 9.2.1; 20.2.44
135	Integrated Steam Generator Issues	5.4.2; 6.7.2; 7.3.1; 7.5.1; 10.3.2; 10.4; 15.6.3; 20.2.45
142	Leakage Through Electrical Isolators in Instrumentation Circuits	20.2.46

Table 1.8-8 Cross-Reference for Unresolved and Generic Safety Issues (Cont'd.)

No.	Title	Section[1]
II.E.4.2	Containment Design; Isolation Dependability	3.1; 6.2.4; 9.4.6; 20.2.112
II.E.4.4	(1-5) Containment Design; Purging	6.2.4; 9.4.6; Ch 16; 20.2.113
II.E.6.1	In-situ Testing of Valves--Test Adequacy Study	3.9.6; 20.2.114
II.F.1	Additional Accident Monitoring Instrumentation	7.5.1.1.5; 20.2.115
II.F.2	Identification and Recovery from Conditions Leading to Inadequate Core Cooling	7.5.1.1.7; 20.2.116
II.F.3	Instrumentation for Monitoring Accident Conditions	7.5.1.1.5; 20.2.117
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	6.7; 7.5.2.5; 20.2.118
II.J.3.1	Organization and Staffing to Oversee Design and Construction	20.2.119
II.K.1 (1)	(1,2,4(A-C),5,7,8,10-13, 17-23) Measures to Mitigate Small Break LOCAs and Feedwater Accidents; NRC Bulletins	6.3.3; 20.2.120
II.K.1 (2)	(3,4d,6,9,14-16,24-28) Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents; NRC Bulletins	6.3.3; 20.2.120
II.K.3	(2,5,6,8,25,30,31,55) Final Recommendations of B&O Task Force to Mitigate Accidents	6.3.3; 9.3.4.3.2; 19.3.3.1; 20.2.121
III.A.1.2	(1-3) Upgrade Licensee Emergency Support Facilities	13.3.3.1; 13.3.3.2; 13.3.3.3; 20.2.122-124
III.D.1.1	(2) Review Information on Provisions for Leak Detection	11.5; 12.1; 12.3; 20.2.125
III.D.3.3	(1-4) In-Plant Radiation Monitoring	7.5.1.1.5; 11.5 12.3.4; 20.2.126
III.D.3.4	Control Room Habitability	6.4; 20.2.127

[1] USIs and GSIs not applicable to the System 80+ Standard Plant Design are identified in Chapter 20, along with the corresponding reason.

Table 1.8-9 Cross-Reference for the TMI Rule (10 CFR 50.34f)

No.	Title	Section[1]
(1)(i)	Plant/Site Specific PRA (II.B.8)	19.15; 20.2.104
(1)(ii)(A)	Simplified AFWs Reliability Analysis (II.E.1.1)	10.4.9; 19.6; 20.2.108
(1)(ii)(B)	Design Review of AFWs (II.E.1.1)	10.4.9; 20.2.108
(1)(ii)(C)	Evaluation of AFWs Flow Design Bases and Criteria	10.4.9
(1)(iii)	Evaluation of RCP Seal Damage Following Small Break LOCA with LOP (II.K.2.16 & II.K.3.25)	5.4.1.3; 8.1.4.2; 9.3.4; 20.2.121
(1)(iv)	Analysis of Probability of Small Break LOCA Caused by PORV (II.K.3.2)	19.15.2.1.2; 20.2.121
(1)(v)	Evaluation of Effectiveness of High Pressure Coolant Injection (BWRs Only) (II.K.3.16)	N/A
(1)(vi)	Reduction of Challenges to Relief Valves (BWRs Only) (II.K.3.16)	N/A
(1)(vii)	Feasibility and Risk Assessment of ADS Design Modifications (BWRs Only) (II.K.3.18)	N/A
(1)(viii)	Effect of Core Cooling Modes Under Accident Conditions (BWRs Only) (II.K.3.21)	N/A
(1)(ix)	Study of Additional Space Cooling Needs for RCIC & HPIC (BWRs Only) (II.K.3.24)	N/A
(1)(x)	Study ADS Capability During and Following Accident Conditions (BWRs Only) (II.K.3.28)	N/A
(1)(xi)	Evaluate Alternate Depressurization Methods (BWRs Only) (II.K.3.45)	N/A
(1)(xii)(A)	Compare Costs and Benefits of Alternative Hydrogen Control Systems	~~10A.5.9~~ 6.2.5
(1)(xii)(B)	Verify Compliance with ~~F)(1)(Ix)~~ (f)(2)(ix) of Selected Hydrogen Control System	6.2.5.1.2; App 19.11K
(1)(xii)(C)	Evaluate Design, Function & Layout of Alternative Hydrogen Control Systems	6.2.5.1.2; App 19.11K
(2)(i)	Simulator Capability (COL Requirement) (I.A.4.2)	Ch 20
(2)(ii)	Program to Improve Procedures (COL Requirement) (I.C.9)	20.2.89
(2)(iii)	Control Room Design (I.D.I.)	18.0; 20.2.90
(2)(iv)	Safety Parameter Display Console (I.D.2)	7.7.1.4; 7.7.1.7; 13.3.3; 18.7.1; 20.2.91
(2)(v)	Indication of Bypassed and Operable Status of Safety Systems (I.D.3)	7.1.2.21; 20.2.92

Table 1.8-9 Cross-Reference for the TMI Rule (10 CFR 50.34f) (Cont'd.)

No.	Title	Section[1]
(2)(xvii)	Provide Instrumentation to Measure, Record and Indicate in the Control Room (II.F.1)	7.5.1.1.5; 20.2.115
(2)(xviii)	Control Room Indication of Inadequate Core Cooling - Saturation Meter (II.F.2)	7.5.1.1.7; 20.2.116
(2)(xix)	Provide Post-Accident Monitoring Instrumentation (II.F.3)	7.5.1.1.5; 20.2.117
(2)(xx)	Power Supplies for RCS Relief and Block Valves and Level Indicators (II.G.1)	6.7; 7.5.2.5; 20.2.118
(2)(xxi)	Auxiliary Heat Removal System Design (BWRs Only) (II.K.1.22)	N/A
(2)(xxii)	FMEA on Integrated Control System (B&W Only) (II.K.2.9)	N/A
(2)(xxiii)	Anticipatory RPS Trip on Loss of MFW and Turbine Trip (B&W Only) (II.K.2.10)	N/A
(2)(xxiv)	Recording of Post-Accident Reactor Vessel Water Level (BWRs Only) (II.K.3.23)	N/A
(2)(xxv)	Onsite Technical Support Center, and Technical Operations Center, and Emergency Operations Facility (III.A.1.2)	13.3.3.1; 13.3.3.2; 13.3.3.3; 20.2.122-124
(2)(xxvi)	Leakage Control and Detection Design and Program for Systems Outside Containment (III.D.1.1)	11.5; 12.1; 12.3; 20.2.125
(2)(xxvii)	Monitoring of Inplant and Airborne Radioactivity (III.D.3.3)	7.5.1.1.5; 11.5; 12.3.4; 20.2.126
(2)(xxviii)	Evaluate Potential Pathways That May Lead to Control Room Habitability Problems Under Accident Conditions (III.D.3.4)	6.4; 20.2.127
(3)(i)	Administrative Procedures for Evaluating Industry Operating, Design, and Construction Experience During Design and Construction (I.C.5)	N/A
(3)(ii)	Ensure that QA List Contains All Systems, Structures and Components Important to Safety Per Criterion II of 10 CFR 50 Appendix B (I.F.1)	3.2; 17.1; 20.2.95
(3)(iii)	Quality Assurance Program (I.F.2)	17.1; 20.2.99
(3)(iv)	Provision of Dedicated Containment Penetration for Future Installation of Systems to Prevent Containment Failure (II.B.8)	App 19A.5.2; 20.2.104
(3)(v)(A)	Containment Integrity During Hydrogen Burn (or Inerting) for 100% Clad/Metal - Water Reaction (II.B.8)	3.8.2; App 19.11K; 20.2.104
(3)(v)(B)	Containment Structural Loading from Inadvertent Actuation of Inerting System (II.B.8)	N/A
(3)(vi)	External Hydrogen Recombiners (II.E.4.1)	6.2.4; 6.2.5; 20.2.111
(3)(vii)	Management Plan for Design and Construction Activities (COL Requirement) (II.J.3.1)	20.2.119

[1] N/A indicates items which are Not Applicable to the System 80+ Standard Plant Design.

Table 1.8-10 Cross-Reference for New NRC Policy Issues (SECY-93-087)

No.	Title	Section[1]
I.A	Use of a Physically Based Source Term	3.11; 6.5; App 15A
I.B	Anticipated Transients Without Scram	7.3; 7.7.1.1.11
I.C	Mid-Loop Operation	App 19.8A
I.D	Station Blackout	8.1.4.2; 8.3.1.1.5
I.E	Fire Protection	9.5.1
I.F	Intersystem Loss-of-Coolant Accident	App 5E
I.G	Hydrogen Control	6.2.5; 19.11.3
I.H	Core Debris Coolability	19.11.3
I.I	High-Pressure Core Melt Ejection	19.11.3
I.J	Containment Performance	19.11.3
I.K	Dedicated Containment Vent Penetration	19.15.5
I.L	Equipment Survivability	19.11.4.4
I.M	Elimination of Operating-Basis Earthquake	2.5; 3.7
I.N	Inservice Testing of Pumps and Valves	3.9.6; 5.2.4; 6.6
II.A	Industry Codes and Standards	1.8
II.B	Electrical Distribution	8.2; 8.3
II.C	Seismic Hazard Curves and Design Parameters	19.7.5
II.D	Leak-Before-Break	3.6.2.1.3; 3.6.3
II.E.	Classification of Main Steamlines in Boiling Water Reactors	N/A
II.F	Tornado Design Basis	2.3.2.1
II.G	Containment Bypass	6.2.2; App 5~~E~~ F
II.H	Containment Leak Rate Testing	3.8.2.7
II.I	Post-Accident Sampling System	9.3.2
II.J	Level of Detail	1.1.1; 1.2.2
II.K	Prototyping	N/A
II.L	ITAAC	14.3
II.M	Reliability Assurance Program	17.3

Table 1.8-10 Cross-Reference for New NRC Policy Issues (SECY-93-087) (Cont'd.)

No.	Title	Section[1]
II.N	Site-Specific Probabilistic Risk Assessments and Analysis of External Events	17.3; 19.7; 19.15
II.O	Severe Accident Mitigation Design Alternatives	19.15.5; App 19A
II.P	Generic Rulemaking Related to Design Certification	N/A
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	7.2; 7.3; 7.7; App 7A
II.R	Steam Generator Tube Rupture	19.15.2.1.2
II.S	PRA Beyond Design Certification	17.3; 19.7; 19.15
II.T	Control Room Annunciator (Alarm) Reliability	7.7
III.A	Regulatory Treatment of Nonsafety Systems in Passive Designs	N/A
III.B	Definition of Passive Failure	N/A
III.C	SBWR Stability (Passive Design)	N/A
III.D	Safe Shutdown Requirements (Passive Design)	N/A
III.E	Control Room Habitability (Passive Design)	N/A
III.F	Radionuclide Attenuation (Passive Design)	N/A
III.G	Simplification of Offsite Emergency Planning	15.6.5
III.H	Role of the Passive Plant Control Room Operator	N/A

[1] "N/A" indicates items which are Not Applicable to the System 80+ Standard Plant Design.

Table 1.10-1 COL License Information (Cont'd.)

COL No.	FSER No.	Section	Subject
5-2	5.2.2.2-1, 5.2.2.3-1, 5.3.1-1, 5.3.2-1	5.2.2.10.2.2, 5.2.3.1, 5.3.2, 5.4.14.3	Verification of the material properties and end-of-life fluence and resulting P-T limits and LTOP temperatures
5-3	5.2.4-1, 6.6-1	5.2.4, 6.6.1, 12.3.1.2	PSI and ISI program plans for NRC staff review
5-4	5.4-1	5.4.2.5	Steam Generator tube inservice inspection program
5-5	5F-1	5F (5.6.3), 6.4.1.2	Leakage monitoring program
6-1	6.1.1-1, 6.1.1-2, 6.1.1-3, 6.1.1-4	6.1.1.1, 5.2.3.3.2.1	Engineered Safety Feature Systems materials selection and fabrication
6-2	6.3.7-1	6.3.4.1, 6.3.4.2	Periodic testing of the safety injection system
6-3	6.4-2	6.4.1.2, 6.4.2.2 14.2.12.1.103	Protection against the effects of toxic substance releases (including TMI III.D.3.4)
6-4	6.4-3	6.4.1.1	Control room habitability system
6-5		6.4.1.2	Pump seal leakage procedure
6-6	6.5-1	6.5.4.1, 6.5.5	Containment spray system operability
6-7	6.2.4-1	Table 6.2.4-1	Containment isolation details
6-8		6.8.2.2.1	IRWST screen area margin analysis
7-1		7.1.2.7	Integrated response time for protection system
7-2		7.4.1.1.8.2	Operating procedures for SCS
7-3		7.3.1.1.10	Procedures for removing ESFAS signals during plant testing
7-4		7.3.2.1	Procedures for ESFAS Reset
7-5		7.3.2.3.2	ESFAS setpoint analyses
7-6		7.4.2.5.2	Cold shutdown procedures
7-7		7.5.2.5.10	Administrative controls associated with PAMI
8-1	8.3.1-1	8.1.4.5, 8.3.1.1.6, Tbls 8.3.1-2,3,4 Tbls 8.3.2-3,4	Electrical power systems sizing, testing, calibration and maintenance
9-1		9.1.2.2.2, 9.1.4.2, 9.1.4.3, 9.1.4.4, 9.1.4.6	Administrative controls and procedures associated with fuel storage and handling systems
9-2	9.2.1-1	9.2.1.1.4, 9.2.1.4, 9.2.5.4, 20.2.13	Organic fouling and inorganic buildup in the SSWS (including GSI-51)
9-3	9.2.1-2	9.2.1.2.1.2	Station service water system pump structure
9-4	9.2.4-1	9.2.4.2	Potable and sanitary water systems
~~9-5~~		~~9.3.4.1.4~~	~~Structures housing boric acid storage tank,~~ reactor makeup water tank, and holdup tank

Table 1.10-1 COL License Information (Cont'd.)

COL No.	FSER No.	Section	Subject
9-~~6~~5	9.5.1.5-1	9.5.1.11, 19.15.3.2	Administrative controls for BTP CMEB 9.5-1 conformance and fire brigade
9-~~7~~6	9.5.1.5-1	9.5.1.12	Fire Hazards Analysis
9-~~8~~7	9.5.2-1, 9.5.2-2, 9.5.2-3	9.5.2.1, 9.5.2.2.5, 9.5.2.2.6	Communications systems
9-~~9~~8		9.5.3.2.2	Security lighting system
9-~~10~~9	9.5.4.1-1	8.3.1.1.4.11, 8.3.1.1.4.13	Diesel operator training
9-~~11~~10	9.5.4.1-2, 9.5.4.2-1, 9.5.5-1, 9.5.5-2, 9.5.6-1, 9.5.6-2, 9.5.7-1, 9.5.8-1, 9.5.9-1	9.5.4, 9.5.5, 9.5.6, 9.5.7, 9.5.8, 9.5.9	Diesel generator auxiliary support systems
9-~~12~~11	9.2.5-1	9.2.5.1.3	Protected area perimeter abutting or crossing a body of water
9-~~13~~12	9.5.1.2.1.2-1	9.5.1.2	Procedures and training for using transfer switches
10-1	10.2-1	10.2.1	Turbine valve closing time
10-2	10.3-1	10.3.2.2	Steam hammer prevention
10-3	10.4.4-1	10.4.4.2.4.1	Pressure drops between the steam generator nozzles and each system valve
10-4	10.4.7-2, 10.4.9-2	10.4.7.2.5, 10.4.9.1.2	Avoidance of water hammer in the condensate, feedwater, and emergency feedwater systems
10-5	10.4.9-3	10.4.9.3, 10.4.9.5.2	Steam binding in the emergency feedwater pump
11-1	11.1-1, 11.5-1	11.1, 11.5.1.1	Conformance with Appendix B to 10 CFR 20, Appendix I to 10 CFR 50, ANSI N13.1, R.G. 1.21 and R.G. 4.15
11-2	11.4-1	11.4.1.1, 11.4.2.3.1	Site-specific solid waste management system operating procedures
11-3	11.5-2	11.5.1.4	Procedures in accordance with Position C of R.G. 4.15
11-4	11.2.1-1	11.2.2.3, 11.2.5, 11.2.6.1, 11.3.1.1, 11.5.1.2.3.1	Setpoints for radiation monitors; Offsite dose calculations.
11-5	11.5.1-1	11.5.1.1, 11.5.2.2, 11.5.2.4, 11.5.2.6	Operation and maintenance manual for monitoring and sampling liquid and gaseous process and effluent streams
12-1	12.1.1-1	12.1.1.2, 12.1.3	Operational ALARA policy

Table 1.10-1 COL License Information (Cont'd.)

COL No.	FSER No.	Section	Subject
17-1	17.1-1, 17.2-1	11.2.1.2, 11.3.1.2, 11.4.1.2, 17.1, 17.2	Construction and Operation QA (including TMI I.F.2, II.J.3.1)
17-2	17.3.1-1, 17.3.5-1	17.3.1, 17.3.5, 17.3.7, 17.3.13	D-RAP completion
17-3	17.3.9-1, 17.7	17.3.1, 17.3.7, 17.3.9, 17.3.10, 17.3.13, Tbl 19.15-1	Operations reliability assurance process implementation
18-1		18.9.3.2	Validation of operating ensemble
18-2	18.6.1.3.4-1	13.2	Operator training on "Plant Safety Parameter Display Console"
19-1	19.1, 19.1.2.2.2-1	19.15.3.1	Vulnerability of the SSWS intake structure to tornado-generated debris
19-2	19.1.2.2.3-1	19.7.5.3	Elements of the plant affecting the performance of systems in seismic events
19-3	19.1.2.4-4, 19.5, 19.6, 19.7	Tbl 19.15-1	Details of the layout of the critical components for fire and flood, interaction of internal flood sources, and effects of fire suppression systems on other systems
19-4	19.8	19.7.5.3 Table 19.15-1	Development of detailed seismic walkdown procedures to verify as-built SSC HCLPFs
19-5	19-10, 19-11	6.5.5, Tbl 19.15-1	Calculation of specific flow rate and consideration of shielding requirements for local operator actions for the emergency containment spray backup system
19-6	19.1.2.2.6-1, 19.1.4-1, 19.1.2.4-1, 19.1.2.4-2, 19.1.2.4-3, 19.12	19.7.5.3, 19.15, Tbl 19.15-1, 20.2.105	Update of PRA to include final design detail and site-specific information including examination of all external event hazards and analysis using site-specific spectra
19-7	19.14	19.15.6 Table 19.15-1	List of risk significant SSCs for D-RAP and operations reliability assurance process
19-8	19.15, 19-16, 19.19	19.15.6, Tbl 19.15-1	Consideration of risk important operator actions in developing procedures, training and human reliability related programs, and systems to address in severe accident management and aligning the alternate AC source (AAC) procedures

Table 2.0-1 Envelope of Plant Site Design Parameters (Cont'd.)

Aircraft Hazards	
Plant to airport distance, and	5mi. < D < 10mi. with annual operation less than $500D^2$ flights or D > 10mi. with an annual operation less than $1000D^2$ flights (D = distance in miles)
Plant to edge of militar training routes, and	D > 5mi. with an annual operation less than 1000 flights (D = distance in miles)
Plant to edge of Federal airway, holding pattern, or airport approach pattern.	D > 2mi. (D = distance in miles)
Meteorology	
Short-term dilution factor	X/Q $1.0x10^{-3}$; EAB = 0.5 mile
Long-term dilution factor	X/Q $2.2x10^{-5}$; LPZ = 2.0 miles

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Notes:

[1] Probable maximum flood level (PMF), as defined in ANSI/ANS-2.8, "Determining Design Basis Flooding at Power Reactor Sites."

[2] Maximum value for 1 hour 1 sq. mile PMP with ratio of 5 minutes to 1 hour PMP of 0.32, as found in National Weather Service Publication HMR No. 52.

[3] Maximum normal power and normal shutdown temperature of the Station Service Water System Intake based on one percent exceedance meteorologic conditions. See Section 9.2.5.1.3 for Ultimate Heat Sink temperature interface requirement for a design basis accident concurrent with a loss-of-offsite power.

[4] 50-year recurrence interval; value to be utilized for design of non-safety-related structures only.

[5] 100-year recurrence interval; value to be utilized for design of safety-related structures only.

[6] 10,000,000-year tornado recurrence interval, with associated parameters based on the NRC's interim position on Regulatory Guide 1.76. Pressure effects associated with potential offsite explosions are assumed to be non-controlling for the design.

[7] Site profiles are given in Section 2.5. Profiles include consideration of variability of soil properties. The lower bound of best estimate of soil shear wave velocity defines the lower bound of dynamic Soil-Structure Interaction analysis of the superstructure.

[8] The control motions are defined in Section 2.5.

[9] Bearing capacity is defined at the foundation level of the Nuclear Island structure.

2.2 Nearby Industrial, Transportation and Military Facilities

Industrial, transportation and military hazards are discussed below. [[Site-specific information on industrial, transportation, and military hazards will be provided by the COL applicant referencing the System 80+ design.]][1]

2.2.1 Aircraft Hazards

A site is acceptable for the System 80+ without further review if the distances from the plant meet the following requirements:

- The plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2 flights, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2 flights.
- The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.
- The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or airport approach pattern.

If the above site proximity acceptance criteria are not met, or if sufficiently hazardous military activities are identified, a detailed review of the aircraft hazards must be performed to qualify a specific site for the System 80+ plant.

2.2.2 Transportation

Site-specific information will include hazards related to transportation.

The ultimate heat sink, which is not included in the System 80+ scope for design certification, provides the source of cooling water for all safety-related plant systems and components during all modes of operation. Interface requirements (Section 9.2.5.1.3) are identified to eliminate the potential impacts on plant operations from boat or barge accident events.

2.2.3 Other Industrial Hazards On and Offsite

Site-specific information will include onsite and offsite industrial hazards.

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1 COL information item; see DCD Introduction Section 3.2.

Chapter 3 Contents

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Chapter 3 Contents (Cont'd.)

Response:

Reactor coolant makeup during normal operation is provided by the Chemical and Volume Control System (CVCS). The design incorporates a high degree of functional reliability by provision of redundant components and an alternate path for charging. The charging pumps can be powered from either onsite or offsite power sources, including the alternate AC generator. The system is described in Section 9.3.4. The CVCS has the capability of replacing the flow loss to the containment due to leaks in small reactor coolant lines such as instrument and sample lines. These lines have 7/32 inch diameter by 1 inch long flow restricting devices to limit loss of RCS coolant due to postulated pipe breaks in CVCS piping.

The CVCS is not required to perform any safety related function, such as accident mitigation, or required to perform a safe shutdown. This does not, however, compromise the "defense in depth" provided by the system as the normal means of maintaining RCS inventory and primary water chemistry. In designing the CVCS as non-safety grade, the following safety functions are performed by dedicated safety systems. Boration and makeup for design basis events will be provided by the Safety Injection System. Pressure control will be provided by the Safety Depressurization System. The Safety Injection System and the Safety Depressurization System are described in further detail in Sections 6.3 and 6.7, respectively. The Chemical and Volume Control System is designed as a non safety related system. Because the CVCS is essential for the day to day operation of the plant, it has been provided with a high degree of reliability and redundancy and has been designed in accordance with accepted industry standards and quality assurance commensurate with its importance to plant operations. Design criteria, including ASME Code classification assignments, are in accordance with ANSI/ANS 51.1, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants". This requires that portions of the CVCS within the RCPB, and all portions which assure containment isolation will have a rigorous safety classification in accordance with these specific functional performance requirements.

3.1.30 Criterion 34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Response:

Residual heat removal capability is provided by the Shutdown Cooling System for reactor coolant temperatures less than 350°F. For temperatures greater than 350°F, this function is provided by the steam generators. The Emergency Feedwater (EFW) System provides a dedicated, independent, safety-related means of supplying secondary side, quality feedwater to the steam generator(s) for removal of heat and prevention of reactor core uncovery. The design incorporates sufficient redundancy, interconnections, leak detection, and isolation capability to ensure that the residual heat removal function can be accomplished, assuming a single active failure. Within appropriate design limits, either system will remove fission product decay heat at a rate such that SAFDLs and the design conditions of the reactor coolant pressure boundary will not be exceeded.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

Structures, systems, and components which are important to safety and designed to remain functional in the event of a Safe Shutdown Earthquake (SSE) are designated as Seismic Category I.

Seismic Category I structures, systems, and components are those necessary to ensure:

- The integrity of the reactor coolant pressure boundary;
- the capability to achieve safe shutdown of the reactor and keep it in a safe shutdown condition; and
- the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of 10 CFR 100 guidelines.

The selection of Category I structures, systems, and components is in accordance with the definition above and the guidance provided by Regulatory Guide 1.29. Individual components in Category I systems are classified by reference to the safety classes assigned in accordance with ANSI/ANS 51.1 (see Section 3.2.2). All components in Safety Classes 1, 2, and 3 are Seismic Category I.

Structures, systems and components which do not perform a nuclear safety related function and whose continued function is not required are classified Non-Nuclear Safety (NNS) (see Section 3.2.2). NNS structures, systems and components whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level or could result in an incapacitating injury to an occupant of the control room are designated as Seismic Category II and are designed and constructed so that the SSE will not cause such failure in a manner that would adversely affect a safety system.

Structures, systems, and equipment which have no enhanced seismic design requirements in addition to those imposed by building codes are designated Non-Seismic (NS).

The seismic category and safety and quality classification of structures, systems, and components within the System 80+ Standard Design are listed in Table 3.2-1. The safety class is also shown on the P&IDs (Chapters 5, 6, and 9). Seismic Category I includes all mechanical components within the safety class boundaries and extends to the first seismic restraint beyond the boundary. All fuel racks are also designated as Seismic Category I. Structures, systems, or components whose failure could reduce the performance of a safety function by a Seismic Category I component to an unacceptable safety level are designed to Seismic Category II requirements for structural integrity only or are separated to the extent required to eliminate that possibility. NNS structures, systems or components whose failure could cause flooding of safety-related structures, systems or components are designed to Seismic Category I requirements. This ensures that any structures, systems, or components that could potentially have a disabling interaction with Seismic Category I structures, systems, or components are either prevented from doing so or are designed to meet Seismic Category I or II structural integrity requirements, depending on the function of the component.

Structural integrity requirements may be demonstrated by dynamic or equivalent static analyses, testing, or a combination thereof. Analyses of Seismic Category II structures, systems, and components are in accordance with the seismic input and methodology criteria described in Sections 3.7.2 and 3.7.3.

Table 3.2-1 Classification of Structures, Systems, and Components

Component Identification	Safety Class	Seismic Category	Location[25]*	Quality Class [29]*
Reactor Coolant System				
Reactor Vessel	1	I	RC	1
Steam Generators (primary/secondary)	1/2 [1]*	I	RC	1
Pressurizer	1	I	RC	1
Reactor Coolant Pumps ~~[2] [3] [9]~~* [2,3,9]	1	I	RC	1
Piping within Reactor Coolant Pressure Boundary [5]	1/2 [4]	I	RC	1
Control Element Drive Mechanisms	[6]	[6]	RC	1
Core Support Structures and Internals Structures [7]	3	I	RC	1
Fuel Assemblies [8]	2	I	RC	1
Control Element Assemblies [8]	3	I	RC	1
Closure Head Lift Rig	NNS	II [10]	RC	2
Heated Junction Thermocouple Probe Assembly	1/3 [12]	I	RC	1
HJTC Pressure Housing	1	I	RC	1
ICI Cable Tray Support Frame	3	I	RC	1
ICI Holding Frame	NNS	NS	RC	3
ICI Guide Tubes	1	I	RC	1
ICI Guide Tube Supports	1	I	RC	1
ICI Seal Housing	1	I	RC	1
ICI Seal Table	1	I	RC	1
Piping [27]	1/2	I	RC	1
Valves [27]	1/2	I	RC	1
In-containment Water Storage System				
IRWST	3	I	RC	1
Holdup Volume Tank	3	I	RC	1
Pressure Relief Dampers	3	I	RC	1
Cavity Flooding System				
Piping	2	I	RC	1
Valves	2	I	RC	1
Safety Depressurization System				
Valves	1/2	I	RC	1
Piping	1/2/NNS	I/NS	RC	1/3
Spargers	2	I	RC	1
Safety Injection System				
Safety Injection Pumps	2	I	RB	1
Safety Injection Tanks	2	I	RC	1
Piping ~~[24] [27]~~ [24,27]	1/2	I	RB/RC	1
Valves [27]	1/2	I	RB/RC	1

* Refer to Notes at end of table.

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location [25]	Quality Class [29]
Shutdown Cooling System				
Shutdown Cooling Heat Exchangers	2/3 [1]	I	RB	1
Shutdown Cooling Pumps	2	I	RB	1
Shutdown Cooling Mini-Flow Heat Exchanger	2/3 [1]	I	RB	1
Piping [27]	1/2/3	I	RB/RC	1
Valves [27]	1/2/3	I	RB/RC	1
Containment Spray System				
Containment Spray Pumps	2	I	RB	1
Containment Spray Heat Exchangers	2/3 [1]	I	RB	1
Containment Spray Mini-Flow Heat Exchanger	2/3 [1]	I	RB	1
Spray Nozzles	2	I	RC	1
Piping [27]	2/3	I	RB/RC	1
Valves [27]	2	I	RB/RC	1
Chemical and Volume Control System (CVCS)				
Regenerative Heat Exchanger	2	I	RC	1
Letdown Heat Exchanger	2/NNS [1,34]	I	RC	1
Seal Injection Heat Exchanger	NNS [34]	I	NA	2
Purification Ion Exchangers	NNS [34]	I	NA	2
Deborating Ion Exchanger	NNS [34]	I	NA	2
Volume Control Tank	NNS [34]	I	NA	2
Chemical Addition Package	NNS	NS	NA	3
Boric Acid Batching Tank	NNS	NS	NA	3
Charging Pumps	NNS [34]	I	NA	2
Dedicated Seal Injection Pump	NNS [34]	I	NA	2
Dedicated Seal Injection Pump Suction Stabilizer/Pulsation Dampener	NNS [34]	I	NA	2
Boric Acid Makeup Pumps	NNS [34]	I	NA	2
Reactor Makeup Water Pumps	NNS	NS	NA	3
Boric Acid Concentrator	NNS	NS	NA	2
Pre-holdup Ion Exchanger	NNS [34]	I	NA	2
Charging Pump Mini-flow Heat Exchanger	NNS [34]	I	NA	2
Boric Acid Condensate Ion Exchanger	NNS	NS	NA	2
Reactor Drain Pumps	NNS [34]	I	NA	2
Holdup Pumps	NNS	NS	NA	3
Reactor Drain Tank	NNS	NS	RC	2
Holdup Tank	NNS	NS	YA	2
Equipment Drain Tank	NNS [34]	I	NA	2
Reactor Makeup Water Tank	NNS	NS	YA	2
Gas Stripper	NNS [34]	I	NA	2
Purification Filters	NNS [34]	I	NA	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
CVCS (Cont'd.)				
Reactor Drain Filter	NNS [34]	I	NA	2
Seal Injection Filters	NNS [34]	I	NA	2
Reactor Makeup Water Filter	NNS	NS	NA	2
Boric Acid Filter	NNS [34]	I	NA	2
Letdown Strainer	NNS [34]	I	NA	2
Pre-holdup Strainer	NNS [34]	I	NA	2
Boric Acid Condensate IX Strainer	NNS	NS	NA	3
Ion Exchanger Drain Header Strainer	NNS	NS	NA	3
Boric Acid Batching Strainer	NNS	NS	NA	3
Chemical Addition Strainer	NNS	NS	NA	3
Boric Acid Storage Tank [33]	NNS [34]	I	YA	2
Boric Acid Batching Eductor	NNS	NS	NA	2
Letdown Orifices	2	I	RC	1
Piping [27]	1/2/3/NNS [35]	I/NS	RC/NA/YA	1/2
Valves [27]	1/2/3/NNS [35]	I/NS	RC/NA/YA	1/2
Emergency Feedwater System				
Cavitating Venturi	2	I	RC	1
Motor-Driven Emergency Feedwater Pumps	3	I	RB	1
Steam-Driven Emergency Feedwater Pumps	3	I	RB	1
Emergency Feedwater Pump Turbines	3	I	RB	1
Emergency Feedwater Storage Tanks	3	I	NA	1
Piping [27]	2/3	I	NA/RB/RC	1
Valves [27]	2/3	I	NA/RB/RC	1
Fuel Handling System				
Refueling Machine	NNS	II	RC	2
Fuel Transfer System	NNS	II	RC/NA	2
1. Transfer Carriage	NNS	II	RC/NA	2
2. Upending Machine	NNS	II	RC/NA	2
3. Hydraulic Power Unit	NNS	II	RC/NA	2
Fuel Transfer Tube, Valve, Stand	NNS	II	RC/NA	2
CEA Change Platform	NNS	II	RC	2
Long and Short Fuel Handling Tools	NNS	NS	RC/NA	3
Upper Guide Structure Lifting Rig	NNS	II [11]	RC	2
Core Barrel Lifting Rig	NNS	II ~~[1]~~ [11]	RC	2
Spent Fuel Handling Machine	NNS	II	NA	2
New Fuel Elevator	NNS	II	NA	2
Underwater Television	NNS	NS	RC/NA	3
Refueling Pool Seal	NNS	NS	RC	2
In-Core Instrumentation and CEA Cutter	NNS	NS	RC	3
Extension Shaft Uncoupling Tool	NNS	NS	RC	3
Fuel Transfer Tube Quick Closure	2	I	RC	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Fuel Handling System (Cont'd.)				
CEA Handling Tools	NNS	NS	RC	3
ICI Insertion and Removal Tools	NNS	NS	RC	3
Spent Fuel Racks	3	I	NA	1
New Fuel Racks	3	I	NA	1
Condensate and Feedwater System				
Condensate Pumps	NNS	NS	TB	2
Feedwater Pumps	NNS	NS	TB	2
Feedwater Pump Controllers	NNS	NS	TB	2
Feedwater Booster Pumps	NNS	NS	TB	2
Startup Feedwater Pump	NNS	NS	TB	2
Low Pressure Feedwater Heaters	NNS	NS	TB	2
High Pressure Feedwater Heaters	NNS	NS	TB	2
Deaerator	NNS	NS	TB	2
Piping (13)	2/NNS	I/NS	TB/NA/RC/MS	1/2/3
Valves (13)	2/NNS	I/NS	TB/NA/RC/MS	1/2/3
Main Condenser System				
Main Condenser	NNS	NS	TB	2
Condensate Storage System				
Condensate Storage Tanks	NNS	NS	YA	2
Condensate Storage Tank Recycle Pumps	NNS	NS	SB	2
Piping	NNS	NS	YA/SB/TB	2/3
Valves	NNS	NS	YA/SB/TB	2/3
Condensate Cleanup System				
Piping	NNS	NS	TB	2/3
Polishers/Demineralizers	NNS	NS	TB	2
Resin Traps	NNS	NS	TB	2
Valves	NNS	NS	TB	2/3
Main Condenser Evacuation System				
Vacuum Pumps	NNS	NS	TB	2
Piping	NNS	NS	TB	2/3
Valves	NNS	NS	TB	2/3
Demineralized Water Makeup System (DWMS)				
Demineralizer Makeup Water Pumps	NNS	NS	SB	3
Demineralizers	NNS	NS	SB	3
Vacuum Degasifier	NNS	NS	SB	3
Demineralized Water Storage Tank	NNS	NS	YA	3
Vacuum Pumps	NNS	NS	SB	3
Demineralizer Recycle Pump	NNS	NS	SB	3
Vacuum Degasifier Transfer Pumps	NNS	NS	SB	3
Demineralized Water Transfer Pumps	NNS	NS	SB	3
Regenerant Waste Neutralization Tank	NNS	NS	SB	3

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Turbine Control System (Cont'd.)				
Piping	NNS	NS	TB	2
Valves	NNS	NS	TB	2
Turbine Generator Cooling System				
Hydrogen Coolers	NNS	NS	TB	2
Piping	NNS	NS	TB	2
Valves	NNS	NS	TB	2
Liquid Waste Management System				
Waste Collection Tanks	NNS	NS	RW	2
Waste Sample Tanks	NNS	NS	RW	2
Process Pumps	NNS	NS	RW	2
Process Demineralizers	NNS	NS	RW	2
Process Filters	NNS	NS	RW	2
Piping [27]	2/NNS	I/NS	TB/NA/RW RC/RB	1/2
Valves [27]	2/NNS	I/NS	TB/NA/RW RC/RB	1/2
Gaseous Waste Management System				
Piping [27]	2/NNS	I/NS	NA/RC	1/2
Gas Coolers/Condenser	NNS	NS	NA	2
Guard/Charcoal Beds	NNS	NS	NA	2
Valves [27]	2/NNS	I/NS	NA/RC	1/2
Solid Waste Management System				
Spent Resin Transfer Pumps	NNS	NS	NA/RW	2
Spent Resin Tanks	NNS	NS	NA/RW	2
HIC Fill/Dewatering Head	NNS	NS	RW	2
Resin Forwarding Pumps	NNS	NS	RW	2
Dry Solids Compactor	NNS	NS	RW	2
Piping	NNS	NS	NA/RW	2
Valves	NNS	NS	NA/RW	2
Heater Drain System				
Piping	NNS	NS	TB	2/3
Reheater Drain Tanks	NNS	NS	TB	2
Moisture Separator Drain Tanks	NNS	NS	TB	2
Heater Drain Tank	NNS	NS	TB	2
Heater Drain Pumps	NNS	NS	TB	2
Valves	NNS	NS	TB	2/3
Process and Effluent Radiation Monitoring System (PERMS)				
Gaseous Process and Effluent Monitors				
Unit Vent	NNS	NS	NA	2
Waste Gas	NNS	NS	RW	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
PERMS (Cont'd.)				
Unit Vent Post-Accident	NNS	NS	NA	2
Containment Purge Exhaust	NNS	NS	NA	2
Condenser Air Ejector	NNS	NS	TB	2
Liquid Process and Effluent Monitors				
Component Cooling Water	NNS	NS	NA	2
Liquid Waste Discharge	NNS	NS	RW	2
Plant Discharge Line	NNS	NS	RW	2
Station Service Water	NNS	NS	CX	2
Reactor Coolant Gross Activity	NNS	NS	NA	2
Turbine Building Drains	NNS	NS	TB	2
Steam Generator Blowdown	NNS	NS	TB	2
Airborne Radiation Monitors				
Containment Atmosphere	3	I	NA	1
Nuclear Annex	NNS	NS	NA	2
Radwaste Building	NNS	NS	RW	2
Fuel Building	NNS	NS	NA	2
Ventilation Systems Multisampler	NNS	NS	NA	2
Control Room Intake (A&B)	3	I	NA	1
Reactor Building Annulus	NNS	NS	NA	2
Subsphere Ventilation	NNS	NS	NA	2
Area Radiation Monitors	NNS	NS	RC/NA/RW	2
Special Purpose Area Monitors				
Main Steam Line	NNS	NS	NA	2
Purification Filter	NNS	NS	NA	2
Containment Area High Radiation	3	I	RC	1
Primary Coolant	3	I	RC	1
Containment Isolation System				
Piping	2	I	RC/RB	1
Valves	2	I	RC/RB	1
Component Cooling Water System [14]				
Piping [27]	2/3/NNS	I/NS	CX/YA/NA RB/RC	1/2/3
Heat Exchangers	3	I	CX	1
Pumps	3	I	NA	1
Surge Tanks	3	I	NA	1
Sump Pumps	NNS	NS	NA	3
Chemical Addition Tank	NNS	NS	NA	3
Heat Exchanger Building Sump Pumps	NNS	NS	CX	3
Valves	2/3/NNS	I/NS	CX/YA/NA RB/RC	1/2/3

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Spent Fuel Pool Cooling System				
Pumps	3	I	NA	1
Exchangers	3	I	NA	1
Piping	3/NNS	I/NS	NA	1/3
Valves	3/NNS	I/NS	NA	1/3
Pool Purification System				
Pumps	NNS	NS	NA	2
Strainers	NNS	NS	NA	3
Demineralizers	NNS	NS	NA	2
Filters	NNS	NS	NA	2
Skimmer	NNS	NS	NA	3
Piping [27]	2/3/NNS	I/NS	NA/RC	1/2
Valves [27]	2/3/NNS	I/NS	NA/RC	1/2
Primary Sampling System				
Pump	NNS	NS	NA	2
Heat Exchangers	NNS	NS	NA	2
Sample Vessels	NNS	NS	NA	2
Piping [27]	2/3/NNS	I/NS	NA/RC	1/2
Valves [27]	2/3/NNS	I/NS	NA/RC	1/2
Sink	NNS	NS	NA	3
Boronometer	NNS	NS	NA	2
Process Radiation Monitor	NNS	NS	NA	2
Secondary Chemistry Control Sampling System				
Heat Exchangers	NNS	NS	NA	2/3
Strainers	NNS	NS	NA	2/3
Monitors	NNS	NS	NA	2/3
Piping [27]	2/NNS	I/NS	NA/RC	1/3
Valves [27]	2/NNS	I/NS	NA/RC	1/3
Station Service Water System				
Piping	3/NNS	I/NS	SP/CX	1/3
Pumps	3	I	SP	1
Strainers	3	I	SP	1
Sump Pumps	NNS	NS	SP	3
Traveling Screens	3	I	YA	1
Valves	3/NNS	I/NS	SP/CX	1/3
Turbine Building Service Water System				
Piping	NNS	NS	YA	2/3
Valves	NNS	NS	YA	2/3
Pumps	NNS	NS	YA	2
Strainers	NNS	NS	YA	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Turbine Building Cooling Water System				
Piping	NNS	NS	TB/YA	2/3
Valves	NNS	NS	TB/YA	2/3
Heat Exchangers	NNS	NS	YA	2/3
Pumps	NNS	NS	TB	2/3
Surge Tank	NNS	NS	TB	2/3
Chemical Addition Tank	NNS	NS	TB	3
Essential Chilled Water System				
Refrigeration Units	3	I	NA	1
Pumps	3	I	NA	1
Compression Tanks	3	I	NA	1
Chemical Addition Tanks	NNS	NS	NA	3
Essential/Normal Heat Exchangers	3/NNS [1]	I	NA	1/2
Piping [27]	2/3/NNS	I/NS	NA/RC/RB	1/2/3
Valves [27]	2/3/NNS	I/NS	NA/RC/RB	1/2/3
Strainers	3/NNS	I/NS	NA	1/3
Normal Chilled Water System [15]				
Refrigeration Units	NNS	NS	NA	2
Pumps	NNS	NS	NA	2
Compression Tanks	NNS	NS	NA	3
Air Separators	NNS	NS	NA	3
Chemical Addition Tanks	NNS	NS	NA	3
Piping [27]	2/NNS	I/NS	NA/RC	1/3
Valves [27]	2/NNS	I/NS	NA/RC	1/3
Strainers	NNS	NS	NA	3
Condenser Circulating Water System				
Pumps	NNS	NS	YA	2
Cooling Towers (mechanical portion)	NNS	NS	YA	2
Piping	NNS	NS	YA/TB	2/3
Valves	NNS	NS	YA/TB	2/3
Strainers	NNS	NS	YA/TB	2
Traveling Screens	NNS	NS	YA	2
Instrument Air System				
Air Compressors	NNS	NS	NA	2
Piping [27]	2/NNS	I/NS	All	1/3
Valves [27]	2/NNS	I/NS	All	1/3
Air Receivers	NNS	NS	NA	3
Desiccant Air Dryers/Filters	NNS	NS	NA	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Station Air System				
Air Compressors	NNS	NS	SB	3
Air Dryers/Filters	NNS	NS	SB	3
Piping [27]	2/NNS	I/NS	All	1/3
Air Receivers	NNS	NS	SB	3
Valves [27]	2/NNS	I/NS	All	1/3
Breathing Air System				
Air Compressors	NNS	NS	SB	3
Piping [27]	2/NNS	I/NS	All	1/3
Valves [27]	2/NNS	I/NS	All	1/3
Air Receivers	NNS	NS	SB	3
Air Dryer/Filters	NNS	NS	SB	3
Compressed Gas Systems				
Piping ~~[26] [27]~~ [26,27]	2/NNS	I/NS	All	1/3
High Pressure Gas Cylinders	NNS	NS	YA	3
Pressure Regulators	NNS	NS	YA	3
Leak Detection Systems	NNS	NS	All	3
Liquid Nitrogen Evaporators	NNS	NS	YA	3
Valves [27]	2/NNS	I/NS	All	1/3
Fire Protection System				
Jockey Pump	NNS	NS	FP	2
Backup Storage Tank	NNS	I	NA	1
Fire Pumps	NNS	NS	FP	2
Backup Fire Pump	NNS	I	NA	1
Storage Tanks	NNS	NS	FB	2
Water Spray Systems (Deluge and Sprinkler) Piping, Valves [16] [27]	2/NNS	I/II/NS	TB/NA/RC/RB/ DG/SB	1/2
Hose Systems/Standpipes [16] [27]	2/NNS	I/NS	All	1/2
Portable Fire Extinguishers [16]	NNS	NS	All	2
Exterior Distribution System				
Piping	NNS	NS	YA	2
Valves	NNS	NS	YA	2
Strainers	NNS	NS	YA	2
Alternate AC Source/Combustion Turbine-Generator	NNS	NS	YA	2
DG Engine Fuel Oil System [17]				
Fuel Oil Storage Tanks	3	I	DF	1
Recirculation Pumps	NNS	NS	DF	3
Booster Pumps	3	I	DG	1
Fuel Oil Day Tanks	3	I	DG	1

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
DG Engine Fuel Oil System [17]				
Fuel Oil Transfer Pumps	3	I	DG	1
Strainers	3/NNS	I/NS	DG/YA	1/3
Filters	3/NNS	I/NS	DG	1/3
Piping	3/NNS	I/NS	DG/DF/YA	1/3
Valves	3/NNS	I/NS	DG/DF	1/3
DG Engine Cooling Water System				
Circulation Pumps	3	I	DG	1
Keep Warm Pumps	3	I	DG	1
Jacket Water Coolers	3	I	DG	1
Jacket Water Standpipes	3	I	DG	1
Chemical Pot Feeders	3	I	DG	1
Piping	3	I	DG	1
Valves	3	I	DG	1
DG Engine Starting Air System [18]				
Compressors	NNS	NS	DG	2
Aftercoolers	NNS	NS	DG	3
Moisture Separators	NNS	NS	DG	3
Filter/Dryer Units	NNS	NS	DG	3
Air Receivers	3	I	DG	1
Strainers	3/NNS	I/NS	DG	1/3
Traps	NNS	NS	DG	3
Filters	3/NNS	I/NS	DG	1/3
Piping	3/NNS	I/NS	DG	1/3
Valves	3/NNS	I/NS	DG	1/3
DG Engine Lube Oil System [19]				
Lube Oil Sump Tanks	3	I	DG	1
Lube Oil Coolers	3	I	DG	1
Oil Transfer Pumps	NNS	NS	DG/YA	3
Prelube Oil Pumps	3	I	DG	1
Clean and Used Lube Oil Storage Tanks	NNS	NS	YA	3
Filters	3	I	DG	1
Strainers	3/NNS	I/NS	DG	1/3
Piping	3/NNS	I/NS	DG/YA	1/3
Valves	3/NNS	I/NS	DG/YA	1/3
DG Engine Air Intake and Exhaust System				
Turbochargers	3	I	DG	1
Aftercoolers	3	I	DG	1
Silencers and Air Filters	3	I	DG	1
Piping	3	I	DG	1

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Equipment and Floor Drainage System				
Reactor Building Subsphere Sump Pumps	3	I	RB	1
Other Sump Pumps	NNS	NS		3
Piping [27]	2/3/NNS	I/NS	All	1/3
Valves [27]	2/3/NNS	I/NS	All	1/3
Diesel Generator Building Sump Pump System				
Sump Pumps	3	I	DG	1
Piping	3/NNS	I/NS	DG/NA/RW	1/3
Valves	3/NNS	I/NS	DG/NA/RW	1/3
Control Complex Ventilation System				
Main Control Room Air ~~Handling~~ Conditioning System				
Air ~~Handling~~ Conditioning Units w/Filters	3	I	NA	1
Fans, Ductwork [31]	3/NNS	I/II	NA	1/2
Water-cooling Coils	3	I	NA	1
Heating Coils	3	I	NA	1
Dampers	3	I	NA	1
Technical Support Center Air ~~Handling~~ Conditioning System				
Air ~~Handling~~ Conditioning Units w/Filters	NNS	II	NA	2
Fans, Ductwork	NNS	II	NA	2
Dampers	NNS	II	NA	2
Computer Room Air ~~Handling~~ Conditioning System				
Air ~~Handling~~ Conditioning Units w/Filters	NNS	II	NA	2
Fans, Ductwork	NNS	II	NA	2
Dampers	NNS	II	NA	2
Essential Electrical Rooms and Vital Instrumentation and Equipment Rooms Conditioning (inc. Battery Rooms)				
Air ~~Handling~~ Conditioning Units w/Filters	3	I	NA	1
Fans, Ductwork	3	I	NA	1
Dampers	3	I	NA	1
Balance of Building Air ~~Handling~~ Conditioning System				
Filters	NNS	NS	NA	3
Water Cooling Coils	NNS	NS	NA	3
Fans, Ductwork	NNS	NS	NA	3
Dampers	NNS	NS	NA	3

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Fuel Building Ventilation System				
Cooling Coil	NNS	NS	NA	3
Heating Coil, Supply	NNS	NS	NA	3
Air Handling Unit w/Filter	NNS	II	NA	2
Ductwork, Supply	NNS	II	NA	2
Exhaust System Filter Train	3	I	NA	1
Exhaust System Fans	3	I	NA	1
Exhaust System Dampers	3	I	NA	1
Ductwork, Exhaust	3	I	NA	1
Dampers, Supply	NNS	II	NA	2
Nuclear Annex Ventilation System [20]				
Supply Air Handling Units	NNS	II	NA	2
Ductwork, Supply	NNS	II	NA	2
Cooling Coils	NNS	II	NA	3
Particulate Exhaust Filter Units	NNS	II	NA	2
Fans, Ductwork	NNS	II	NA	2
Dampers	NNS	II	NA	2
Radwaste Building Ventilation System				
Supply Air Handling Units	NNS	NS	RW	2
Cooling Coils	NNS	NS	RW	3
Exhaust Filter Units	NNS	NS	RW	2
Fans	NNS	NS	RW	2
Ductwork	NNS	NS	RW/NA	2
Dampers	NNS	NS	RW	2
Reactor Building Subsphere Ventilation System				
Individual Cooling Units	3/NNS	I/II	RB	1/2
Exhaust Fans	3	I	NA	1
Cooling Coils and Heating Coils	3	I	NA	1
Exhaust System Filter Train	3	I	NA	1
Ductwork, Exhaust	3	I	NA/RB	1
Supply Fans	NNS	II	NA	2
Supply Air Handling Units	NNS	II	NA	2
Ductwork, Supply	NNS	II	NA/RB	2
Dampers, Exhaust	3	I	NA	1
Dampers, Supply	NNS	II	NA	2
Diesel Building Ventilation System				
Space Heater	3	I	DG	1
Emergency/Normal Fans	3/NNS	I/II	DG	1/2
Ductwork	3/NNS	I/II	DG	1/2
Dampers	3/NNS	I/II	DG	1/2
Filter, Normal Supply	NNS	NS	DG	2

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Annulus Ventilation System				
Filter Trains	3	I	NA	1
Fans	3	I	NA	1
Dampers	3	I	NA	1
Ductwork	3	I	NA/RB	1
Containment Purge Ventilation System				
Water Cooling Coil	NNS	NS	NA	3
Heating Coil	NNS	NS	NA	3
Supply and Exhaust Fans	NNS	II	NA	2
Valves [27]	2/NNS	I/II	NA/RC	1/2
Filter Trains	NNS [28]	II	NA	2
Ductwork [27] [30]	2/NNS	I/II	NA/RC	1/2
Containment Cooling and Ventilation System				
Containment Cooling Subsystem	NNS	II	RC	2
Control Element Drive Mechanism Cooling Subsystem	NNS	II	RC	2
Containment Air Cleanup System	NNS	II	RC	2
Cavity Cooling Subsystem	NNS	II	RC	2
Ductwork	NNS	II	RC	2
Dampers	NNS	II	RC	2
Turbine Building Ventilation System				
Fans	NNS	NS	TB	3
Dampers	NNS	NS	TB	3
Exhausters	NNS	NS	TB	3
Ductwork	NNS	NS	TB	3
Station Service Water Pump Structure Ventilation System				
Fans	3	I	SP	1
Dampers	3	I	SP	1
Ductwork	3	I	SP	1
Component Cooling Water Heat Exchanger Structure(s) Ventilation Systems				
Fans	NNS	II	CX	3
Dampers	NNS	II	CX	3
Space Heaters	NNS	II	CX	3
Ductwork	NNS	II	CX	3

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Main Steam Supply System Piping [21]				
Piping [21]				
Steam Generator to MSIV's	2	I	RC/MS	1
Other	NNS	NS	MS/NA/TB	3
Valves [21] Main Steam Supply System				
Safety Valves	2	I	MS	1
MSIV's, MSIV Bypass Valves	2	I	MS	1
Atmospheric Dump Valves	2	I	MS	1
Valves	2/NNS	I/NS	NA/MS/TB	1/3
Containment Hydrogen Recombiner System				
Piping [27]	2	I	NA/RC	1
Hydrogen Recombiners	2	I	NA	1
Hydrogen Analyzers	2	I	NA	1
Hydrogen Recombiner Control Panel	3	I	NA	1
Valves [27]	2	I	NA/RC	1
Steam Generator Blowdown System [22]				
Piping [27]	2/NNS	I/NS	RC/TB/MS	1/2
Flash Tank	NNS	NS	TB	2
Heat Exchanger	NNS	NS	TB	2
Filter	NNS	NS	TB	2
Demineralizers	NNS	NS	TB	2
Valves [27]	2/NSS	I/NS	RC/TB/MS	1/2
Steam Generator Wet Layup Recirculation System [22]				
Piping [27]	2/NNS	I/NS	RC/TB/MS	1/3
Valves [27]	2/NSS	I/NS	RC/TB/MS	1/3
Hydrogen Mitigation System				
Hydrogen Igniters	NNS	I	RC	2
Potable and Sanitary Water Systems	NNS	NS	YA	3
Instrumentation and Control Systems				
Plant Protection System (PPS)				
The PPS includes the electrical and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating the signals associated with the two protective functions defined below:				

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Instrumentation and Control Systems (Cont'd.)				
Reactor Protective System (RPS) That portion of the PPS which generates signals that actuate reactor trip	3	I	NA/RC	1
Engineered Safety Features Actuation System (ESF) That portion of the PPS which generates signals that actuate engineered safety features	3	I	NA/RC	1
Safe Shutdown Systems The safe shutdow systems include those systems required to secure and maintain the reactor in a safe shutdown condition	3	I	DG/NA/CX SP/MS/ RB/RC	1
All other systems required for safety	3	I	NA/DG/CX SP/MS/ RB/RC	1
Equipment required to comply with	NNS	2	NA/RC	
10CFR50.62	NNS	NS	All	2
Equipment specified in Section	NNS	NS	All	2/3
3.3.1.4 of ANSI/ANS-51.1	3	I	NA	2/3
Control systems not required for safety	NNS	II	NA	1
Control Room Panels (safety-related)				1
Control Room Panels (other)				1
Instrument valves and piping downstream of Safety Class 2 or 3 root valves (For safety-related instruments)				
Piping, tubing, and fittings	2/3	I	All	1
Instrument valves	NNS	NS	All	3
Electric Systems				
Class 1E AC Equipment (includes associated transformers, protective relays, instrumentation and control devices				
4.16 kV Buses	3	I	NA	1
480V Load Centers	3	I	NA	1
480V Motor Control Centers	3	I	NA/CX/DG/SP	
Class 1E DC Equipment				
125V Station Batteries and Racks	3	I	NA	1

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Electric Systems (Cont'd.)				
Battery Chargers	3	I	NA	1
125V Switchgear and Distribution Panels	3	I	NA	1
120V Vital AC System Equipment				
Inverters	3	I	NA	1
120V Distribution Panels	3	I	NA	1
Electrical Cables for Class 1E Systems				
125V DC Cables (including cable splices, connectors, and terminal blocks)	3	I	NA	1
5 kV Power Cables (including cable splices, connectors, and terminal blocks)	3	I	NA/DG/CX/ SP	1
600V Power Cables (including cable splices, connectors, and terminal blocks)	3	I	NA/DG/CX/ SP/MS/ RB/RC	
Control and Instrumentation Cables (including cable splices, connectors, and terminal blocks)	3	I	DG/CX/NA SP/MS/RB	1
Conduit and cable trays and their supports containing Class 1E cables and those whose failure during a seismic event may damage other safety-related items	3	I	DG/CX/NA/ SP/MS/RB RC	1
Miscellaneous Class 1E Electrical Systems				
Containment building electrical penetration assemblies	3	I	RC	1
Non-Class 1E Electrical Systems	NNS	II/NS	All	2/3
Instrumention and Display Systems not required for safety [32]	NNS	NS	All	2/3
~~Structures~~				
Reactor Building structure				
Containment Shield Building	3	I	RB	1
Steel Containment Vessel	2	I	RB	1
Internal Structure	3	I	RC	1
Equipment Hatch	2	I	RC	1
Personnel Airlocks	2	I	RC	1
Subsphere (Including Containment Support Dish)	3	I	RB	1

Table 3.2-1 Classification of Structures, Systems, and Components (Cont'd.)

Component Identification	Safety Class	Seismic Category	Location	Quality Class
Nuclear Annex Structure				
Control Area	3	I	NA	1
EFW Tank/Main Steam Valve House Area	3	I	NA	1
Emergency Diesel Generator Areas	3	I	NA	1
CVCS/Maintenance Area	3	I	NA	1
Fuel Handling Area	3	II	NA	1
(Other Structures) Bold				
Unit Vent	NNS	II	NA/RB	2
Turbine Building	NNS	II	TB	2
Radwaste Building [28]	NNS	I	RW	2
Station Service Water Pump/Intake Structure	3	I	SP	1
Component Cooling Water Heat Exchanger Structures and Pipe Tunnels	3	I	CX/YD	1
Diesel Fuel Storage Structure	3	I	DF	1
Station Services Building/Auxiliary Boiler Structure	NNS	NS	SB	3
Administration Building	NNS	NS	ADB	3
Warehouse	NNS	NS	WH	3
Fire Pump House	NNS	NS	FP	3
(Dikes) Bold				
Dike (Holdup, Boric Acid Storage and Reactor Makeup Water Tanks) [28]	NNS	II	YA	2
Dike (Condensate Storage Tank) [28]	NNS	II	YA	2
Alternate AC Source/Combustion Turbine-Generator Structure and Fuel Tank	NNS	NS	YA	2
Cranes				
Polar Crane	NNS	II	RC	2
Cask Handling Hoist	NNS	II	NA	2
New Fuel Handling Hoist	NNS	II	NA	2
(Component Supports [23]) Bold	1/2/3/NNS	I/NS	All	1/2/3

Contents (Cont'd.)

Tables

- Design Condition

The T_a load is not included in the combination because thermal loads are considered as secondary stresses. The ASME Code does not require an analysis of secondary stresses for the Design condition.

The pipe reaction R_a and R_o loads on the steel containment are eliminated as described in the System 80+ design by the use of bellows and/or guard pipes at the containment penetrations where pipe reactions would exist.

The reduced load combination is:

$D + L + P_a$

- Service Conditions

Service Level A:

Pipe reactions R_a and R_o are eliminated as described in the Design combination.

The stresses resulting from the operating temperature and pressure loads, T_o and P_o, are enveloped by the accident temperature and pressure loads and therefore are not analyzed separately.

The primary membrane stress evaluation for Service Level A is the same as the Design Condition.

When evaluating secondary stress effects, the reduced load combination is:

$D + L + T_a + P_a$

Service Level C:

Pipe reactions R_a and R_o are eliminated as described in the Design combination.

The stresses resulting from the operating pressure loads, P_o, are enveloped by the accident pressure loads and therefore are not analyzed separately.

The T_a and T_o loads are not included in the combination because thermal loads are considered as secondary stresses as described in the Design combination. The ASME code does not require an analysis of secondary stresses for Service Level C.

The reduced Service Level C loads are the same as the reduced Service Level D loads. The ASME Service Level D allowable stresses are lower than the Service Level C allowable stresses; therefore, the analysis is performed for the reduced Service Level D loading combination and compared with the lower allowable stresses of Service Level D.

10.4.2.2 Design Requirements

The Crane Wall provides supports for the polar crane and protects the steel containment vessel from internal missiles. In addition to providing biological shielding for the coolant loop and equipment, the Crane Wall also provides structural support for pipe supports/restraints and platforms at various levels.

The design shall address the vertical alignment of the Crane Wall with the corresponding structure below the Containment Vessel and provides special construction tolerances, as necessary, to ensure potential misalignment is appropriately considered. The design also considers potential differential basemat settlement and the effect on the Crane Wall alignment.

10.4.2.3 Design Loads (Reference Section 3.8.3.3)

Crane Wall.

Refer to Table 3.8A-1 for additional loads that are applicable to the ~~Refueling Cavity~~.

10.4.3 Refueling Cavity

10.4.3.1 Description

The Refueling Cavity is the reinforced concrete enclosure that provides a pool filled with borated water above the reactor vessel to facilitate the fuel handling operation without exceeding the acceptable level of radiation inside the Containment Vessel. The Refueling Cavity has the following sub-compartments.

- Storage Area for Upper Guide Structure
- Storage area for Core Support Barrel
- Refueling Canal

The Reactor Vessel flange is sealed to the bottom of the Refueling Cavity to prevent leakage of refueling water into the reactor cavity. The Fuel Transfer Tube connects the Refueling Cavity to the Spent Fuel Pool. The shield walls that form the Refueling Cavity are a minimum of six feet thick.

10.4.3.2 Design Requirements

The Refueling Cavity walls and floor shall be covered with stainless steel plate for leak tightness and for contamination and corrosion control.

10.4.3.3 Design Loads (Reference Section 3.8.3.3)

Refer to Table 3.8A-1 for additional design loads that are applicable to the Refueling Cavity.

10.4.4 Operating Floor

10.4.4.1 Description

The Operating Floor at El. 146'-0" provides access for operating personnel functions and provides biological shielding. Inside the Crane Wall, the operating floor is a reinforced concrete slab with a covered hatch that is aligned with hatches in the two lower floors. Outside the Crane Wall, the Operating Floor consists of steel grating.

Table 3.9-10 Loading Combinations for ASME Section III Class 1 Piping

Service Level	Loading Combination
Design	Design Pressure, Weight, Other Sustained Mechanical Loads
Level A	Level A Transients, Weight, Operating Pressure, Thermal Expansion, Anchor Movements, Other Mechanical Loads, Dynamic Fluid Loads
Level B	Level B Transients, Weight, Coincident Pressure, Thermal Expansion, Anchor Movements, Safe [[*Shutdown Earthquake,*[1,3]]][4] Other Mechanical Loads, Dynamic Fluid Loads
Level C	Maximum Pressure, Other Mechanical Loads, Weight, Dynamic Fluid Loads
Level D	Maximum Pressure, Other Mechanical Loads, Weight, Safe Shutdown Earthquake, Pipe Break Loads, Dynamic Fluid Loads, [[*SSE SAMS (Full Range) Thermal TAMS,*[2] *Thermal Expansion*[2]]][4]

Notes: The dynamic loads are combined by the square root of the sum of the squares.

[[*1. Alternatively, a lower level of SSE motion may be used in accordance with Section 3.7.3.2.*

2. Loading combination for Eq. 12a of Reference 50.

3. Loading combination for Eq. 10; primary plus secondary stress producing load.]][4]

Table 3.9-11 Loading Combinations for ASME Section III Classes 2 and 3 Piping

Service Level	Loading Combination
Design	Design Pressure, Weight
Level A & B	Operating Pressure, Weight, Other Occasional Loads (DFL, Wind) Thermal Expansion, Anchor Movements
Level C	Maximum Pressure, Weight, Other Occasional Loads (DFL, Tornado)
Level D	Maximum Pressure, Weight, DFL, Safe Shutdown Earthquake, Pipe Break, [[*Anchor Movements*[1], *Thermal Expansion*[1]]][4]

Notes: Dynamic fluid loads (DFL) are occasional loads such as safety/relief valve thrust, steam hammer, water hammer, or loads associated with plant upset or faulted condition as applicable.

[[*1. Loading Combination for Eq. 10b of Reference 50.*]] [4]

[4] NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

1.5 Analysis Tecchniques

1.5.1 Model Boundaries

Piping models ideally run from anchor to anchor (equipment nozzle, or penetration). Where this is not feasible, the piping is separated by decoupling, overlapping, isolation, or in-line anchors as described in the following subsections to form more manageable models for analysis. Where the piping cannot be separated to form smaller analysis models by these methods, the use of an intermediate anchor is considered in order to separate models, subject to the considerations of Section 1.5.5 of this appendix.

1.5.2 Decoupling

1.5.2.1 General

Small branch lines are allowed to be decoupled from larger run piping regardless of seismic classification. In some instances, decoupling is also applied for in-line pipe size changes (such as at a reducer or reducing insert). In the description in Section 1.5.2.2, the smaller line is defined as the "branch" and the larger line is defined as the "run". To meet decoupling criteria, piping meets the size or moment of inertia ratios as detailed in the following paragraphs. For decoupling criteria to be meaningful, the branch line must be designed flexible enough to absorb the anchor motions of the run pipe. Therefore, the branch line flexibility is maintained by avoiding placement of branch line supports close to the run pipe.

1.5.2.2 Branch Decoupling Criteria

[[*Branch lines meeting the following criteria may be decoupled from the main run:*

$D_b/D_r \leq 0.33$, *or*

$I_b/I_r \leq 0.04$,]][1]

where:

D_b = Branch nominal pipe size

D_r = Run nominal pipe size

WHOLE ¶ IS [[ ]] I_b = Branch moment of inertia

I_r = Run moment of inertia

[[*An appropriate stress intensity factor (SIF) is included on the branch and main run lines at the point where the piping is decoupled. Mass effects of the branch line are considered in the analysis of the run line. The branch point is considered as an anchor in the analysis of the branch pipe.*]][1] Thermal and seismic anchor movement analyses of the decoupled branch lines are performed with the thermal, seismic inertial, seismic anchor movement (SAM), or pipe break movements of the larger pipe header applied Italics

[1] NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

italics e

as anchor displacements and/or rotations to the smaller branch line wherever these movements are significant. [[*The inertia effects of the run pipe on branch pipe are considered, where significant.*]][1]

Piping is also decoupled at flexible hose wherever each interfacing analysis considers the flexible hose weight and significant stiffness, and wherever the flexible hose qualifies for the net end displacements of the interfacing analysis problems. Analysis results of the interfacing problems are not combined. The flexible hose is not allowed to experience loads beyond those recommended by the manufacturer.

Also refer to Section 3.7.2.3.3 for general decoupling criteria.

1.5.2.3 Seismic to Non-seismic Decoupling Criteria

Two methods for designing the region of a seismic/non-seismic piping interface are as follows:

1. Use of structural anchors for isolation

   Structural isolation anchors provide an effective means of protecting seismic piping from the seismic response of non-seismically designed piping. Anchors are designed assuming that a plastic hinge forms at the interface with non-seismic piping.

2. Use of isolation restraints

   Piping restraints are utilized to isolate the seismic response of non-seismically designed piping from seismically designed piping. Isolation restraints are designed as follows:

   - Two restraints located in each of the three orthogonal directions are used to isolate the seismically caused pipe moments, and forces from the non-seismic piping to the seismic piping.

   - An isolation restraint is designed for anticipated seismic loads including the additional loads resulting from the potential failure of the non-seismic piping and pipe supports.

   - The stress allowables given in ASME Section III NF for Level D loadings are used for qualification of seismic loads.

1.5.3 Overlapping

1.5.3.1 General

Overlapping is used to separate seismically analyzed piping problems. Isolation of non-seismic piping from seismic piping is addressed in Section 1.5.2.3 of this appendix.

Seismic piping that cannot be separated by decoupling as described in Section 1.5.2 of this appendix may be separated using an overlap region. Where an overlap region is used, an adequate number of rigid restraints and bends in three directions to prevent the transmission of motion due to seismic excitation from one end to the other is included. The following criteria is used for applying overlapping:

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1 NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

3.11 Environmental Design of Mechanical and Electrical Equipment

The design criteria with respect to environmental effects on the electrical and mechanical equipment of the Reactor Protective System and the Engineered Safety Feature systems to ensure acceptable performance in all environments (normal and accident) depend upon equipment location and function. Such equipment is qualified to meet its performance requirements under the environmental and operating conditions in which it will be required to function and for the length of time for which its function is required. As far as practical, equipment for these systems is located outside the containment building in a mild environment. If this is not practical, the equipment is qualified for the environment in which it is required to operate.

- For operation under normal conditions the systems are designed and qualified to remain functional after exposures to the following ranges of environmental conditions:

    1. Temperature ranges given in Appendix 3.11A.
    2. Relative humidity ranges given in Appendix 3.11A.
    3. Pressure ranges given in Appendix 3.11A.
    4. Expected integrated radiation exposures for 60 years given in Appendix 3.11A.

- In addition to the normal environment, the mechanical and electrical components required to mitigate the consequences of a design basis accident (DBA) or to attain a safe shutdown of the reactor are designed to remain functional after exposure to the environment anticipated following the specific DBA which they are intended to mitigate. Anticipated environmental conditions and requirements are listed below.

    1. The temperature, pressure, and humidity ranges following the design basis accidents such as the loss of coolant accident (LOCA), the main steam line break (MSLB) or "worst case" combined (LOCA & MSLB) are indicated in Appendix 3.11A.
    2. The time integrated "worst case" post-accident radiation doses are indicated in Appendix 3.11A. Equipment will be designed for the types and levels of external radiation associated with normal operation plus the external radiation associated with the limiting design basis accident (DBA) for which it provides a safety function and for the length of time both during and after the accident for which it is required to be functional. If more than one type of radiation is significant, each type may be considered separately.

[[The COL applicant will make the specific details of the plant specific environment al qualification program available for NRC evaluation. This includes a detailed maintenance/surveillance program and documentation of test reports and analyses.]][1]

3.11.1 Equipment Identification and Environmental Conditions

Appendix 3.11B lists the equipment required to mitigate a DBA or to attain a safe shutdown. Specific equipment for each system is discussed in the appropriate section of the Safety Analysis Report as

1 COL information item; see DCD Introduction Section 3.2.

Table 3.11A-1 Environmental Data

Environmental Parameters[2]	Range and Duration
Containment Vessel - Category A-1 (LOCA)	
Temperature, °F	Figure 3.11A-1A
Pressure, psig	Figure 3.11A-1B
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads plus LOCA, [1,3,4]	$< 4.3 \times 10^7$ Gamma $< 3.5 \times 10^8$ Beta
Chemical Spray	4,400 ppm Boron as H_3BO_3 pH of 7.0-8.5 after 4 hours using Trisodium phosphate
Containment Vessel - Category A-2 (MSLB)	
Temperature, °F	Figure 3.11A-3D 0-30 min. Figure 3.11A-1A after 30 min.
Pressure, psig	Figure 3.11A-1B
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads[1,3]	$< 3.1 \times 10^6$ Gamma
Chemical Spray	4,400 ppm Boron as H_3BO_3 pH of 7.0-8.5 after 4 hours using Trisodium phosphate
Containment Annulus - Category A-3 (Post DBA)	
Temperature, °F	Figure 3.11A-2
Pressure, psig	atmospheric, continuous
Relative Humidity, %	Saturated/Superheated Steam/Air Mixture
Radiation, 60 Yr. TID Rads Plus LOCA[4]	3×10^5 Gamma 4×10^5 Beta
Chemical Spray	N/A
Containment Vessel - Category B (Normal)	
Temperature, °F	60-110, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, continuous
Radiation, 60 Yr. TID Rads[3]	$< 3 \times 10^6$ Gamma
Chemical Spray	N/A
Nuclear Annex/Subsphere - Category C (Normal)	
Temperature, °F	55-104, continuous
Pressure, psig	atmospheric, continuous
Relative Humidity, %	20-90, continuous

- Inconel 625

1. Swelling

Available information indicates that Inconel 625 is highly resistant to radiation swelling. Exposure of Inconel 625 to a fluence of 3 x 10^{22} nvt (E>0.1 MeV) at a temperature of 400°C (725°F) showed no visible cavities in metallographic examinations (Reference 52) so that swelling, if any, would be very minor. Direct measurements made after exposure of Inconel 625 to a fluence 5 x 10^{22} nvt (E>0.1 MeV) at LMFBR conditions showed no evidence of swelling (Reference 53). Further exposure to 6 x 10^{22} nvt (E>0.1 MeV) at 500°C (932°F) showed essentially no swelling as measured by immersion density, but did show small cavities. Thus, Inconel 625 is not expected to swell below fluences of 3 x 10^{22} nvt (E>1 MeV).

2. Ductility

The ductility of Inconel 625 decreases after irradiation. Extrapolation of lower fluence data on Inconel 625 and 500 indicates that the values of uniform and total elongation of Inconel 625 after 1 x 10^{22} nvt (E>1 MeV) are 3 and 6%, respectively.

3. Strength

The value of yield strength for Inconel 625 increases after irradiation in the manner typical of metals. However, no credit is taken for increases in yield strength in the design analyses above the value initially specified.

4.2.1.5 Surveillance Program

4.2.1.5.1 Requirements for Surveillance and Testing of Irradiated Fuel Rods

High burnup performance experience, as described in Section 4.2.3, has provided evidence that the fuel will perform satisfactorily under the design conditions. Two irradiation programs were developed for fuel performance surveillance in Arkansas Nuclear One-Unit 2 (ANO-2). The fuel rods in these 16x16 fuel assemblies are similar to those in the System 80 design.

The first fuel performance program in ANO-2 has been completed. This program followed six standard assemblies through three irradiation cycles. Each assembly contained pre-characterized fuel rods which were examined during refueling shutdowns. The results of the program demonstrated that the fuel assemblies performed reliably through averaged burnups of 37.2 GWd/MTU. Zircaloy oxide thicknesses, fuel rod growth and bowing, and assembly dimensional stability were consistent with initial predictions (Reference 54).

The second program at ANO-2 irradiated two fuel assemblies containing both standard and advanced design fuel rods to extended burnups. Both assemblies were extensively pre-characterized. One assembly was irradiated for three reactor cycles and reached an assembly-averaged burnup of 33 GWd/MTU. A second assembly was exposed to 5 cycles and reached an assembly-averaged burnup of 52 GWd/MTU (Reference 55). Both assemblies were examined after each reactor cycle. Visual and destructive hot cell examinations, oxide thickness measurements, and other dimensional measurements result in the conclusion that the performance of the fuel has been satisfactory (Reference 81).

- Fuel Surveillance Programs

  C-E has conducted a number of fuel surveillance programs on fuel in operating plants. ~~Thus far, a total of thirty-eight poolside~~ Poolside fuel inspection programs of varying detail have been performed by C-E (see Table 4.2-3). A large number of assemblies have been visually examined, and dimensional measurements have been obtained on a large number of these assemblies. Fuel bundle disassembly operations have been conducted either to obtain information on particular performance aspects or as part of test assembly surveillance programs. A listing of these programs and a summary of the results is provided in Reference 70. The results of the C-E poolside inspection program have been used to verify fuel assembly operation and provide data in support of design. A poolside fuel surveillance program ~~is being~~ has been conducted at Palo Verde-1 for C-E's System 80 fuel (see Section 4.2.1.5.1).

  [[The COL applicant will perform on-line fuel failure monitoring and post-irradiation surveillance to detect anomalies.]][2]

4.2.3.2.11 Temperature Transient Effects Analysis

4.2.3.2.11.1 Waterlogged Fuel

The potential for a fuel rod to become waterlogged during normal operation is discussed in Section 4.2.3.2.9. In the event that a fuel rod does become waterlogged at low or zero power, it is possible that a subsequent power increase could cause a buildup of hydrostatic pressure. It is unlikely that the pressure would build up to a level that could cause cladding rupture because a fuel pin with the potential for rupture requires the combination of a very small defect together with a long period of operation at low or zero power.

space

Tests which have been conducted using intentionally waterlogged fuel pins (capsule drive core at SPERT)(References 71 and 72) showed that the resulting failures did eject some fuel material from the rod and greatly deformed the test specimens. However, these test rods were completely sealed, and the transient rates used were several orders of magnitude greater than those allowed in normal operation.

In those instances where waterlogged fuel rods have been observed in commercial reactors, it has not been clear that waterlogging was the cause, and not just the result, of associated cladding failures; and C-E has not observed and is not aware of any case in which material was expelled from waterlogged fuel rods or in which the fuel cladding was significantly deformed in a normal power reactor.

It is therefore concluded that the effect of normal power transients on waterlogged fuel rods is not likely to result in cladding rupture and even if rupture does occur it will not produce the sort of postulated burst failures which would expel fuel material or damage adjacent fuel rods or fuel assembly structural components.

4.2.3.2.11.2 Intact Fuel

The thermal effects of anticipated operational occurrences on fuel rod integrity are discussed in the following paragraphs.

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2 COL information item; see DCD Introduction Section 3.2.

56. "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," Combustion Engineering, Inc., CENPD-178, Rev. 1, August 1981.

57. "Fuel and Poison Rod Bowing," Combustion Engineering, Inc., CENPD-225-P-A (Proprietary), June, 1983.

58. "Application of Zircaloy Irradiation Growth Correlations for the Calculation of Fuel Assembly and Fuel Rod Growth Allowances," Supplement 1 to CENPD-198-P, (Proprietary), December 1977.

"Oh" not zero

59. Pickman, D. 0., "Properties of Zircaloy Cladding," Nuclear Engineering and Design, Vol. 21, No. 2 (1972).

60. Joon, K., "Primary Hydride Failure of Zircaloy Clad Fuel Rods," ANS Transactions, Vol 15, No. 1.

61. Caye, T. E. "Saxton Plutonium Project, Quarterly Progress Report for the Period Ending March 31, 1972," WCAP-3385-31, November 1972.

62. Berman, R. M., Meieran, H. B., and Patterson, P., "Irradiation Behavior of Zircaloy-Clad Fuel Rods Containing Dished End UO_2 Pellets," (LWBR-LSBR Development Program), WAPD-TM-629, July 1967.

63. Baroch, S. J., et al, "Comparative Performance of Zircaloy and Stainless Steel Clad Fuel Rods Operated to 10,000 MWd/MTU in the VBWR," GEAP-4849, April 1966.

64. Megerth, F.H., "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program," Quarterly Progress Report No. 8, August 1969-October 1969, GEAP-10121, November 1969.

65. Megerth, F. H., "Zircaloy-Clad UO_2 Fuel Rod Evaluation Program," Quarterly Progress Report No. 1, November 1967-January 1968, GEAP-5598, March 1968.

66. Garde, A. M., and Pati, S. R., "Gas Release Densification, Swelling and Microstructural Evaluation of Four-Cycle Fuel Rods from Calvert Cliffs-1," C-E NPSD-211, April 1983.

67. San Onofre Nuclear Generating Station, Units 2 & 3, Final Safety Analysis Report, Volume-9, pages 4.2-59 - 4.2-61.

68. Brite, D. W. et al, "EEI/EPRI Fuel Densification Program Final Report," Battelle Pacific Northwest Laboratories, March 1975.

69. J. C. LaVake and M. Gartner, "High Burnup PWR Ramp Test Program, Final Report," DOE/ET/34030-10, December 1984.

70. Andrews, M. G., Smith, G. ~~S~~ P., and ~~Garde, A. M.~~ Shubert, M.A., "Experience and Developments with Combustion Engineering Fuel," Proceedings of ANS Topical Meeting on LWR Fuel Performance, Williamsburg, Virginia, April 1988.

71. Stephan, L. A., "The Response of Waterlogged UO_2 Fuel Rods to Power Bursts," IDO-ITR-105, April 1969.

72. Stephan, L. A., "The Effects of Cladding Material and Heat Treatment on the Response of Waterlogged UO_2 Fuel Rods to Power Burst," IM-ITR-111, January 1970.

73. "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," Combustion Engineering, Inc., CENPD-161-P-A, (Proprietary), April 1986.

74. "Methodology for Core Designs Containing Erbium Burnable Absorbers," CENPD-382-P (Proprietary), October, 1990 and CENPD-382-P, Supplement 1-P (Proprietary), February 1992.

75. "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," Vol. 5, EPRI NP-3966-CCM, April 1985.

76. "Hot Cell Examination of Extended Burnup Fuel from Fort Calhoun," Combustion Engineering Inc. CEND-427, September 1986.

77. Inchikawa, Uchida, Yanagisawa, Nakajima, Nakamura and Kawaski, JAERI; Hayevik, Knudsen and Kolstad, Halden, "Studies of LWR Fuel Performance Under Power Ramping and Power Cycling Utilizing In-Pile Measurement and Fuel Modeling," Proceedings of the ANS Topical Meeting, Williamsburg, Virginia, April 1988.

78. "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel," CEN-386-P-A, August 1992.

79. Marlowe, Mo. O., "High Temperature Isothermal Elastic Modulii of UO_2," Journal of Nuclear Materials, Vol. 33 (1969), pages 242-244.

80. "Joint C-E/EPRI Fuel Performance Evaluation Program," Task C, EPRI NP-218, Project 586-1, Final Report, November, 1976.

81. "The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization," Final Report, DOE/ET/34013-15 UC-523, CEND-432, August, 1994.

Table 4.2-3 C-E Poolside Fuel Inspection Program Summary (Reference 70)

center

Reactor	Shutdown Date/Cycle	Inspection Program Scope[1]
Palisades	1973/1A	VE, GS, CS
Maine Yankee	1974/1	VE, S, SRE, CS
	1975/1A	VE, S
	1977/2	VE, SRE
	1980/4	VE, S, SRE
	1987/9	VE, UT, SRE
Ft. Calhoun	1975/1	VE
	1975/2	VE, CS
	1977/3	VE
	1978/4	VE, DM on DOE Test Bundles
	1980/5	VE, DM on DOE Test Bundles
	1981/6	VE, DM and SRE on DOE Test Bundles
	1982/7	VE, DM and SRE on DOE Test Bundles
St. Lucie-1	1976/1	VE, SRE
	1978/1A	VE
	1985/6	VE, UT, SRE
	1987/7	VE, UT, SRE
Calvert Cliffs-1	1976/1	VE, SRE on C-E/EPRI Test Bundles
	1978/2	VE, SRE on C-E/EPRI Test Bundles
	1979/3	VE, DM SRE on C-E/EPRI Test Bundles
	1980/4	VE, DM SRE on C-E/EPRI Test Bundles
	1982/5	VE, SRE on C-E/EPRI and C-E/BG&E Test Bundles
	1983/6	VE, DM
	1985/7	VE, DM, SRE on C-E/BG&E Test Bundles
	1986/8	VE, DM, UT, SRE on C-E/BG&E Test Bundles
	1988/9	VE, UT, SRE
Calvert Cliffs-2	1984/5	VE, DM, S, SRE
	1987/7	VE, UT, SRE
	1989/8	VE, UT, SRE

Table 4.2-3 C-E Poolside Fuel Inspection Program Summary (Reference 70) (Cont'd.)

Reactor	Shutdown Date/Cycle	Inspection Program Scope[1]
Yankee Rowe	1987/18	VE, UT, SRE
Millstone-2	1977/1	VE
	1982/4	VE
St. Lucie-2	1987/3	VE, UT
	1989/4	VE, UT, SRE
ANO-2	1981/1	VE, DM, SRE on C-E/EPRI Test Bundles
	1982/2	VE, DM
	1983/3	VE, DM, SRE on C-E/EPRI and DOE Test Bundles
	1985/4	VE, DM, SRE on ~~C-E/EPRI~~ DOE Test Bundles
	1986/5	VE, DM, UT
	1988/6	VE, DM, SRE on DOE Test Bundles
San Onofre-2	1984/1	VE, DM
	1985/2	VE, DM
	1987/3	VE, UT, GS, SRE
	1989/4	VE, UT, SRE, DM
San Onofre-3	1985/1	VE, UT
	1988/3	VE, UT, SRE
Palo Verde-1	1987/1	VE, DM
	1989/2	VE, DM
Palo Verde-2	1988/1	VE, DM
Waterford-3	1988/1-2	VE, UT, SRE

[1] VE Visual Examination
GS Gamma-Scanning
CS Crud Sampling
S Sipping
UT Ultrasonic Testing
SRE Disassembly and Single Rod Examinations
DM Dimensional Measurements

5.2 Integrity of Reactor Coolant Pressure Boundary

This section discusses the measures employed to provide and maintain the integrity of the Reactor Coolant Pressure Boundary (RCPB) throughout the facility's design lifetime. The RCPB is defined in accordance with ANSI/ANS 51.1-1983. Included are all pressure containing components such as pressure vessels, piping, pumps, and valves which are:

1. Part of the Reactor Coolant System, or

2. Connected to the Reactor Coolant System, up to and including the following:

    - The outermost containment isolation valve in piping which penetrates the containment;

    - The second of two valves normally closed during reactor operation in piping which does not penetrate the containment.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

The codes and component classifications are listed in Table 5.2-1 and are in accordance with the provisions of 10 CFR 50.55a. The ASME Code Edition and Addenda used in the design and construction of the reactor coolant pressure boundary components are specified in Table 1.8-6 and discussed in Section 1.8. [[If ASME Code Editions and Addenda other than those specified in Table 1.8-6 and Section 1.8 are used they will be identified to the Commission by the COL applicant and shall have been endorsed by 10 CFR 50.55a.]][1]

5.2.1.2 Applicable Code Cases

Reactor Coolant Pressure Boundary components are fabricated in accordance with the ASME Code, Section III.

The applicable ASME Code cases listed in Table 1.8-7 are utilized in the plant design and manufacturing.

[[If code cases other than those specified in Table 1.8-7 are used they will be identified to the Commission by the COL applicant and shall have been endorsed by Regulatory Guides 1.84, 1.85 or 1.147 as applicable.]][1] Code cases not endorsed by the above Regulatory Guides may be used with specific authorization from the Commission under 10 CFR 50.55a.

5.2.2 Overpressure Protection

5.2.2.1 Design Bases

Appendix 5A presents the design bases for sizing the overpressurization protection system. The loss of load transient which is used to size the primary safety valves is not intended to be used as a design transient for any other NSSS equipment.

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1 Col information item; see DCD Introduction Section 3.2.

A surface examination of all exposed surfaces and 100% volumetric examination by ultrasonic methods will be conducted at about ten-year intervals during the plant shutdown coinciding with the in-service inspection schedule as required by the ASME Code, Section XI.

Each flywheel will receive a preservice baseline inspection incorporating the methods defined above for an inservice inspection. Examination procedures and acceptance criteria will be in accordance with the ASME Code Section III.

5.4.1.2 Description

Table 5.4.1-1 lists the principal parameters of the reactor coolant pumps and Figure 5.4.1-1 depicts the arrangement of the pump and motor. Reactor coolant pump supports are discussed in Section 5.4.14. The pump piping and instrument diagram is given in Figure 5.1.2-2.

The four reactor coolant pumps are vertical, single stage bottom suction, horizontal discharge, motor-driven centrifugal pumps. The pump impeller is splined and locked to its shaft. Pump shaft alignment is maintained by a water lubricated radial bearing within the pump and by radial and thrust bearings located in the motor stand. The pump and motor shafts are directly connected by a coupling.

The shaft seal assembly consists of two face-type, mechanical seals in series, with controlled leakage bypass to provide the same pressure differential across each seal. The seal assembly is designed for 2500 psi differential and to reduce the leakage pressure from Reactor Coolant System pressure to the volume control tank pressure. A third, face-type, low-pressure vapor seal at the top is designed to withstand system operating pressure when the pumps are not operating. The leakage past the second pressure seal and the controlled leakage is piped to the volume control tank in the Chemical and Volume Control System. Leakage past the low-pressure vapor seal is collected and piped to the reactor drain tank.

The temperature of the water in the seal assembly is maintained within acceptable limits by a water-cooled heat exchanger. Water is also injected into the seal area from an external seal injection system. The performance of the shaft seal system is monitored by pressure and temperature sensing devices in the seal system. The seal assembly can be replaced without draining the pump casing or removing the shaft.

ATTACHMENT 1

The seal assemblies are designed to limit seal leakage plus controlled bypass flow to approximately the values given below:

Seal leakage plus controlled bypass flow, per pump:

All seals functioning (normal)	3.9 gpm
One seal functioning (abnormal)	4.6 gpm

The motor is sized for continuous operation at the flows resulting from four-pump or one-pump operation with 1.0 to 0.74 specific gravity water. The motors are designed to start and accelerate to speed under full load with a drop to 80 percent of normal rated voltage at the motor terminals.

Each motor is provided with an anti-reverse rotation device. The device is designed to prevent impeller rotation in the reverse direction due to each of the following conditions: motor starting torque, if the motor was incorrectly wired for reverse rotation; and, reactor coolant flow through the pump in the reverse direction due to the largest remaining pipe break after application of leak before break as described in Section 3.6, which could result in reverse flow through the pump.

Attachment 1

The System 80+ Reactor Coolant Pump (RCP) shaft seals are cooled by (1) seal injection water from the Chemical and Volume Control System, and (2) the Component Cooling Water System (CCWS) through a high pressure seal cooler. Pump operation may continue indefinitely provided either seal injection flow or the CCWS is available. The System 80+ Standard Plant design includes an additional support system, the Dedicated Seal Injection System, which is not included in the System 80 design. This system features a positive displacement pump to provide a diverse means of seal injection to the RCPs if normal means of seal cooling are lost.

In the event of loss of either seal injection to the seal assembly or loss of CCWS flow to the high pressure seal cooler, the seal cooling water temperature will increase. Performance tests and analyses have shown that a minimum margin of 22°F exists between the seal cooling water outlet temperature and the seal cooling water temperature limit specified by the pump manufacturer.

Table 5.4.10-2 Pressurizer Tests

Component	Tests[1]
Heads	
Plates	UT, MT
Cladding	UT, PT
Shell	
Plates	UT, MT
Cladding	UT, PT
Heaters	
Tubing	UT, PT
Centering of elements	RT
End Plug	UT, PT
Nozzle (Forgings)	UT, MT
Studs	UT, MT
Welds	
Shell, longitudinal	RT, MT
Shell, circumferential	RT, MT
Cladding	UT, PT
Nozzles	RT, MT
Nozzle safe ends	RT, PT
Instrument connections	PT
Support Skirt	MT, RT
Temporary attachment after removal	MT
All welds after hydrostatic test	MT or PT
Heater assembly, end plug weld	PT

[1]

UT = Ultrasonic testing
MT = Magnetic particle testing
PT = Dye-penetrant testing
RT = Radiographic testing

Systems susceptible to an ISLOCA are to be designed so that all of the following conditions are satisfied without any operator action... :

- the system retains its structural integrity throughout the event (structural integrity is preserved if, by definition, the system maintains its pressure boundary despite distortion and/or loss of function);

- any leakage caused by the event is limited to the makeup system capabilities; and

- offsite doses are limited to a small fraction of those specified in 10 CFR 100 as is assumed in the design bases for the Chapter 15 analyses.

3.3 Compliance Methods

Design responses to ISLOCA challenges discussed in this report constitute :

- increasing the design pressure rating of equipment or systems, Option A, and

- incorporating design features which terminate and limit the scope of the ISLOCA event, Option B.

Option A design features rely on inherent physical attributes of a system or subsystem which will prevent failure when it is pressurized to normal RCS operating pressure. Option A features do not require any immediate action by equipment or operators to satisfy the ISLOCA acceptance criteria. This approach is intended to provide the optimum protection against ISLOCA challenges and to allow the operator the necessary time to properly assess and restore the system to normal conditions. [[*Examples of*]][1] Option [[*A features satisfying the ISLOCA acceptance criteria include:*]][1]

Both sides

- locating the system or subsystem completely within containment;

- designing the system or subsystem to normal RCS design pressure;

- [[*designing the system or subsystem to a pressure of at least 40% of the RCS normal pressure. Austenitic stainless steel piping will use a minimum wall thickness corresponding to standard weight for sizes less than 16 inch NPS and schedule 40 for 16 inch NPS and larger sizes;*]][1] and

- physically separating the system or subsystem from the RCS during conditions when the RCS pressure exceeds its design pressure.

Option B design features are design responses to ISLOCA events consisting of specific equipment and instrumentation which perform actions to prevent or mitigate the consequences of an ISLOCA. Option B design responses that have been considered will not require operators to prevent or mitigate the event, but will eventually require operators to perform remedial action, inspection of equipment following the event and returning the plant systems to normal configuration.

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1 NRC Staff approval is required prior to implementing a change in this information; see DCD Introdution Section 3.5.

Option B design features are intended to be applied to systems for which it is impractical to apply Option A design features.

Examples of Option B design features are:

- the isolation of a system or subsystem in the pressurization pathway at the interface between the lower pressure system or subsystem and its pressurization source; and

- pressure relief to limit the pressurization to within the design capabilities of the system.

3.4 ISLOCA Evaluation Process

The evaluation performed for this report of ISLOCA events and the subsequent determination of appropriate design responses is characterized by the following steps.

(1) The RCS P&ID's in Chapter 5, were reviewed to identify all systems or subsystems that interface with the RCS. All systems or subsystem within containment or designed to full RCS operating pressure were noted to meet the acceptance criteria. The remaining systems or subsystems that were located outside containment and were designed for less than RCS pressure were evaluated. Subsequently, these systems were evaluated to determine pressurization pathways from the RCS. These pathways represent all the possible ways of pressurizing low-pressure systems connected to the RCS.

(2) These pressurization pathways were further evaluated to determine the impact of systems or subsystems interfacing with the initiating pressurization pathway. A summary of these pressurization pathways is illustrated in Figures 5E.3.4-1 to 5E.3.4-10. The figures show a pyramid structure beginning at the top with a system or subsystem that is directly connected with the RCS. A given pyramid represents a class of potential ISLOCA events characterized by a finite set of pressurization pathways. These pathways identify the various systems that can potentially be pressurized if the interfacing system valves were postulated to be open.

(3) The pressurization pathways were analyzed to identify any pattern which may suggest a hierarchy of design responses to prevent or mitigate the ISLOCA event. Any hierarchical pattern would be motivated by the desire to satisfy the ISLOCA acceptance criteria with a design response commensurate with the perceived benefit and without degradation to safety.

(4) Each interface in the pressurization pathways was analyzed to identify the type and location of design response to satisfy the ISLOCA acceptance criteria.

A review of the pressurization pathways in Figures 5E.3.4-1 to 5E.3.4-10 led to the following observations.

- The following systems or subsystems are directly connected to the RCS for one or more ISLOCA events (i.e., for one or more pressurization pathways):

    - Safety Injection System (SIS),

    - Shutdown Cooling System (SCS),

6.5 Containment Spray System

6.5.1 Design Bases

6.5.1.1 Summary Description

The Containment Spray System (CSS) is a safety grade system designed to reduce containment pressure and temperature from a main steam line break or loss-of-coolant-accident and to remove fission products from the containment atmosphere following a loss of coolant accident. Fission product removal is required so that in the event of containment leakage, activity at the site boundary due to radioactive iodine will be reduced. ~~No spray additives are required.~~ No spray additive is required for pH control during the initial stage of a LOCA. However, additive is required for post-accident pH control of the spray water.

The CSS uses the In-Containment Refueling Water Storage Tank (IRWST) and has two independent trains (two containment spray pumps, two containment spray heat exchangers, two independent spray headers, and associated piping valves and instrumentation). The system is shown in Figures 6.3.2-1A, 6.3.2-1B and 6.3.2-1C. Post-accident pH control of the sprayed fluid is provided using trisodium phosphate dodecahydrate that is stored in the Holdup Volume Tank (HVT).

The CSS provides sprays of borated water to the containment atmosphere from the upper regions of the containment. The spray flow is provided by the containment spray pumps which take suction from the IRWST. The containment spray pumps start upon the receipt of a Safety Injection Actuation Signal (SIAS) or a Containment Spray Actuation Signal (CSAS). The pumps discharge through the containment spray heat exchangers and the spray header isolation valves to their respective spray nozzle headers, then into the containment atmosphere. Spray flow to the containment spray headers is not provided until a CSAS automatically opens the containment spray header isolation valves. The spray headers are located in the upper part of the containment building to allow the falling spray droplets time to approach thermal equilibrium with the steam-air atmosphere. Condensation of the steam by the falling spray results in a reduction in containment pressure and temperature.

The CS pumps are designed to be functionally interchangeable with the Shutdown Cooling System (SCS) pumps. The CS pumps and CS heat exchangers can be used as a backup to the SCS pumps and heat exchangers to provide residual heat removal or to provide cooling of the IRWST.

6.5.1.2 Functional Design Bases

The following functional design bases apply to the CSS:

- The CSS is designed to remove heat from the containment atmosphere following either a loss of coolant accident, control element assembly ejection, or a main steam or feedwater line break inside containment to aid in the reduction of containment pressure and temperature.

- The heat removal capacity of the system in conjunction with other acceptably defined active or passive heat sinks will be sufficient to prevent exceeding containment design pressure and temperature and to reduce containment pressure to at least one-half the calculated peak pressure in twenty-four hours for the above mentioned events.

- The CSS is designed to perform its heat removal function without undue penalty on SIS performance.

- The CSS is designed to aid in the removal of fission products from the containment atmosphere.

Table 8.2-1 Failure Modes and Effects Analysis for the Offsite Power System

Component	Malfunction	Resulting Consequences
1. Transmission System - Preferred Switchyard Interface I	Loss of Power	(a) The switchyard PCB connecting the unit to the system (Switchyard) trips automatically (b) If main turbine generator is available, all unit and Class 1E auxiliaries continue to receive an uninterrupted flow of power from the main turbine generator through the main generator circuit breaker. (c) If the main turbine generator is available, all unit and Class 1E auxiliaries continue to receive an uninterrupted flow of power from the main turbine generator through the main generator circuit breaker.
2. Preferred Switchyard Interface I power circuit breaker connecting the step-up transformers to the switchyard or Circuit from Preferred Switchyard Interface I to main unit transformer or Main Unit Transformer	Loss of one due to a fault or breaker failure	(a) The faulted equipment is isolated by protective relaying and protective equipment. (b) The other independent offsite circuit remains unaffected (c) If on-line, the unit main turbine generator is automatically tripped. (d) If the unit main turbine generator is off line, the other offsite circuit from Preferred Switchyard Interface II is available for the 4,160V Permanent Non-Safety, 13.8KV Non-Safety and Class 1E Division Auxiliaries via the Reserve Auxiliary Transformer
3. Transmission System Preferred Switchyard Interface II	Loss of Power	(a) No consequence to unit (b) The faulted equipment is isolated by protective relaying and protective equipment

Replace w/ Attached from CESSAR-DC

TABLE 8.2-1

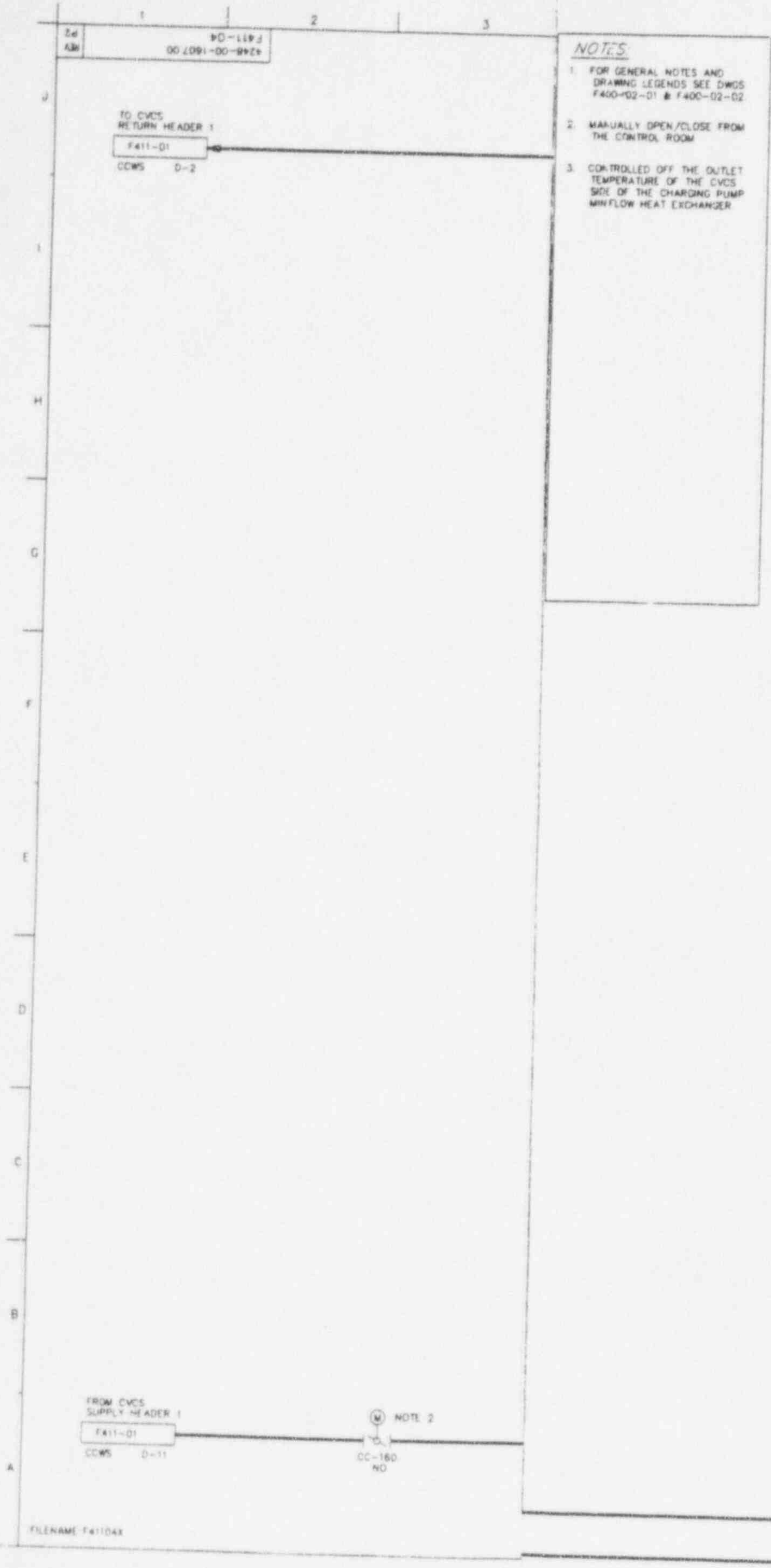
(Sheet 1 of 3)

FAILURE MODES AND EFFECTS ANALYSIS FOR THE OFFSITE POWER SYSTEM

Component	Malfunction	Resulting Consequences
1. Transmission System - Preferred Switchyard Interface I	Loss of Power	(a) The switchyard PCB connecting the unit to the system (switchyard) trips automatically. (b) If main turbine generator is available, all unit and Class 1E auxiliaries continue to receive an uninterrupted flow of power from the main turbine generator through the main generator circuit breaker. Insert for DCD Tbl 8.2-1 (Pg 8.2-8) (c) If the main turbine generator is not available, the 13.8KV non-safety bus may receive power from its alternate source. On the 4,160V Permanent Non-Safety bus, an automatic transfer takes place and the 4,160V Safety buses continue to receive an uninterrupted flow of power from the Preferred Switchyard Interface II.
2. Preferred Switchyard Interface I power circuit breaker connecting the step-up transformers to the switchyard	Loss of one due to a fault or breaker failure	(a) The faulted equipment is isolated by protective relaying and protective equipment. (b) The other independent offsite circuit remains unaffected.

Chapter 9 Contents (Cont'd.)

Chapter 9 Tables



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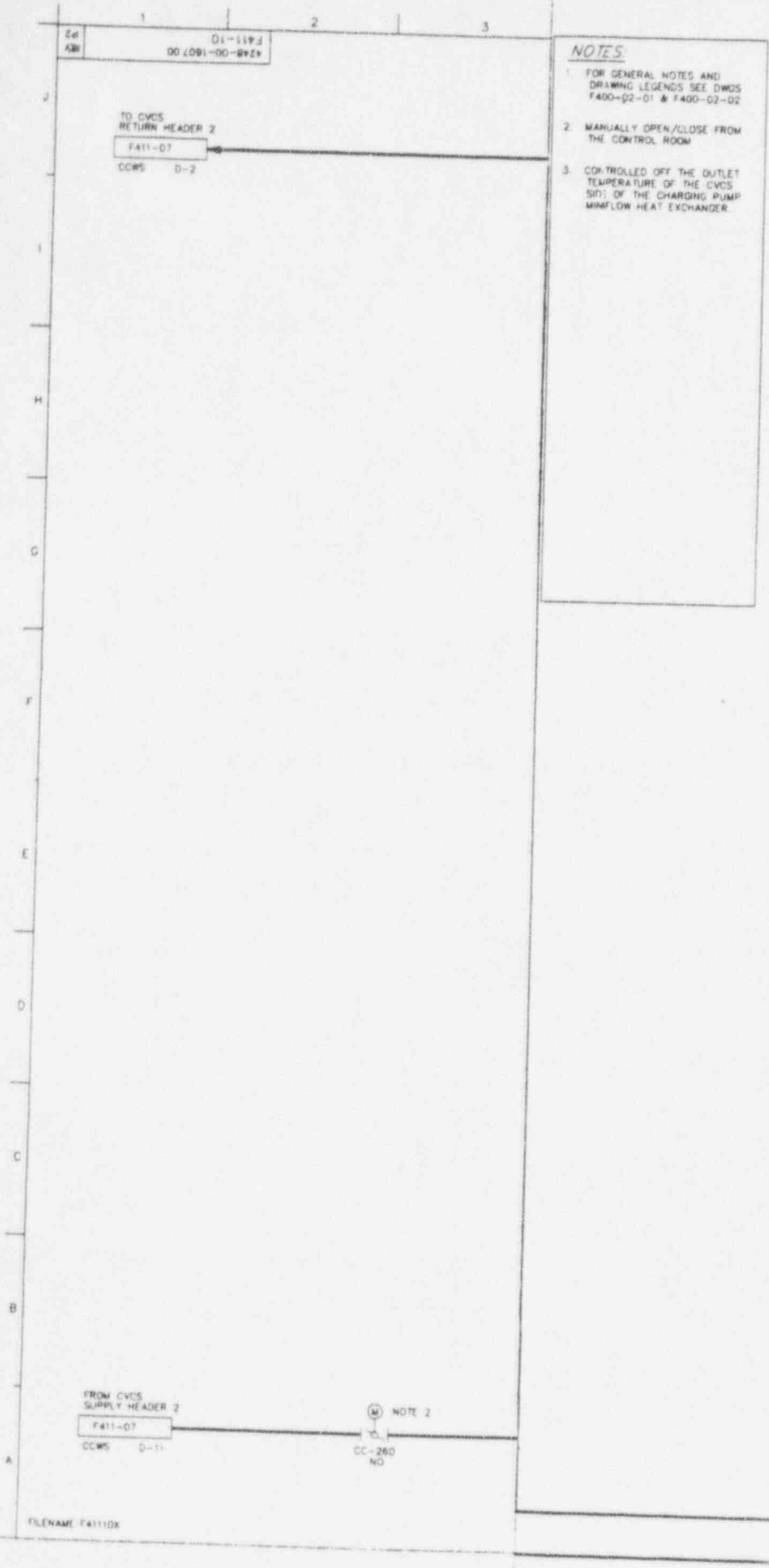

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Figure 9.2.2-1 Sheet 10 of 18

measurements can be obtained at 24 hours after plant shutdown. These features are consistent with the NRC recommendations in the Commission's Staff Requirements Memorandum on SECY-93-087, dated July 21, 1993.

2. The system design appropriately integrates the normal and post-accident functions so as to maximize the familiarity of plant operators with post-accident operation(s). The design utilizes common sample lines and points for both normal and post-accident sampling to the maximum extent possible.

3. Any function which is not performed during normal sampling operations has testing capability to enable periodic verification of operability and familiarization with system operation.

4. The system provides the capability for obtaining reactor coolant and containment atmosphere samples for the analyses identified in paragraph H.1 above. These analyses are performed either continuously, or by grab sample and analysis. Backup grab samples are provided for any on-line monitoring capability consistent with Clarification (8) of NUREG-0737, Item II.B.3. Gas chromatography equipment is not used for on-line analysis.

5. Provisions are made for dilution of liquid and gas grab samples for subsequent laboratory analysis. Dilution of the liquid and gas grab samples for subsequent laboratory analysis. Dilution of the liquid and gas grab samples shall be performed either at the sampling station, or in the laboratory, whichever leads to simpler equipment consistent with ALARA practices. Collection and dilution of the post-accident samples is performed remotely to the maximum extent feasible, and is the responsibility of the COL applicant.

6. All remotely operated valves required for post-accident sampling have assured power supplies and system level reset features which allow reopening of the valves after containment isolation without clearing the isolation signal for other containment isolation valves. Individual valve reset features are provided to allow opening of individual sampling valves after system reset. Valves and operators which are inaccessible during an accident are environmentally qualified to ensure operability under accident conditions.

7. Two independent non-1E sources are available to provide electrical power for post-accident sampling. After loss of normal off-site power, power is automatically supplied from on site. During loss of offsite power, an alternate backup power source, not necessarily the vital 1E bus, is available that can be energized in sufficient time to meet the time limit requirements of 8 hours and 24 hours specified in paragraph H.1 above for boron, and total dissolved gas and radiological measurements, respectively.

- Fire Protection

In the event of a fire, reactor coolant boron sampling is available to verify shutdown margin, consistent with 10 CFR 50, Appendix R.

significant quantities of fluorides or chlorides and significant amounts of dissolved oxygen are present. During heatup, any dissolved oxygen is scavenged by the hydrazine, eliminating the potential for general corrosion. At higher temperatures, the hydrazine decomposes, forming ammonia. The resultant increase in pH aids in the development and maintenance of passive oxide films on Reactor Coolant System surfaces. It has been well established that the corrosion rates of Ni-Cr-Fe Alloy and 300-series stainless steels decrease with time when exposed to normal reactor coolant chemistry conditions, approaching low steady state values within approximately 200 days. A high pH minimizes corrosion product release and assists in the rapid development of the passive oxide film. Most of the film is established during pre-core operations within seven days at hot, high pH conditions.

To aid in maintaining the pH during system passivation, lithium in the form of lithium hydroxide, is added to the coolant and maintained within a 1-2 ppm lithium-7 range.

At power, oxygen concentration is limited by maintaining excess dissolved hydrogen gas in the coolant. The excess hydrogen forces the water decomposition/synthesis reaction in the reactor core toward water synthesis, rather than hydrogen and oxygen formation. Oxygen added via makeup water is removed in this way.

In order to minimize the effect of crud deposition on the reactor core heat transfer surfaces, lithium-7 hydroxide additions are made. Lithium-7 hydroxide produces pH conditions within the reactor coolant at operating temperatures that reduce the corrosion product solubility and, hence, the dissolved crud inventory in the circulating reactor coolant. The elevated pH promotes conditions within the coolant for selective deposition of corrosion products on cooler surfaces (steam generators) rather than hotter surfaces (core). An additional advantage is the formation of a more stable and tenacious passive oxide layer on out-of-core system surfaces. The lithium concentration is maintained within a 0.2-2.2 ppm lithium-7 range during normal operation.

9.3.4.1.3.3 Reactivity Control

Boron concentration is normally controlled by feed-and-bleed. To change concentration, the makeup system supplies either reactor makeup water or boric acid to the VCT, and the letdown stream is diverted to the holdup tank via the pre-holdup ion exchanger and the gas stripper. Toward the end of a fuel cycle, with low boric acid concentration in the coolant, feed-and-bleed to further reduce boron concentration becomes inefficient, and the deborating ion exchanger is used. The deborating ion exchanger contains an anion resin initially in the hydroxyl form, which is converted to a borate form as boron is removed from the reactor coolant.

Confining Dike Structure

9.3.4.1.4 ~~Interface Requirements~~

[[The structures housing the Boric Acid Storage Tank, the Reactor Makeup Water Tank, and the Holdup Tank are out of scope items which shall be provided by the applicant. The licensee shall verify that the structural interface requirements listed in this section are met to ensure adequacy with the System 80+ Standard Design.]][1]

dike surrounding is

- The structures ~~housing~~ the tanks ~~shall be~~ designed to meet the requirements of NRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." Specifically, the tanks ~~shall~~ are

contained within a reinforced concrete dike structure described in Section 3.8.4.1.11, and Section 11.6 of Appendix 3.8A. Specific design requirements include the following:

1 COL information item; see DCD Introduction Section 3.2.

~~be~~ located within a seismically-designed dike or retention pond of sufficient height/size capable of preventing runoff in the event of tank overflow/rupture.

- All three tanks are ~~may be~~ located within a common dike structure designed to contain the maximum combined liquid inventory in the tanks.

- The licensee shall ensure that the dike structure/dike will be designed to comply ~~design complies~~ with applicable state and local regulations.

9.3.4.2 System Description

9.3.4.2.1 System

The normal reactor coolant flow path through the CVCS is indicated by the heavy lines on the flow diagrams (Figure 9.3.4-1, Sheets 1 through 4). Design parameters for the major components are shown in Table 9.3.4-4. Normal operating parameters for the CVCS are listed in Table 9.3.4-5. Process flow data is shown in Table 9.3.4-6.

Letdown flow from the RCS passes through the tube side of the regenerative heat exchanger where an initial temperature reduction takes place via heat transfer to cooler charging fluid on the shell side of the heat exchanger. The regenerative heat exchanger is designed to cool letdown flow to less than 450°F for all normal operations and to heat the charging flow by a minimum of 100°F. A final temperature reduction to the purification subsystem operating temperature is made by the letdown heat exchanger. The letdown heat exchanger is sized to cool inlet water from the maximum regenerative heat exchanger outlet temperature to 120°F (or lower) for most operating conditions. Both the letdown and the regenerative heat exchangers are designed for full RCS pressure and both are located inside containment.

Letdown fluid pressure is reduced from RCS pressure to the operating pressure of the purification subsystem in two stages. The first pressure reduction occurs at the letdown orifices and the second occurs at the letdown control valves located downstream of the orifices. The letdown orifices are located inside containment. The letdown orifices are sized to pass the maximum letdown flow at full RCS pressure with one control valve full open. The orifice provides the pressure reduction necessary to minimize erosion of the letdown control valve seating surfaces during normal RCS operations. A bypass valve around the orifices is provided for low pressure operations. The process flow is then filtered via the purification filter purified via a purification ion exchanger, and sprayed into the VCT. An excess hydrogen inventory is maintained in the RCS by keeping a hydrogen overpressure on the VCT contents.

The charging pumps normally take suction from the VCT and discharge to the RCS. During normal operations, one charging pump is running and the other is in standby. An interlock is provided so that no more than one charging pump is operating at a time during all modes of plant operation. One letdown and one charging pump flow control valve are normally selected for use. Seal injection water is supplied to the Reactor Coolant Pumps (RCPs) by diverting a portion of the charging flow just downstream of the charging pumps. This seal flow is then heated in the seal injection heat exchanger to approximately 125°F before filtering. Once the flow has been filtered, the seal injection fluid is distributed to the four RCPs. The undiverted charging fluid is sent to the regenerative heat exchanger where it is heated before injection into the RCS.

A chemical addition tank and a chemical addition metering pump are used to transfer chemical additives to the charging line downstream of the seal injection takeoff connection. Sufficient connections exist

During normal operation, return air from the control room is mixed with a small quantity of outside air for ventilation, is filtered and conditioned in the control room air-conditioning unit, and is delivered to the control room through supply ductwork. Duct-mounted heating coils and humidification equipment provide final adjustments to the control room temperature and humidity for maintaining normal comfort conditions.

Each air inlet structure is provided with redundant radiation monitoring devices and a smoke detector. The designated MCR filtration units and ventilation fan start automatically on a Safety Injection Actuation Signal (SIAS) or high radiation signal. Upon failure of the designated filtration unit, the redundant filtration unit starts automatically. The MCR filtration unit filters particulates and potential radioactive iodines from all of the return air, and delivers the filtered air to the inlet of the main air-conditioning unit.

The Technical Support Center air-conditioning system consists of an air-conditioning unit, return air and smoke purge fans, and an emergency filter unit. The TSC is maintained at 1/8" water gauge positive pressure with respect to adjacent areas during post-accident conditions. A common supply air header and common outside air intake dampers are shared by the TSC and the control room to protect the TSC from the contaminants in the outside air intakes. The TSC can be isolated from the Main Control Room by using manual controls. The TSC is automatically isolated if control room pressurization falls below its design value.

The TSC is provided with shielding protection from direct radiation from an external radioactive cloud and internal radioactive sources. The combined effect of all radiation protection measures is designed to be adequate to limit the overall calculated radiation exposure to the personnel inside the TSC to the requirements of General Design Criteria 19. The computer room air-conditioning system consists of two 100% air-conditioning units and associated fans. Both the Technical Support Center and computer room air-~~handling~~ conditioning systems are non-safety and non-seismic.

The balance of control complex air-conditioning systems consists of two redundant air-conditioning units, each with roughing filters, safety-related chilled water cooling coils and fans serving Division I electrical rooms, Channel A and Channel C. Two equal units ~~are serving~~ serve Division II, Channels B and D. Each Division will function with one of the redundant air conditioning units delivering filtered, conditioned air to the various electrical equipment rooms including essential battery rooms. Chilled water is supplied from the Essential Chilled Water System. Each Division also contains redundant battery rooms with fan operating continuously to maintain the hydrogen concentration below two percent. Outlet ducts in battery rooms are located near ceiling for hydrogen control.

The Remote Shutdown Panel Room is located in the Division I area. Normally this room is cooled by the 70' Elevation Division I Electrical Equipment Room Air ~~handling~~ Conditioning Unit. For redundancy purposes, the Remote Shutdown Panel Room is also cooled by a Division II- powered air ~~handling~~ conditioning unit which receives Division II Safety-related Chilled Water.

Return air from the various essential electrical equipment areas is mixed with a portion of outside air for ventilation, is filtered and conditioned in the air-conditioning unit, and is delivered to the rooms through supply ductwork. Duct-mounted heating coils provide final adjustments to temperature in selected equipment rooms.

The Operation Support Center, personnel decon rooms, Break Room, Shift Assembly and Offices, Radiation Access Control and Central Alarm Station and Security Group areas all are served by an individual air conditioning unit consisting of a centrifugal fan, non-safety related chilled water coil and

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52[1]

Regulatory Guide 1.52 Position C	System 80+
1. Environmental Design Criteria	
~~1.1~~ a. The design of an engineered-safety-feature atmosphere cleanup system should be based on the maximum pressure differential, radiation dose rate, relative humidity, maximum and minimum temperature, and other conditions resulting from the postulated DBA and on the duration of such conditions.	Complies.
~~1.2~~ b. The design of each ESF system should be based on the radiation dose to essential services in the vicinity of the adsorber section, integrated over the 30-day period following the postulated DBA. The radiation source term should be consistent with the assumptions found in Regulatory Guides 1.3, 1.4 and 1.25. Other engineered safety features, including pertinent components of essential services such as power, air, and control cables should be adequately shielded from the ESF atmosphere cleanup systems.	Complies, except radiation source term is consistent with NUREG 1465 in lieu of Regulatory Guide 1.4.
~~1.3~~ c. The design of each adsorber should be based on the concentration and relative abundance of the iodine species (elemental, particulate, and organic), which should be consistent with the assumptions found in Regulatory Guides 1.3, 1.4 and 1.25.	Complies, except radiation sources term is consistent with NUREG 1465 in lieu of Regulatory Guide 1.4.
~~1.4~~ d. The operation of any ESF atmosphere cleanup system should not deleteriously affect the operation of other engineered safety features such as a containment spray stem, nor should the operation of other engineered safety features such as a containment spray system deleteriously affect the operation of any ESF atmosphere cleanup system.	Complies.
~~1.5~~ e. Components of systems connected to compartments that are unheated during a postulated accident should be designed for post-accident effects of both the lowest and highest predicted temperatures.	Complies.

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(Revision 2)

[1] Design requirement of this Regulatory Guide as applicable to the System 80+ Control Complex Ventilation System, safety-related components of other filtration trains and carbon adsorbers credited with more than 70% efficiency in 10CFR20 and 10CFR50, Appendix I analyses.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C		System 80+
2.	**System Design Criteria**	
~~2.1~~ a.	ESF atmosphere cleanup systems designed and installed for the purpose of mitigating accident doses should be redundant. The systems should consist of the following sequential components: (1) demisters, (2) prefilters (demisters may serve this function), (3) HEPA filters before the adsorbers, (4) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as metal zeolites), (5) HEPA filters after the adsorbers, (6) ducts and valves, (7) fans, and (8) related instrumentation. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration.	Complies, except for Control Complex Ventilation System demisters are not provided. Water droplets will not be entrained in the airstream. Humidity control is provided by safety-related air-conditioning system which has provisions for both dehumidifying and heating to maintain relative humidity below 60%. Heaters are provided in the filtration unit.
~~2.2~~ b.	The redundant ESF atmosphere cleanup systems should be physically separated so that damage to one system does not also cause damage to the second system. The generation of missiles from high-pressure equipment rupture, rotating machinery failure, or natural phenomena should be considered in the design for separation and protection.	Complies.
~~2.3~~ c.	All components of an engineer-safety-feature atmosphere cleanup system should be designated as Seismic Category I (see Regulatory Guide 1.29) if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environments.	Complies.
~~2.4~~ d.	If the ESF atmosphere cleanup system is subject to pressure surges resulting from the postulated accident, the system should be protected from such surges. Each component should be protected with such devices as pressure relief valves so that the overall system will perform its intended function during and after the passage of the pressure surge.	Not applicable. The systems are located outside of the containment and not exposed to pressure surges.
~~2.5~~ e.	In the mechanical design of the ESF system, the high radiation levels that may be associated with buildup of radioactive materials on the ESF system components should be given particular consideration. ESF system construction materials should effectively perform their intended function under the postulated radiation levels. The effects of radiation should be considered not only for the demisters, heaters, HEPA filters, adsorbers, and fans, but also for any electrical insulation, controls, joining compounds, dampers, gaskets, and other organic-containing materials that are necessary for operation during a postulated DBA.	Complies.

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Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52.Position C		System 80+
2.	System Design Criteria (Cont'd)	
~~2.6~~ f.	The volumetric air flow rate of a single cleanup train should be limited to approximately 30,000 ft^3/min. If a total system air flow in excess of this rate is required, multiple trains should be used. For ease of maintenance, a filter layout three HEPA filters high and ten wide is preferred.	Complies.
~~2.7~~ g.	The ESF atmosphere cleanup system should be instrumented to signal, alarm, and record pertinent pressure drops and flow rates at the control room.	Complies.
~~2.8~~ h.	The power supply and electrical distribution system for the ESF atmosphere cleanup system described in Section C.2.a above should be designed in accordance with Regulatory Guide 1.32. All instrumentation and equipment controls should be designed to IEEE Standard 279. The ESF system should be qualified and tested under Regulatory Guide 1.89. To the extent applicable, Regulatory Guides 1.30, 1.100, and 1.118 and IEEE 334 should be considered in the design.	Complies.
2.9 i	Unless the applicable engineered-safety-feature atmosphere cleanup system operates continuously during all times that a DBA can be postulated to occur, the system should be automatically activated upon the occurrence of a DBA by (1) a redundant engineered-safety-feature signal (i.e., temperature, pressure) or (2) a signal from redundant Seismic Category I radiation monitors.	Complies.
2.10 j	To maintain radiation exposures to operating personnel as low as is reasonably achievable during plant maintenance, ESF atmosphere cleanup systems should be designed to control leakage and facilitate maintenance in accordance with the guidelines of Regulatory Guide 8.8. The ESF atmosphere cleanup train should be totally enclosed. Each train should be designed and installed in a manner that permits replacement of the train as an intact unit or as a minimum number of segmented sections without removal of individual components.	Complies.
2.11 k.	Outdoor air intake openings should be equipped with louvers, grills, screens, or similar protective devices to minimize the effects of high winds, rain, snow, ice, trash, and other contaminants on the operation of the system. If the atmosphere surrounding the plant could contain significant environmental contaminants, such as dusts and residues from smoke cleanup systems from adjacent coal burning power plants or industry, the design of the system should consider these contaminants and prevent them from affecting the operation of any ESF atmosphere cleanup system.	Complies.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C		System 80+
2. ~~2.12~~ e.	System Design Criteria (Cont'd.) ESF atmosphere cleanup system housings and ductwork should be designed to exhibit on test a maximum total leakage rate as defined in Section 4.12 of ANSI N509-1976. Duct and housing leak tests should be performed in accordance with the provisions of Section 6 of ANSI N510-1975.	Complies.
3.	**Component Design Criteria and Qualification Testing**	
~~3.1~~ a.	Demisters should be designed, constructed, and tested in accordance with the requirements of Section 5.4 of ANSI N509-1976. Demisters should meet UL Class 1 requirements.	Not applicable. See response to Regulatory Position 2.~~1~~a above.
~~3.2~~ b.	Air heaters should be designed, constructed, and tested in accordance with the requirements of Section 5.5 of ANSI N509-1976.	Complies.
~~3.3~~ c.	Materials used in the prefilters should withstand the radiation levels and environmental conditions prevalent during the postulated DBA. Prefilters should be designed, constructed, and tested in accordance with the provisions of Section 5.3 of ANSI N509-1976.	Complies.
~~3.4~~ d.	The HEPA filters should be designed, constructed, and tested in accordance with Section 5.1 of ANSI N509-1976. Each HEPA filter should be tested for penetration of dioctyl phthalate (DOP) in accordance with the provisions of MIL-F-51068 and MIL-STD0282.	Complies.
~~3.5~~ e.	Filter and adsorber mounting frames should be constructed and designed in accordance with the provisions of Section 5.6.3 of ANSI N509-1976.	Complies.
~~3.6~~ f.	Filter and adsorber banks should be arranged in accordance with the recommendations of Section 4.4 of ERDA 76-21.	Complies.
~~3.7~~ g.	System filter housings, including floors and doors, should be constructed and designed in accordance with the provisions of Section 5.6 of ANSI N509-1976.	Complies.
~~3.8~~ h.	Water drains should be desinged in accordance with the recommendations of Section 4.5.8 of ERDA 76-21.	Complies.
~~3.9~~ i.	The adsorber section of the ESF atmosphere cleanup system may contain any adsorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodides) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this adsorbent is discussed in this guide.	Complies.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C	System 80+
3. Component Design Criteria and Qualification Testing (Cont'd) i. Cont'd Each original or replacement batch of impregnated activated carbon used in the adsorber section should meet the qualification and batch test results summarized in Table 5.1 of ANSI N509-1976. In this table, a "qualification test" should be interpreted to mean a test that establishes the suitability of a product for a general application, normally a one-time test reflecting historical typical performance of material. In this table, a "batch test" should be interpreted to mean a test made on a production batch of product to establish suitability for a specific application. A "batch of activated carbon" should be interpreted to mean a quantity of material of the same grade, type, and series that has been homogenized to exhibit, within reasonable tolerance, the same performance and physical characteristics and for which the manufacturer can demonstrate by acceptable tests and quality control practices such uniformity.	Complies.
All material in the same batch should be activated, impregnated, and otherwise treated under the same process conditions and procedures in the same process equipment and should be produced under the same manufacturing release and instructions. Material produced in the same charge of batch equipment constitutes a batch; material produced in different charges of the same batch equipment should be included in the same batch only if it can be homogenized as above. The maximum batch size should be 350 ft^3 of activated carbon.	Complies.
If an adsorbent other than impregnated activated carbon is proposed or if the mesh size distribution is different from the specifications in Table 5.1 of ANSI N509-1976, the proposed adsorbent should have demonstrated the capability to perform as well as or better than activated carbon in satisfying the specifications in Table 5.1 of ANSI N509-1976.	Complies.
If impregnated activated carbon is used as the adsorbent, the adsorber system should be designed for an average atmosphere residence time of 0.25 sec per two inches of adsorbent bed.	Complies.
The adsorption unit should be designed for a maximum loading of 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon. No more than 5% of impregnant (50 mg of impregnant per gram of carbon) should be used. The radiation stability of the type of carbon specified should be demonstrated and certified (see Section C.1.b of this guide for the design source term).	~~Complies.~~

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C		System 80+
B.	Component Design ... (etc)	
~~3.10~~ j.	Adsorber cells should be designed, constructed, and tested in accordance with the requirements of Section 5.2 of ANSI N509-1976.	Complies.
~~3.11~~ k.	The design of the adsorber section should consider possible iodine desorption and adsorbent auto-ignition that may result from radioactivity-induced heat in the adsorbent and concomitant temperature rise. Acceptable designs include a low-flow air bleed system, cooling coils, water sprays for the adsorber section, or other cooling mechanisms. Any cooling mechanism should satisfy the single-failure criterion. A low-flow air bleed system should satisfy the single-failure criterion for providing low-humidity (less than 70% relative humidity) cooling air flow.	Complies. Anticipated charcoal bed loading is not sufficient to raise bed temperature to the desorption range.
~~3.12~~ l.	The system fan, its mounting, the ductwork connections should be designed, constructed, and tested in accordance with the requirements of Sections 5.7 and 5.8 of ANSI N509-1976.	Complies.
~~3.13~~ m.	The fan or blower used on the ESF atmosphere cleanup system should be capable of operating under the environmental conditions postulated, including radiation.	Complies.
~~3.14~~ n.	Ductwork should be designed, constructed, and tested in accordance with the provisions of Section 5.10 of ANSI N509-1976.	Complies.
~~3.15~~ o.	Ducts and housings should be laid out with a minimum of ledges, protrusions, and crevices that could collect dust and moisture and that could impede personnel or create a hazard to them in the performance of their work. Straightening vanes should be installed where required to ensure representative air flow measurement and uniform flow distribution through cleanup components.	Complies.
~~3.16~~ p.	Dampers should be designed, constructed, and tested in accordance with the provisions of Section 5.9 of ANSI N509-1976.	Complies.
4.	**Maintenance**	
~~4.1~~ a.	Accessibility of components and maintenance should be considered in the design of ESF atmosphere cleanup systems in accordance with the provisions of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.	Complies.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C	System 80+
4. Maintenance (cont'd.)	
4.2 b. For ease of maintenance, the system design should provide for a minimum of three feet from mounting frame to mounting frame between banks of components. If components are to be replaced, the dimension to be provided should be the maximum length of the component plus a minimum of three feet.	Complies.
4.3 c. The system design should provide for permanent test probes with external connections in accordance with the provisions of Section 4.11 of ANSI N509-1976.	Complies.
4.4 d. Each ESF atmosphere cleanup train should be operated at least 10 hours per month, with the heaters on (if so equipped), in order to reduce the buildup of moisture on the adsorbers and HEPA filters.	Complies.
4.5 e. The cleanup components (i.e., HEPA filters, prefilters, and adsorbers) should not be installed while active construction is still in progress.	Complies.
5. In-Place Testing Criteria	
5.1 a. A visual inspection of the ESF atmosphere cleanup system and all associated components should be made before each in-place airflow distribution test, DOP test, or activated carbon adsorber section leak test in accordance with the provisions of Section 5 of ANSI N510-1975.	Complies.
5.2 b. The airflow distribution to the HEPA filters and iodine adsorbers should be tested in place for uniformity initially and after maintenance affecting the flow distribution. The distribution should be within ±20% of the average flow per unit. The testing should be conducted in accordance with the provisions of Section 8 of "Industrial Ventilation" and Section 8 of ANSI N510-1975.	Complies.
5.3 c. The in-place DOP test for HEPA filters should conform to Section 10 of ANSI N510-1975. HEPA filter sections should be tested in place (1) initially, (2) at least once per 18 months thereafter, and (3) following painting, fire, or chemical release in any ventilation zone communicating with the system to confirm a penetration of less than 0.05% at rated flow. An engineered-safety-feature air filtration system satisfying this condition can be considered to warrant a 99% removal efficiency for particulates in accident dose evaluations. HEPA filters that fail to satisfy this condition would be replaced with filters qualified pursuant to regulatory position C.3.d of this guide. If the HEPA filter bank is entirely or only partially replaced, an in-place DOP test should be conducted.	Complies.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C	System 80+
5. In-Place Testing Criteria (Cont'd.) c. (cont'd) If any welding repairs are necessary on, within, or adjacent to the ducts, housing or mounting frames, the filters and adsorbers should be removed from the housing during such repairs. The repairs should be completed prior to periodic testing, filter inspection, and in-place testing. The use of silicone sealants or any other temporary patching material on filters, housing, mounting frames, or ducts should not be allowed.	complies.
~~5.4~~ d. The activated carbon adsorber section should be leak tested with a gaseous halogenated hydrocarbon refrigerant in accordance with Section 12 of ANSI N510-1975 to ensure that bypass leakage through the adsorber section is less than 0.05%. After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm. Adsorber leak testing should be conducted (1) initially, (2) at least once per 18 months thereafter, (3) following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected, and (4) following painting, fire, or chemical release in any ventilation zone communicating with the system.	Complies.
6. **Laboratory Testing Criteria for Activated Carbon** ~~6.1~~ a. The activated carbon adsorber section of the ESF atmosphere cleanup system should be assigned the decontamination efficiencies given in Table 2 for elemental iodine and organic iodides if the following conditions are met: (1) The adsorber section meets the conditions given in regulatory position C.5.d of this guide. (2) New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976, and (3) Representative samples of used activated carbon pass the laboratory tests given in Table 2. If the activated carbon fails to meet any of the above conditions, it should not be used in engineered-safety-feature adsorbers.	Complies.

Table 9.4-5 Design Comparison to Regulatory Positions of Regulatory Guide 1.52 (Cont'd.)

Regulatory Guide 1.52 Position C	System 80+
6. Laboratory Testing ... (cont'd)	
6.2 b. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section. Each representative sample should be not less than two inches in both length and diameter, and each sample should have the same qualification and batch test characteristics as the system adsorbent. There should be a sufficient number of representative samples located in parallel with the adsorber section to estimate the amount of penetration of the system adsorbent throughout its service life. The design of the samplers should be in accordance with the provisions of Appendix A of ANSI N509-1976. Where the system activated carbon is greater than two inches deep, each representative sampling station should consist of enough two-inch samples in series to equal thickness of the system adsorbent. Once representative samples are removed for laboratory test, their positions in the sampling array should be blocked off.	Complies.
Laboratory tests of representative samples should be conducted, as indicated in Table 2 of this guide, with the test gas flow in the same direction as the flow during service conditions. Similar laboratory tests should be performed on an adsorbent sample before loading into the adsorbers to establish an initial point for comparison of future test results. The activated carbon adsorber section should be replaced with new unused activated carbon meeting the physical property specifications of Table 5.1 of ANSI N509-1976 if (1) testing in accordance with the frequency specified in Footnote c of Table 2 results in a representative sample failing to pass the applicable test in Table 2 or (2) no representative sample is available for testing.	Complies.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1

ALL Two Pages

Regulatory Guide 1.140 Position C	System 80+
1. Environmental Design Criteria	
~~1.1~~ a. The design of each atmosphere cleanup system installed in a normal ventilation exhaust system should be based on the anticipated range of operating parameters of temperature, pressure, relative humidity, and radiation levels.	Complies.
~~1.2~~ b. If the atmosphere cleanup system is located in an area of high radiation during normal plant operation, adequate shielding of components and personnel from the radiation source should be provided.	Complies.
~~1.3~~ c. The operation of any atmosphere cleanup system in a normal ventilation exhaust system should not degrade the expected operation of any engineered-safety-feature system that must operate after a design basis accident.	Complies.
~~1.4~~ d. The design of the atmosphere cleanup system should consider any significant contaminants such as dusts, chemicals, or other particulate matter that could degrade the cleanup system's operation.	Complies.
2. System Design Criteria	
~~2.1~~ a. Atmosphere cleanup systems installed in normal ventilation exhaust systems need not be redundant nor designed to Seismic Category I classification, but should consist of the following sequential components: (1) HEPA filters before adsorbers, (2) iodine adsorbers (impregnated activated carbon or equivalent adsorbent such as metal zeolites), (3) fans, and (4) interspersed ducts, dampers, and related instrumentation. If it is desired to reduce the particulate load on the HEPA filters and extend their service life, the installation of prefilters upstream of the initial HEPA bank is suggested. Consideration should also be given to the installation of a HEPA filter bank downstream of carbon absorbers to retain carbon fines. Heaters or cooling coils used in conjunction with heaters should be used when the humidity is to be controlled before filtration. Whenever an atmosphere cleanup system is designed to remove only particulate matter, a component for iodine adsorption need not be included.	Complies. Heaters or cooling coils are not required. However, heaters are provided.
~~2.2~~ b. To ensure reliable in-place testing, the volumetric air flow rate of a single cleanup train should be limited to approximately 30,000 ft^3/min. If a total system air flow in excess of this rate is required, multiple trains should be used. For ease of maintenance, a filter layout that is three HEPA filters high and ten wide is preferred.	Complies.

Note:

Design requirements of this Regulatory Guide (Revision 1) are applicable to the Non-safety-related Ventilation Exhaust Systems of System 80+.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

Regulatory Guide 1.140 Position C		System 80+
2.	System Design Criteria (Cont'd)	
~~2.3~~ c.	Each atmosphere cleanup system should be instrumented to monitor and alarm pertinent pressure drops and flow rates in accordance with the recommendations of Section 5.6 of ERDA 76-21.	Complies.
~~2.4~~ d.	To maintain the radiation exposure to operating and maintenance personnel as low as is reasonably achievable, atmosphere cleanup systems and components should be designed to control leakage and facilitate maintenance, inspection, and testing in accordance with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."	Complies.
~~2.5~~ e.	Outdoor air intake openings should be equipped with louvers, grills, screen, or similar protective devices to minimize the effects of high winds, rain, snow, ice, trash, and other contaminants on the operation of the system. If the atmosphere surrounding the plant could contain significant environmental contaminants, such as dusts and residues from smoke cleanup systems from adjacent coal burning power plants or industry, the design of the system should consider these contaminants and prevent them from affecting the operation of any atmosphere cleanup system.	Complies.
~~2.6~~ f	Atmosphere cleanup system housings and ductwork, as defined in Section 5.10.8.1 ANSI N509-1976 should be designed to exhibit on test a maximum total leakage rate as defined in Section 4.12 of ANSI N509-1976. Duct and housing leak tests should be performed in accordance with the provisions of Section 6 of ANSI N510-1975.	Complies.
3.	**Component Design Criteria and Qualification Testing**	
~~3.1~~ a	Adsorption units function efficiently at a relative humidity of 70% or less. If the relative humidity of the atmosphere entering the air cleanup system is expected to be greater that 70% during normal reactor operation, heaters or cooling coils used in conjunction with heaters should be designed to reduce the relative humidity of the entering atmosphere to 70% or less. Heaters should be designed, constructed, and tested in accordance with the requirements of Section 5.5 of ANSI N509-1976 exclusive of sizing criteria.	See response to Regulatory Position 2.~~1~~ a. above. Relative Humidity need not be controlled to 70% or less for iodine removal decontamination factors of 95% or less. Penetration tests will be conducted at 95% Relative Humidity instead of 70% Relative Humidity per ASTM D3803-89.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

Regulatory Guide 1.140 Position C	System 80+
3. Component Design... (Cont'd)	
~~3.2~~ b. The HEPA filters should be designed, constructed, and tested in accordance with the requirements of Section 5.1 of ANSI N509-1976. Each HEPA filter should be tested for penetration of dioctyl phthalate (DOP) in accordance with the provisions of MIL-F-51068 and MIL-STD-282.	Complies.
~~3.3~~ c. Filter and adsorber mounting frames should be designed and constructed in accordance with the provisions of Section 5.6.3 of ANSI N509-1976.	Complies.
~~3.4~~ d. Filter and adsorber banks should be arranged in accordance with the recommendations of Section 4.4 of ERDA 76-21.	Complies.
~~3.5~~ e. System filter housings, including floors and doors, and electrical conduits, drains, and piping installed inside filter housings should be designed and constructed in accordance with the provisions of Section 5.6 of ANSI N509-1976.	Complies.
~~3.6~~ f. Ductwork associated with the atmosphere cleanup system should be designed, constructed, and tested in accordance with the provisions of Section 5.10 of ANSI N509-1976.	Complies.
~~3.7~~ g. The adsorber section of the atmosphere cleanup system may contain any adsorbent material demonstrated to remove gaseous iodine (elemental iodine and organic iodides) from air at the required efficiency. Since impregnated activated carbon is commonly used, only this adsorbent is discussed in this guide. Each original or replacement batch of impregnated activated carbon used in the adsorber section should meet the qualification and batch test results summarized in Table 1 of this guide.	Complies.
If an adsorbent other than impregnated activated carbon is proposed or if the mesh size distribution is different from the specifications in Table 1, the proposed adsorbent should have demonstrated the capability to perform as well as or better than activated carbon in satisfying the specifications in Table 1. If impregnated activated carbon is used as the adsorbent, the adsorber system should be designed for an average atmosphere residence time of at least 0.25 sec per 2 inches of adsorbent bed.	Complies.
~~3.8~~ h. Adsorber cells should be designed, constructed, and tested in accordance with the requirements of Section 5.2 of ANSI N509-1976.	Complies.
~~3.9~~ i. The system fan and motor, mounting, and ductwork connections should be designed, constructed, and tested in accordance with the requirements of Sections 5.7 and 5.8 of ANSI N509-1976.	Complies.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

Regulatory Guide 1.140 Position		System 80+
3.	Component Design (cont'd)	
~~3.10~~ j.	The fan and motor used in the atmosphere cleanup system should be capable of operating under the environmental conditions postulated.	Complies.
~~3.11~~ k.	Ducts and housing should be laid out with a minimum of ledges, protrusions, and crevices that could collect dust and moisture and that could impede personnel or create a hazard to them in the performance of their work. Turning vanes or other air flow distribution devices should be installed where required to ensure representative air flow measurement and uniform flow distribution through cleanup components.	Complies.
~~3.12~~ l.	Dampers should be designed, constructed, and tested in accordance with the provisions of Section 5.9 of ANSI N509-1976.	Complies.
~~3.13~~ m.	If prefilters are used in the atmosphere cleanup system, they should be designed, constructed, and tested in accordance with the provisions of Section 5.3 of ANSI N509-1976.	Complies.
4.	**Maintenance**	
~~4.1~~ a.	Accessibility of components and maintenance should be considered in the design of atmosphere cleanup systems in accordance with the provisions of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.	Complies.
~~4.2~~ b.	For ease of inspection and maintenance with minimum danger of damage to the system, its design should provide for a minimum of 3 feet clear access space in each compartment after allowing for the component dimension itself and the maximum length of the component during changeout.	Complies.
~~4.3~~ c.	The system design should provide for permanent test probes with external connections in accordance with the provisions of Section 4.11 of ANSI N509-1976.	Complies.
~~4.4~~ d.	The cleanup components (e.g., HEPA filters and adsorbers) should be installed after construction is completed.	Complies.
5.	**In-Place Testing Criteria**	
~~5.1~~ a.	A visual inspection, in accordance with the provisions of Section 5 of ANSI N510-1975, of the atmosphere cleanup system and all associated components should be made before each in-place airflow distribution test, DOP test, or activated carbon adsorber section leak test.	Complies.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

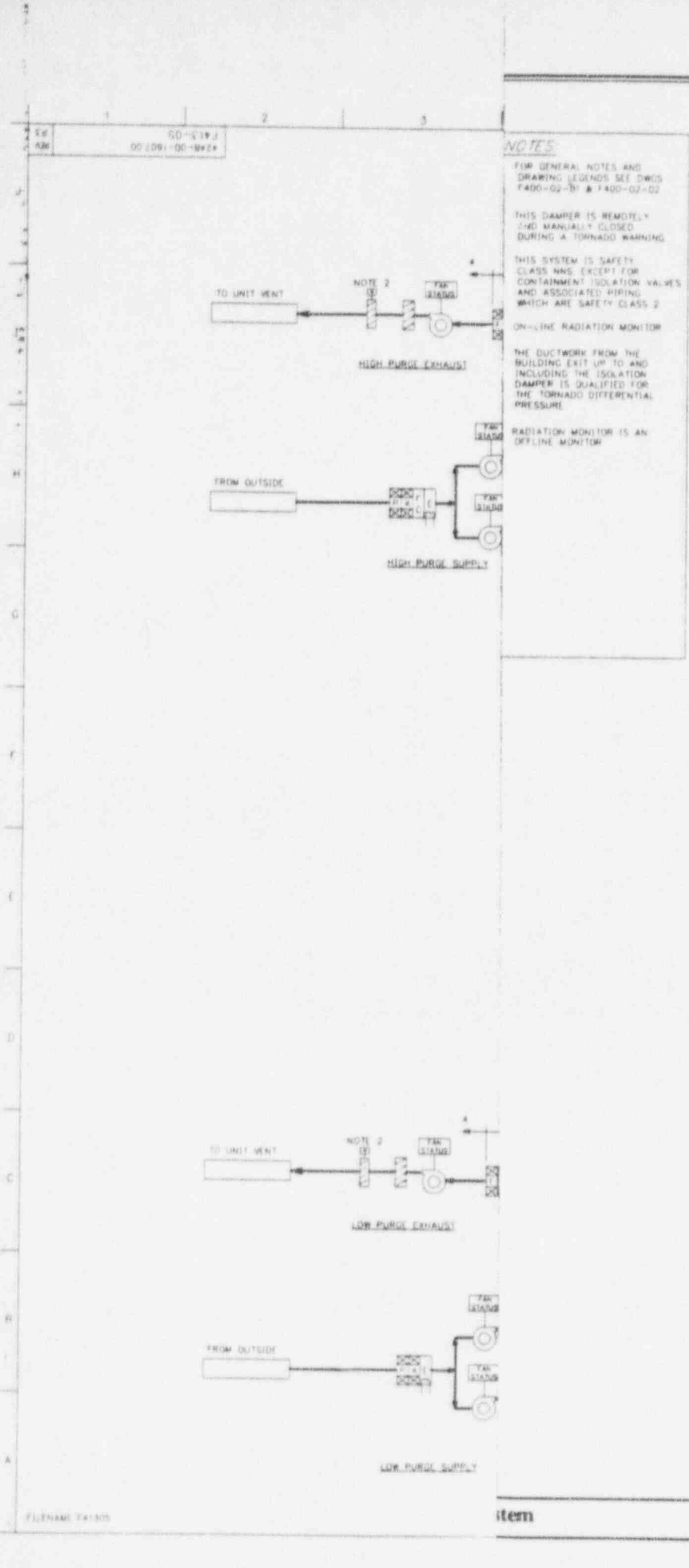
	Regulatory Guide 1.140 Position	System 80+
5.	In-Place Testing (cont'd.)	
~~5.2~~ b.	The airflow distribution to the HEPA filters and iodine adsorbers should be tested in place for uniformity initially and after maintenance affecting the flow distribution. The distribution should be within ±20% of the average flow per unit when tested in accordance with the provisions of Section 9 of "Industrial Ventilation" and Section 8 of ANSI N510-1975.	Complies.
~~5.3~~ c.	The in-place DOP test for HEPA filters should conform to Section 10 of ANSI N510-1975. HEPA filter sections should be tested in place initially and at interval of 18 months thereafter. The HEPA filter bank upstream the adsorber section should also be tested following painting, fire, or chemical release in any ventilation zone communicating with the system in such a manner that the HEPA filters could become adversely affected by the fumes, chemicals, or foreign materials. DOP penetration tests of all HEPA filter banks should confirm a penetration of less than 0.05% at rated flow. A filtration system satisfying this condition can be considered to warrant a 99% removal efficiency for particulates. HEPA filters that fail to satisfy the in-place test criteria should be replaced with filters qualified pursuant to Regulatory Position C.3.b of this guide. If the HEPA filter bank is entirely or only partially replaced, an in-place DOP test should be conducted.	Complies.
	If any welding repairs are necessary on, within, or adjacent to the ducts, housing or mounting frames, the filters and adsorbers should be removed from the housing during such repairs. Those repairs should be completed prior to periodic testing, filter inspection, and in-place testing. The use of silicone sealants or any other temporary patching material on filters, housing, mounting frames, or ducts should not be allowed.	Complies.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

Regulatory Guide 1.140 Position	System 80+
5. In-Place Testing Criteria (Cont'd.) ~~5.4~~ d. The activated carbon adsorber section should be leak tested with a gaseous halogenated hydrocarbon refrigerant in accordance with Section 12 of ANSI N510-1975 to ensure that bypass leakage through the adsorber section is less than 0.05%. After the test is completed, air flow through the unit should be maintained until the residual refrigerant gas in the effluent is less than 0.01 ppm. Adsorber leak testing should be conducted (1) initially, (2) at least once per 18 months thereafter, (3) following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected, and (4) following painting, fire, or chemical release in any ventilation zone communicating with the system in such a manner that the charcoal adsorbers could become adversely affected by the fumes, chemicals, or foreign materials.	Complies.
6. Laboratory Testing Criteria for Activated Carbon ~~6.1~~ a. The activated carbon adsorber section of the ESF atmosphere cleanup system should be assigned the decontamination efficiencies given in Table 2 for elemental iodine and organic iodides if the following conditions are met: (1) The adsorber section meets the conditions given in regulatory position C.5.d of this guide. (2) New activated carbon meets the physical property specifications given in Table 5.1 of ANSI N509-1976, and (3) Representative samples of used activated carbon pass the laboratory tests given in Table 2. If the activated carbon fails to meet any of the above conditions, it should not be used in adsorber units.	Complies.

Table 9.4-6 Design Comparison to Regulatory Positions of Regulatory Guide 1.140, Revision 1 (Cont'd.)

Regulatory Guide 1.140 Position	System 80+
c. Laboratory Testing ... (Cont'd.) ~~6.2~~ b. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section. Each representative sample should be not less than two inches in both length and diameter, and each sample should have the same qualification and batch test characteristics as the system adsorbent. There should be a sufficient number of representative samples located in parallel with the adsorber section to estimate the amount of penetration of the system adsorbent throughout its service life. The design of the samplers should be in accordance with the provisions of Appendix A of ANSI N509-1976. Where the system activated carbon is greater than two inches deep, each representative sampling station should consist of enough two-inch samples in series to equal and thickness of the system adsorbent. Once representative samples are removed for laboratory test, their positions in the sampling array should be blocked off.	Complies.
Laboratory tests of representative samples should be conducted, as indicated in Table 2 of this guide, with the test gas flow in the same direction as the flow during service conditions. Similar laboratory tests should be performed on an adsorbent sample before loading into the adsorbers to establish an initial point for comparison of future test results. The activated carbon adsorber section should be replaced with new unused activated carbon meeting the physical property specifications of Table 1 if (1) testing in accordance with the frequency specified in Footnote c of Table 2 results in a representative sample failing to pass the applicable test in Table 2 or (2) no representative sample is available for testing.	Complies.



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stem	Figure 9.4-6

9.5.2.2.4 Intraplant Sound-Powered Telephone Systems

Intraplant sound-powered telephone systems, independent of the PABX and PA systems, are provided for normal and abnormal/accident conditions. These sound-powered systems include, but are not limited to, the following:

- Maintenance Circuit consists of phone jacks located throughout the plant which can be patched together to establish communications between areas as necessary.

- Refueling Circuit consists of phone jacks located in areas required for refueling operations.

- Emergency Circuit consists of phone jacks connecting specific areas of the plant for the purpose of communication during auxiliary shutdown operations.

9.5.2.2.5 Offsite Communications

[[Normal offsite communications is provided by public telephone lines and the utility private network which is connected to the PABX telephone switch.]][1]

[[Emergency offsite communications, independent of the PABX telephone switch, is provided by public telephone lines and the utility private network lines connected directly to specific telephones located in critical areas of the plant and support facilities.]][1] Emergency telephones are color-coded to distinguish them from the intraplant telephone system. The emergency telephones include, but are not limited to, the following:

- Emergency Notification System (ENS)

  Provides a communications link with the Nuclear Regulatory Commission (NRC).

- Health Physics Network (HPN)

  Provides a communications link with the NRC's health physics personnel.

- Ringdown Phone System

  Provides communications link with local and state agencies.

[[In addition, a security radio system is provided in accordance with 10 CFR 73.55(f) and a crisis management radio system is provided in accordance with the intent of NUREG-0654.]][2]

9.5.2.2.6 System Operation

[[In areas of high noise levels, noise-cancelling devices, sound isolation booths and/or visual alerting are utilized.]][2]

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1 Conceptual Design information; see DCD Introduction Section 3.4.

2 COL information items; see DCD Introduction Section 3.2.

Table 10.3.2-1 Main Steam Supply System Design Data

Component	Parameter
Main Steam Piping:	
Steam flow, lb/hr	17.64×10^6
Number of main steam lines	4
Pipe size, I.D. inches	28
Design pressure, psia	1200
Design temperature, °F	570
Pipe material	carbon steel
Main Steam Isolation Valves:	
Number per main steam line	1
Total number provided	4
Atmospheric Dump Valves:	
Number per main steam line	1
Total number provided	4
Design relieving capacity per valve, 100% open, lb/hr (at 1,000 psia)	950,000
Controllable capacity per valve, lb/hr (at 1,100 psia)	63,000
Number per main steam line	5
Set pressure, psig	
No. 1	1185
No. 2	1220
No. 3	1245
No. 4	1245
No. 5	1245
Orifice size, in^2	
No. 1	16.0
No. 2	16.0
No. 3	16.0
No. 4	16.0
No. 5	16.0
Inlet/outlet size, in/in	
No. 1	6 x 10
No. 2	6 x 10
No. 3	6 x 10
No. 4	6 x 10
No. 5	6 x 10
Relieving capacity, per valve, lb/hr	0.95×10^6
Total relieving capacity (20 valves), lb/hr	19×10^6
Total number provided	20

Insert *

Indent

* Main Steam Safety Valves:

Table 10.3.5-1 Operating Chemistry Limits for Secondary Steam Generator Water

			Power Operation	
Variable	**Wet Layup**	**Startup[9]**	**Normal[1] Specifications**	**Abnormal Limits**
pH (mixed system)[2]	≥9.8	8.5 - 9.2	8.5 - 9.2[6]	
(copper free)	≥9.8	≥9.0	≥9.0[5]	
Cation Conductivity[3]	—	≤2.0 μmhos/cm	≤ 0.8 μmhos/cm	0.8-2.0 μmhos/cm
Silica[7]	—	—	≤ 300 ppb	
Chloride	≤1000 ppb	≤100 ppb	≤ 20 ppb	20-100 ppb
Sodium[4]	≤1000 ppb	≤100 ppb	≤ 20 ppb	20-100 ppb
Sulfate	≤1000 ppb	≤100 ppb	≤ 20 ppb	20-100 ppb
Hydrazine	75 - 200 ppm[10]	—		
N_2 (over pressure)	>5 psig	—		
Dissolved Oxygen	≤100 ppb [8]	≤5 ppb		

Notes:

[1] Normal specifications are those which should be maintained by continuous steam generator blowdown during proper operation of secondary systems.

[2] A mixed system is any secondary system containing copper alloy components.

[3] If the immediate shutdown limit of 7.0 μmhos/cm is exceeded, the unit should be shut down within four hours.

[4] If the immediate shutdown limit of 500 ppb is exceeded, the unit should be shut down within four hours.

[5] In plants where condensate polishers are in operation, the pH of a copper-free system can be controlled to a value of ≥8.8, with action required at < 8.8.

[6] Action required only if experience shows increased copper transport at pH >9.2.

[7] This parameter is used for problem diagnosis.

[8] Oxygen value applies to steam generator fill source.

[9] Startup values shall be met prior to exceeding 5% reactor power.

[10] Limits for wet layup of seven or more days. For outages of less than seven days' duration from cold shutdown to startup, the steam generators should be filled with feedwater containing greater than 5 ppm hydrazine, and a nitrogen overpressure established (>5 psig)

Table 10.3.5-2 Operating Chemistry Limits for Feedwater

Variable	Startup Specifications[1]	Normal Specifications[2]
pH		
a. Mixed system	—	8.8 - 9.2[3]
b. Copper-free system	—	≥ 9.3[4]
Conductivity (Intensified cation)[5]	—	≤ 0.2 μmhos/cm
Hydrazine[6]	≤ 3 x $[O_2]$	Note 7
Dissolved Oxygen	≤ 100 ppb[8]	≤ 5 ppb
Sodium [5]	—	≤ 3 ppb
Iron	—	≤ 20 ppb
Copper[9]	—	≤ 2 ppb

Notes:

[1] Startup values apply when the RCS > 210°F, but reactor power is ≤ 5%

[2] Normal specifications are those which should be maintained during proper operation of secondary system.

[3] Action required only if experience shows increased copper transport at pH > 9.2.

[4] In plants where condensate polishers are in operation, the pH of a copper-free system can be controlled to a value of ≤ 9.0, with action required at $<$ 9.0.

[5] Conductivity and sodium are diagnostic parameters. These values were set as a means of addressing steam purity concerns. It is realized that lower values will be needed to meet blowdown limitations in Table 10.3.5-1. Feedwater sodium values of ≪ 1 ppb are required to meet steam generator water quality. Likewise, cation conductivity values ≪ 0.2 are generally required to meet steam generator water quality.

[6] The hydrazine limit applies to feedwater/condensate downstream of the normal chemical addition point.

[7] Feedwater hydrazine normal operating concentration shall be greater than, or equal to, three times the dissolved oxygen concentration (≤ 3 x $[O_2]$, but no less than 20 ppb.

[8] It may not be possible to control oxygen at this value before turbine steam seals can be established. This value should be met, however, prior to reaching 5% power.

[9] Analysis not required for copper-free systems.

Table 10.3.5-5 Secondary Sampling/Laboratory Analysis Frequencies During Startup and Wet Layup

Item	Sampling Frequency[1]		
	Steam Generator Water		Feedwater
	Startup	Feedwater	Startup
Cation Conductivity	D	--	--
pH	D	Note 2	--
Dissolved O_2	D	Note 2	D
Sodium	D	Note 2	--
Hydrazine	--	Note 2	D[3]
Chloride	D	Note 2	--
Sulfate	D	Note 2	--

[1] Frequencies are defined by:

D = Daily
W = Weekly
DR/W = Daily recording of inline instrument reading with a weekly comparison of the instrument to laboratory results
DR/M = Daily recording of inline instrument reading with a monthly comparison of the instrument to laboratory results

Frequencies should be increased if abnormal conditions are detected.

[2] Analysis every other day until stable, then weekly.
~~Hydrazine analysis during normal operation must be done downstream of the normal chemical addition point.~~

[3] Hydrazine feed rate may be verified during startup if actual sampling is not possible.
~~Analysis of sample passed through filter and cation resin membrane.~~

- FDT - These are high conductivity wastes and variable suspended solids. The waste is collected from floor drains sumps. Additionally, steam generator drains are sent to the FDT.

- LHST - These are laundry wastes, wastes from personnel decontamination stations, and detergent type decontamination solutions.

- CWT - These are laboratory wastes and chemical decontamination solutions.

- NT - These tanks, which are located in the Turbine Building, contain neutralized waste water from the regeneration of the condensate cleanup system polishers.

The Waste Collection Tanks are each sized for the anticipated peak daily input taking into account anticipated operational occurrences but not considering events which might occur less often than once per fuel cycle. The waste collection tank vents are sized adequately to preclude buckling of the tank during drain down, or vacuum breakers will be provided, as necessary.

Waste collection tanks are all equipped with fluid-driven mixers, manways and material addition ports accessed from the top of the tanks. Additionally, all tanks have sloped bottoms to facilitate settled sludge removal. The tanks are all located in lined rooms in which the walls constitute appropriate shielding as well as the seismic containment required by Regulatory Guide 1.143.

Waste collection tanks are all made of stainless steel and are designed for atmospheric pressure plus maximum overflow line back-pressure.

As shown in Figure 11.2-1, oil separators are provided for gravity separation of oil which may contaminate the equipment or floor drain wastes collected. This oil layer is collected in drums and is stored as required in the Radwaste Building for offsite treatment and disposal by a licensed contractor. Oil treatment and disposal are not included in the certified design scope.

11.2.2.2.2 Waste Monitor Tanks

The LWMS subsystem associated with each of the waste monitor tanks are listed below:

Tank(s)	Subsystem
Waste Monitor Tanks (WMT) 1 and 2	High level waste
Waste Monitor Tanks (WMT) 3 and 4	Low level waste
Detergent Sample Tanks (DST) 1 and 2	Laundry and hot shower
Chemical Sample Tanks (CST) 1 and 2	Laundry and hot shower

The WMT, DST, and CST are each sized for the anticipated peak daily input taking into account anticipated operational occurrences but not considering events which might occur less often than once per fuel cycle. The waste ~~collection~~ monitor tank vents are sized adequately to preclude buckling of the tank during drain down, or vacuum breakers will be provided, as necessary.

The waste monitor tanks are equipped with fluid-driven mixers and provisions for recirculation to assure uniform contents for sampling and manways accessed from the top of the tanks.

Waste monitor tanks are all made of stainless steel and are designed for atmospheric pressure plus maximum overflow line back-pressure.

The waste water collected in the neutralization tank from regeneration of the condensate cleanup system polishers is sampled in this tank also. There is no process capability provided in the condensate cleanup system located in the Turbine Building; however, flow can be manually diverted to the Floor Drain Tank in the low level subsystem of the LWMS, as necessary, based on sampling results. The neutralization tank is a process tank and not a waste monitor tank per se.

11.2.2.2.3 Process Pumps

Each waste stream is provided with a centrifugal pump which can be cross connected with another in case of failure. Pumps can be flushed and drained prior to maintenance activity and can be readily replaced with on-site spares if necessary. The wetted parts of the pumps are corrosion resistant in order to minimize the buildup of contamination and prolong their service life.

Table 11.2-6 lists the design parameters for the process pump provided. The regenerant waste stream is provided with these centrifugal pumps.

11.2.2.2.4 Process Filters

The waste process filters use bag type polyethylene filters. The filters are contained in top loading, vertical stainless steel pressure vessels. Inlet flow is forced into the bag filter media from the top, directed down into the bag to minimize process flow bypass. Top loading bag filters provide process flexibility by tailoring filter media micron exclusion characteristics to the specific waste stream and are effective in maintaining occupational exposures as low as reasonably achievable due to their infrequent and simple changeout requirements. Occupational exposure associated with filter changeout is further reduced through the use of remote handling tools.

Bag filter process vessels are skid mounted in groups of 2 or 4 with each skid being individually shielded and movable. Connections, valves and flow paths are provided to allow the filters to be used individually or in series.

Expended bag filters are generally dewatered and placed directly in shielded disposal containers placed in close proximity to the filter skid to minimize radiation exposure associated with manual filter changeout. Dewatering is accomplished in the filter process vessel by purging the filter housing and filter media with process air prior to filter changeout.

11.2.2.2.5 Media Bed Process Vessels

The media bed process vessels are stainless steel pressure vessels with inlet distributors, screened outlets and sluice outlets. The normal use of the process vessels is as an ion-exchange bed or a carbon bed.

Three demineralizer trains are provided. Each train has five vessels and is sized to process the total subsystem flow based on the sources and volumes specified in Table 11.2-2 and Figure 11.2-2 plus allowances for increases in influent flow during selected off-normal operation. Each train is dedicated to a subsystem; i.e., low level waste subsystem, high level waste subsystem, or laundry, hot shower and chemical waste subsystem. The containment cooler subsystem (Section 11.2.2.2.8) is low activity and is, therefore, normally routed to the Industrial Waste Discharge. There is automatic diversion to the ~~collection tank~~; however, subsequent processing by the Low Level Waste Subsystem is manually initiated

FDT if the subject subsystem tank contents are found to be radioactive;

by the operator. Capability is provided to process this stream as liquid waste through any of the other subsystem demineralizer trains.

Access is provided to manually load vessels if appropriate. The normal disposition of fully expended (high differential pressure, high radiation or loss of desired isotopic removal capability) media is sluicing to the Low Activity Spent Resin Tank in the Solid Waste Management System (SWMS) or directly to a disposal container for processing and shipment offsite.

11.2.2.2.6 Provisions for Mobile Equipment

It is anticipated that it may be advantageous to use additional mobile treatment or direct solidification equipment at times. This may be true because of changing waste streams or changing economics of processing, shipping and burial. Adequate space has been allocated for solid waste shipment vehicles, process dewatering, and a permanent solidification system. Piping provisions are made to permit connection of mobile process equipment while using the installed Waste Collection Tanks, process pumps, and Waste Monitor Tanks. Rapid re-alignment of a process flow path can be accomplished using remote operated valves (outside of skid shielding), quick-connect fittings and flexible high pressure industrial hoses.

11.2.2.2.7 Dilution Pumps

A dedicated source of dilution water is necessary to maintain liquid waste effluent concentrations in the environment below 10 CFR 20 concentration limits and 10 CFR 50 Appendix I as low as reasonably achievable offsite dose objectives. The dilution flow is provided by four centrifugal pumps. The pumps are sized such that any two pumps can provide a minimum of 100 CFS dilution flow to facilitate LWMS discharges.

11.2.2.2.8 Containment Cooler Condensate Tank

Two containment cooler condensate tanks (CCCT) are provided. The containment cooler condensate tank discharge will normally be routed to Industrial Waste Discharge since typically this stream has low activity. The capability to process this stream for processing as liquid waste ~~will be~~ provided.

through any of the other subsystem demineralizer trains is

The CCCTs are fabricated of stainless steel.

11.2.2.2.9 Condensate Cleanup System Waste

The radioactive liquid waste water generated during regeneration of the condensate cleanup system polishers is collected in the neutralization tanks located in the Turbine Building. The contents of the neutralization tanks typically require no further processing and are discharged directly to the environment through a single designated discharge point. The neutralization tanks will be sampled prior to release.

Separate piping is provided from the neutralization tanks, which are located in the Turbine Building, to a common plant discharge header. A radiation monitor is provided downstream of the neutralization tank. Upon a receipt of radiation signal above the monitor setpoint, the discharge from the neutralization tanks will be terminated automatically. The operator would then sample the contents of the neutralization tanks and manually divert flow, as necessary based on the sampling results, to the Floor Drain Tank for processing in the low level waste subsystem of the LWMS prior to release to the environment.

Automatic diversion to the FDT occurs if the subject subsystem tank contents are found to be radioactive; however, subsequent processing by the Low Level Waste Subsystem is manually initiated by the operator.

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A dike is provided around the neutralization tanks designed to be of sufficient height to contain maximum expected liquid inventory in these tanks. A dry sump is also provided to collect any spillage from the neutralization and route it to the LWMS for processing. Curbing and floor drains are provided in the regeneration area. This is discussed in Section 10.4.6.

11.2.2.2.10 Laundry and Hot Shower Tank

The laundry, hot shower and chemical waste subsystem is designed to provide the capability to terminate the discharge and divert flow to the collection tanks upon detection of high radiation in the discharge. After sampling the detergent waste sample tank contents, the operator would either recirculate liquid waste to the subsystem collection tank for reprocessing, or manually divert flow to the low level radwaste subsystem for processing, as necessary, based on sampling results. Similarly, the condensate cooler tank discharge would be automatically terminated upon receipt of a high radiation signal. Subsequently, the operator could manually divert flow, as necessary based on sampling results of the subject tank, for further processing in the low level waste subsystem.

11.2.2.3 System Operation

During normal operation, each pair of Waste Collection Tanks will have one available to accept waste inputs and the other will be available for processing if necessary. Since the LWMS operators will have level indication on waste volumes, they will anticipate system collection and processing requirements.

Liquid waste processing and release are separate batched processes. After a Waste Collection Tank has received as much waste as the operators deem appropriate, its inlet valve is closed to permit sampling. Generally, the Collection Tank contents are recirculated and mixed when waste liquids are entering the tank and the tank level exceeds a minimum level. This expedites the batch sampling process and assures the final sample will be representative. Based on the results obtained from the initial collection tank sample, appropriate pre-process chemical addition and processing is performed in the Collection Tank, as necessary. Based on final sample results, the decision will be made to process the tank using the existing process vessel or to provide a more appropriate process. Normally, the effluent for the Chemical Waste Tanks and the Laundry and Hot Shower Tanks are not processed and directly discharged. Because of segregation of inputs, the size of the collection tanks, and the flexibility of the normal ion-exchange process, a revised process should not be necessary. However, if a change is considered necessary, it will be implemented based on status of individual process vessels inferred from previous influent and effluent sampling. Re-alignment of the flow path can be rapidly accomplished using remote operated valves.

The LWMS subsystems collection tanks are sampled prior to processing. Each respective process stream is processed, as appropriate, based on the sampling results. The contents of the neutralization tank and each of the respective LWMS collection tanks, waste monitor or sample tanks are processed, as necessary, to ensure compliance with 10 CFR 20, Appendix B of Sections 20.1001 through 20.2402, Table 2, Column 2 effluent concentration limits. After processing, the waste water is collected in waste monitor or sample tanks where it is sampled prior to discharge to the environment. A radiation monitor is provided downstream of the last possible input of radioactive liquid waste. Upon detection of a radiation signal above the monitor setpoint, the discharge would be automatically terminated. [[The setpoint is determined by the COL Applicant and provided in the Offsite Dose Calculation Manual.]][1] Section 11.5 provides a more detailed discussion regarding the determination of this setpoint. The operator would then sample the appropriate tanks in the LWMS subsystem(s) and the neutralization tank

1 COL information item; see DCD Introduction Section 3.2.

Table 11.2-5 Design Basis Average Annual Liquid Effluent Concentration

Nuclide	Cp(i) (μCi/ml)	EC(i) (μCi/ml)	FEC(i)
Sr-89	1.17E-10	8.00E-06	1.46E-05
Sr-90	5.39E-12	5.00E-07	1.08E-05
Y-90	3.36E-13	7.00E-06	4.80E-08
Y-91	1.02E-10	8.00E-06	1.27E-05
Y-93	7.34E-13	2.00E-05	3.67E-08
Zr-95	3.13E-11	2.00E-05	1.56E-06
Nb-95	5.63E-11	3.00E-05	1.88E-06
Nb-95m	2.24E-13	3.00E-05	7.46E-09
Mo-99	1.06E-09	2.00E-05	5.32E-05
Tc-99m	8.32E-10	1.00E-03	8.32E-07
Ru-103	2.04E-10	3.00E-05	6.79E-06
Rh-103m	1.99E-10	1.00E-02	1.99E-08
Ru-106	3.03E-09	7.00E-05	4.33E-05
Rh-106	2.91E-09	3.00E-06	9.70E-04
Ag-110m	5.49E-11	6.00E-06	9.16E-06
Ag-110	5.37E-12	---	
Sb-124	4.81E-12	7.00E-06	6.87E-07
Te-129m	1.54E-10	7.00E-06	2.20E-05
Te-129	3.66E-12	4.00E-04	9.15E-09
Te-131m	2.68E-11	8.00E-06	3.35E-06
Te-131	5.14E-13	8.00E-05	6.43E-09
~~I-131~~	7.30E-08	1.00E-06	7.30E-02
TE-132	8.80E-10	9.00E-06	9.77E-05
I-132	5.96E-11	1.00E-04	5.96E-07
I-133	2.91E-09	7.00E-06	4.16E-04
Cs-134	5.46E-09	9.00E-07	6.07E-03
I-135	2.65E-10	3.00E-05	8.85E-06
Cs-136	7.53E-10	6.00E-06	1.26E-04
Cs-137	9.86E-09	1.00E-06	9.86E-03

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[[The COL Applicant will provide the operational setpoint for the termination of the gaseous waste management system, high and low purge, and fuel building ventilation system discharges to the environment in the plant-specific offsite dose calculation manual (ODCM).]][1] This setpoint is based on the instantaneous dose rates in unrestricted areas due to the release of radioactive materials released via gaseous effluent. This setpoint ensures that the instantaneous dose rates offsite are less than the following:

Nobles Gases	500 mrem/yr total body;
	3000 mrem/yr skin
Other ~~Gases~~ radionuclides	1500 mrem/yr to any organ

- Accidental releases of radioactive materials from a single component of the GWMS must not result in offsite doses which exceed the guidelines of Branch Technical Position ESTB-11-5.

  Section 11.3.7 provides a discussion of the analysis of a single component failure of the GWMS. The methodology used in this analysis is in accordance with Branch Technical Position (BTP) ESTB-11-5 for the design basis source term. The results of this analysis confirm that the dose consequence of a single failure of a GWMS component is within the dose limits of the Branch Technical Position (500 mrem total body).

- The system must also contribute to meeting the occupational exposure design objective by keeping operation and maintenance exposure ALARA.

  The GWMS is designed in accordance with guidance provided in Regulatory Guide 8.8, ANSI/ANS-55.4, and Regulatory Guide 1.143 and 1.140. This ensures that the GWMS will meet ALARA objectives.

- Protection will be provided to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen in accordance with 10 CFR 50, Appendix A (General Design Criterion 3).

  The GWMS is designed to preclude the buildup of an explosive mixture of hydrogen and oxygen in accordance with the Standard Review Plan, Section 11.3. The charcoal vessels, condenser cooler, piping, analyzer pressure boundary and valves within the GWMS will be designed to withstand a hydrogen explosion (i.e., twenty times normal operating pressure) in accordance with ANSI Standard 55.4. One hydrogen and one oxygen gas analyzer is utilized to monitor H_2 and O_2 gas concentrations in the GWMS. Alarms are provided locally in the Nuclear Annex and in the Main Control Room to alarm on high oxygen concentration.

11.3.1.2 Codes and Standards

The GWMS is designed in accordance with the guidance of Regulatory Guide 1.143 from applicable regulatory positions (C.2, C.4, C.5 and C.6). These include:

- The GWMS is designed and tested in accordance with regulatory position C.2 of Regulatory Guide 1.143.

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1 COL information item; see DCD Introduction Section 3.2.

Table 11.3-4 Estimated Annual Airborne Effluent Releases (Cont'd.)

Nuclide	Waste Gas System	Fuel Handling	Reactor Bldg. Purge[1]	Aux. Bldg. Vent[2]	Turbine Bldg. Vent	Main Condenser Evacuation Exhaust[3]	Total[4] (Ci/yr.)
Cs-136	5.3E-08	0.0E+00	9.8E-06	4.8E-07	---	---	1.0E-05
Cs-137	7.7E-07	2.7E-05	1.7E-05	7.2E-06	---	---	5.2E-05
Ba-140	2.3E-07	0.0E+00	0.0E+00	4.0E-06	---	---	4.2E-06
Ce-141	2.2E-08	4.4E-09	4.0E-06	2.6E-07	---	---	4.3E-06
Total H-3 Released Via Airborne Pathway							1200
C-14[5] Released Via Airborne Pathway							7.3
Ar-41 Released Via Containment Vent							34

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[1] Includes releases from 2 high volume building purges per year at shutdown and 1% operation of low volume purge (i.e., 12.5 SCFM continuous purge.)

[2] Includes Nuclear Annex, Subsphere, and Radwaste Building release contributions.

[3] Includes Blowdown System Flash Tank vent release contributions.

[4] Values of "0.0" appearing anywhere within this table indicates release is less than 1.0 Ci/yr. for noble gases, and 0.0001 Ci/yr. for iodines/particulates.

[5] C-14 not included in offsite dose calculations since it is not addressed in 10 CFR 50, Appendix I objectives. ee

Table 11.4-2 Radioactive Waste Generation at Duke Facilities (1/89 - 6/92) Burial Volumes Generated from Specific Plants

	Center	Burial Volume - Cubic Feet			
Station	Period	Total High[1] Activity Bead	Total Low[2] Activity Bead	Mechanical Filters	Dry Active Waste ~~(Cubic Feet)~~[3]
Oconee	Jan-Jun '89	338.1	1866.6	536.2	5070.6
3 units,	Jul-Dec'89	808.3	1451.8	421.0	4315.4
2538 MWe	Jan-Jun '90	233.0	0.0	279.1	2184.1
	Jul-Dec '90	566.8	2488.8	341.4	8950.8
	Jan-Jun '91	118.0	0.0	27.1	1098.0
	Jul-Dec '91	410.0	0.0	595.0	2288.3
	Jan-Jun '92	436.0	1519.8	331.8	2373.2
	Jul-Dec '92	NA	NA	NA	NA
McGuire	Jan-Jun '89	0.0	3312.9	112.5	2172.7
2 Units,	Jul-Dec '89	698.7	1866.6	114.9	4930.9
2258 MWe	Jan-Jun '90	121.0	1866.6	388.0	2261.0
	Jul-Dec '90	539.4	1866.6	0.0	2161.1
	Jan-Jun '91	166.8	0.0	0.0	500.2
	Jul-Dec '91	83.4	0.0	30.6	813.4
	Jan-Jun '92	1017.4	1313.2	311.4	1889.1
	Jul-Dec '92	NA	NA	NA	NA
Catawba	Jan-Jun '89	602.6	0.0	520.6	4496.5
2 Units,	Jul-Dec '89	205.8	622.2	285.6	948.9
2258 MWe	Jan-Jun '90	0.0	0.0	120.3	1978.1
	Jul-Dec '90	0.0	829.6	0.0	1288.3
	Jan-Jun '91	205.8	1451.8	36.5	859.2
	Jul-Dec '91	446.4	0.0	120.3	969.4
	Jan-Jun '92	1991.4	1769.8	277.1	409.9
	Jul-Dec '92	NA	NA	NA	NA

Center

[1] High Activity Resins include primary cleanup and liquid radwaste processing resins.

[2] Low Activity Resins include secondary side condensate and steam generator blowdown cleanup resins.

[3] Burial volume following volume reduction. Extensive offsite volume reduction assumed for Dry Active Waste consistent with current industry practice. Average volume reduction factor for DAW shipped from Duke System is approximately 18:1.

Table 11.4-3 Radioactive Waste Generation at Duke Facilities (1/89 - 6/92) Average Waste Activity Generated from Specific Plants

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		Average Waste Activity - Curies			
Station	Period	Total High[1] Activity Bead	Total Low[2] Activity Bead	Mechanical Filters	Dry Active Waste (Curies)
Oconee	Jan-Jun '89	125.3	1.0	53.5	37.7
3 Units,	Jul-Dec '89	1171.0	0.7	39.1	27.7
2538 MWe	Jan-Jun '90	768.2	0.0	20.2	14.4
	Jul-Dec '90	933.2	1.5	49.0	1.1
	Jan-Jun '91	89.9	0.0	0.0	3.6
	Jul-Dec '91	265.4	0.0	40.7	4.6
	Jan-Jun '92	719.0	0.4	19.1	6.3
	Jul-Dec '92	NA	NA	NA	NA
McGuire	Jan-Jun '89	0.0	28.9	0.4	1.8
2 Units,	Jul-Dec '89	559.4	0.0	11.6	30.2
2258 MWe	Jan-Jun '90	111.8	0.0	35.0	4.6
	Jul-Dec '90	880.7	0.0	0.0	22.2
	Jan-Jun '91	377.9	0.0	0.0	1.4
	Jul-Dec '91	209.9	0.0	31.5	3.1
	Jan-Jun '92	101.6	0.1	25.5	20.9
	Jul-Dec '92	NA	NA	NA	NA
Catawba	Jan-Jun '89	106.7	0.0	78.3	8.4
2 Units,	Jul-Dec '89	4.0	0.0	115.1	4.2
2258 MWe	Jan-Jun '90	0.0	0.0	6.7	8.2
	Jul-Dec '90	0.0	0.0	0.0	0.0
	Jan-Jun '91	2.4	0.0	27.2	3.4
	Jul-Dec '91	257.0	0.0	39.9	5.5
	Jan-Jun '92	621.2	0.0	10.4	54.2
	Jul-Dec '92	NA	NA	NA	NA

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[1] High Activity Resins include primary cleanup and liquid radwaste processing resins.

[2] Low Activity Resins include secondary side condensate and steam generator blowdown cleanup resins.

Table 11.4-5 System 80+ Solid Waste Source Term for Normal Operation Estimated Average Radionuclide Concentrations

	Radionuclide Concentration -			
Nuclide	**High Activity Resin**	**Low Activity Resin**	**Mechanical Filters**	**Dry Active Waste (μCi/gm)[1]**
H-3	8.4E-03	8.7E-04	1.0E-02	1.8E-03
C-14	5.8E-02	3.3E-05	7.8E-02	9.1E-04
Cr-51	1.8E-01	1.4E-05	1.4E+00	6.2E-03
Mn-54	2.1E+00	4.8E-05	4.4E-01	3.1E-02
Fe-55	6.3E+00	7.9E-04	7.9E+00	8.2E-01
Co-57	1.8E-02	4.7E-07	6.0E-02	8.9E-04
Co-58	9.2E+00	8.3E-05	1.4E+01	8.7E-02
Fe-59	4.7E-02	1.8E-06	1.0E-01	1.6E-03
Ni-59	2.5E-02	4.4E-05	5.7E-02	1.5E-03
Co-60	7.3E+00	5.0E-05	3.2E+00	1.6E-01
Ni-63	5.7E+00	2.8E-03	9.4E+00	1.2E-01
Zn-65	3.5E-02	1.6E-06	5.3E-02	1.7E-03
Sr-89	1.7E-02	4.2E-06	2.1E-01	4.0E-04
Sr-90	9.4E-02	4.5E-04	5.2E-02	1.0E-03
Nb-94	7.7E-03	2.4E-05	2.0E-02	2.4E-03
Nb-95	4.0E-02	1.4E-05	1.7E+00	2.9E-03
Zr-95	3.5E-02	6.3E-06	5.1E-01	1.3E-03
Tc-99	1.1E-02	7.3E-05	9.4E-05	8.1E-04
Ru-103	3.1E-02	2.3E-06	2.6E-01	7.1E-04
Ru/Rh-106	4.4E-01	4.0E-05	4.2E+00	3.2E-03
Ag-108m	1.7E-02	1.2E-06	2.7E-02	4.0E-04
Ag-110m	6.6E-02	8.5E-07	4.5E+00	9.7E-04
Sb-124	2.2E-02	1.8E-06	5.9E-02	4.2E-04
Sb-125	1.7E-01	3.6E-05	4.3E-01	3.4E-03
I-129	1.7E-03	5.3E-06	4.9E-04	6.0E-05
I-131	2.6E-01	9.7E-05	3.0E+00	1.5E-03
Cs-134	1.7E+01	8.6E-04	2.1E-01	1.7E-02
Cs-137	1.9E+01	1.7E-03	2.6E-01	4.9E-02
Ba/La-140	2.2E-01	2.4E-05	5.3E-01	1.3E-03
Ce-141	5.7E-04	3.0E-06	1.8E-01	1.0E-04
Ce/Pr-144	1.2E-01	1.8E-05	1.4E+00	1.1E-03
Np-237/Pu-242	1.6E-05	1.2E-07	1.5E-05	6.2E-06
Pu-238	1.7E-04	1.8E-05	1.6E-02	4.1E-05
Pu-239/240	1.0E-04	4.6E-06	6.1E-03	3.6E-05
Pu-241	8.1E-02	1.9E-04	8.6E-01	2.6E-03
Am-241	4.8E-05	5.0E-07	2.3E-03	1.5E-05
Cm-242	1.7E-04	8.5E-07	1.6E-02	1.6E-05
Am-243	1.3E-06	8.4E-08	0.0	7.8E-07
Cm-243/244	6.1E-05	5.0E-07	5.1E-03	9.4E-06

[1] Based on Duke Power facility averages for January '89 through June '92.

- The System 80+ design of the Process and Effluent Radiological Monitoring and Sampling Systems provides instrumentation to measure, record, and readout in the Main Control room, as well as control releases of radioactive materials in plant process systems and effluent streams. This system is designed to provide for continuous sampling and monitoring of radioactive iodine and particulate, as well as the capability to take grab samples in gaseous process and/or effluent streams in all potential accident release points.

  A particulate/iodine fixed filter cartridge is provided for all plant ventilation systems, with the exception of the nuclear annex and radwaste building ventilation system, which have their own particulate and iodine monitoring systems. Except for the turbine building exhaust, containment purge, the main condenser evacuation system, the Nuclear Island ventilation systems, and the Gaseous Waste Management System, exhausts discharge through the unit vent. Provisions for taking grab samples are provided as specified in Table 11.5-6. Additional discussion regarding sampling capabilities for gaseous process and effluent streams is addressed in Section 11.5.2.2.

  A fixed iodine absorption filter and detector assembly, as well as a moving filter and detector assembly are provided for the unit vent monitor as discussed in Section 11.5.1.2.3.1. The ventilation systems are provided with a fixed iodine absorption filter and detector assembly only, with the exception of the Nuclear Annex and Radwaste Building ventilation systems which are provided with its own particulate and iodine detection systems discussed in Section 11.5.1.2.4.

  The capability for taking grab samples from the unit vent and ventilation system exhausts are provided, as specified in Table 11.5-6 at the respective radiation monitor locations. [[These grab samples are taken for analysis, at a frequency established by the COL Applicant]][1], onsite to the primary chemistry lab and counting room during normal operating and post-accident conditions.

  Continuous sampling of all potential post-accident release points is provided by the continuous unit vent sampler. This sampler contains a fixed filter particulate and iodine cartridge which receives continuous flow of sample air from the unit vent duct. This sample cartridge is routinely replaced and taken for detailed onsite laboratory analysis in the primary chemistry laboratory and counting room, where precise assessment of releases is performed for the period during which the cartridge collected the sample. [[The frequency at which the cartridge is replaced and analyzed is determined by the COL Applicant and specified in the operations and maintenance manual.]][1] The sampler is designed to be used during normal operation and post-accident conditions to meet the sampling requirements specified in 10 CFR 50.34 (f) (2) (xvii) and NUREG-0737, Attachment 2, Section II.F.1.

  The Process and Effluent Monitoring and Sampling System is designed with a continuous control room interface via the DPS and DIAS systems. Primary indication of radiation levels and status of alarms for post-accident and non-post-accident radiation monitors are processed through the DPS and DIAS systems.

[[The COL applicant will demonstrate conformance with 10 CFR 50 Appendix I, ANSI N13.1, RG 1.21 and RG 4.15.]][1]

The RMS monitors normal and potential paths for release of radioactive materials to provide continuous indication and recording of gaseous and liquid radioactivity levels leaving the plant. As a minimum,

1 COL information item; see DCD Introduction Section 3.2.

likely to perform maintenance or surveillance activities. A particulate/iodine fixed filter cartridge is included in the inlet sample tubing to this monitor for collecting periodic grab samples. Under post-accident conditions, this monitor can be used as a supplement to Regulatory Guide 1.97 monitors to measure activity from expected containment leakage or from an unexpected breach in containment.

- Reactor Building Subsphere Ventilation Monitor

  Each division is continuously monitored by an off-line monitor. These monitors continuously sample the exhaust from both divisions of the Reactor Building Subsphere Ventilation System. Sample points are upstream of the exhaust filters and downstream of the last entry point to the exhaust subsystem. Detection of activity is indicative of equipment failure or leakage in the subsphere areas. A particulate/iodine fixed filter cartridge is included in the sample inlet for grab sample collection.

- Portable Airborne Monitor

  This monitor includes detector channels for particulate, iodine, and gaseous activity. The samplers, detectors, auxiliary equipment, and associated electronics are assembled on a mobile cart. This monitor can be moved to areas where work or surveillance activities are at an unusual risk of airborne exposure. Design and operation of this monitor allows for the transfer of the particulate sample filters and iodine sample cartridges to the counting room for further sample analysis. The Portable Airborne Monitor meets the equipment requirements stated in Section III.D.3.3 of NUREG-0737. This includes requirements on sample media, purging, and calibration.

- Emergency Operations Facility (EOF) Ventilation Monitor

  ~~While it is in use during an emergency, the EOF is continuously monitored in the same manner as the TSC described above~~. While it is in use during an emergency, air entering the EOF is continuously monitored by a shielded off-line gaseous activity detector and returned to the ventilation duct downstream of the intake. If the gaseous activity exceeds a preset limit, an alarm is actuated in the EOF.

11.5.1.2.5 Area Monitors

The Area Radiation Monitoring System monitors the radiation levels in selected areas throughout the plant. Most area monitors are designed to provide normal operation indication of unusual radiological events in order to warn operators and station personnel. Some area monitors are designed for post-accident indication for areas where access for maintenance to equipment important to safety may be necessary. These post-accident monitors are designed to the standards required by Regulatory Guide 1.97. Area radiation monitors will have local visual and audible alarms. High noise areas may have additional visual indication provided if needed to insure prompt recognition by nearby personnel of high radiation conditions. One exception would be the Control Room Area Monitor which will use the existing Main Control Room indications in order not to create a nuisance or distraction due to a spurious alarm. A list of area radiation monitors and their ranges is presented in Table 11.5-4. Area monitor locations are provided in Table 12.3-5.

are lines to detect radioactivity resulting from a steam generator tube (SGTR) rupture. Alarms located in the Main Control Room to alert the operator when these monitors detect specified primary to secondary leakage.

11.5.1.3 Calibration and Maintenance

Commercially available equipment with industry proven technology is incorporated into the design of the Radiation Monitoring System. Monitoring equipment is factory tested and calibrated with provisions made for periodic field calibrations to verify proper detector response. Factory calibration includes isotopic calibration using an adequate number of isotopes to accurately determine the response of the equipment. The accuracy of these calibrations can be traced to the National Institute of Standards and Technology. Secondary calibration sources and decay curves are supplied with the equipment.

Radiation Monitoring System equipment is checked and inspected on a periodic basis. Setpoint checks are performed on a monthly basis with detector calibrations performed once per refueling cycle. Detectors are also calibrated if an inadequate checksource response indicates a problem or following any other equipment maintenance that could affect the accuracy of the instrument indication.

11.5.1.4 Administrative and Procedural Controls

In accordance with the COL applicant

[[~~As specified in~~ Position C of Regulatory Guide 4.15, written procedures will be prepared, reviewed, and approved for sample collection, preparation, and analysis.]][1] Procedures will also exist for the use of radioactivity reference standards, detector calibration and checks of the radiation monitor systems, and for reduction, evaluation and reporting of data. The accuracy of sample flow rate devices will be determined on a regularly scheduled basis. Adjustments to the instrumentation will be made as needed to bring performance into specified limits. The frequency of these calibrations will be specified and results will be recorded. Also, collection efficiencies of the samplers used will be documented.

11.5.2 Process and Effluent Radiological Sampling Program

11.5.2.1 Program Overview

Periodic sampling is performed to supplement the function of the process and effluent radiation monitors. The sampling programs will be designed in accordance with Regulatory Guide 1.21 and the sampling requirements defined in the Technical Specifications.

All continuous effluents that are potentially radioactive are periodically sampled and analyzed. All stored wastes are sampled and samples analyzed before release of wastes to the environment. Comparisons will be made between gross radioactivity measurements of continuous monitors and analyses of specific radionuclides as required by Regulatory Guide 1.21.

Special provisions are made for post-accident sampling of effluent pathways in accordance with NUREG-0737, 10 CFR 50, and Regulatory Guide 1.97 requirements.

[1] COL information item; see DCD Introduction Section 3.2.

Also, sampling of some systems is necessary because of required actions in the Technical Specifications when certain continuous monitors are out of service. [[The Technical Specification actions specify the frequency of sampling and any other special requirements which apply to the sampling procedure developed by the COL Applicant.]][1]

11.5.2.3 Expected Composition and Concentrations

The specific radionuclide compositions will vary for each batch release from the containment or the liquid radioactive waste management system and fluctuate somewhat from day to day from the unit vent continuous releases due to variations in plant operating conditions. The estimated radioactive releases for liquid effluents are given in Table 11.2-1. The estimated annual airborne effluent releases are given in Table 11.3-4.

11.5.2.4 Sampling Equipment and Procedures

Samples are collected by plant technicians and analyzed and measured in the counting room in accordance with station operation procedures concerning the release of radioactive waste. The frequency of sampling is in accordance with Regulatory Guide 1.21 guidelines. Station sampling procedures will establish methods of sampling for each sampling location to assure that representative samples are taken and that these methods will be consistent for all personnel performing the sampling. Tables 11.5-6 and 11.5-7 list the process systems, sampling capabilities, sampling provisions, and the approximate location that samples are taken from gaseous and liquid process and effluent streams, respectively. [[The COL Applicant will determine the sampling frequency, purpose, sensitivity, and type of analysis and provide them in the operations and maintenance manual.]][1]

The liquid contents of a tank being sampled are recirculated prior to taking the sample to ensure thorough mixing of sediments and particulate solids in the tank. All sample connections are located in a free flowing stream or in a location where a representative sample may be taken. The sample lines are purged for an adequate period of time before the sample is taken to ensure that the sample is representative.

Effluent ventilation ducts are sampled isokinetically in accordance with ANSI N13.1 for radioactive gases, particulates, and iodines. Iodine samples are collected using special iodine filtering cartridges and taken to the counting room for analysis. Particulate sampling utilizes fixed paper filters for laboratory analysis. Gas sampling utilizes special gas collection canisters which allow easy connection and disconnection from sample taps for transport to the counting room.

11.5.2.5 Analytical Procedures and Sensitivity

Samples of process and effluent gases and liquids are analyzed in the counting room in accordance with station procedures and Regulatory Guide 1.21. Analytical procedures used are based on methodology utilized in general practice in the nuclear industry or in applicable standards and the accuracy and precision of the results are standardized with central or outside laboratories using radioactivity standards traceable to the National ~~Bureau of Standards~~. Laboratory equipment is provided for the counting room to perform gross beta counting, gross alpha counting, gamma spectrometry, liquid scintillation counting, and radiochemical separations. |

Institute for Science and Technology.

1 COL information item; see DCD Introduction Section 3.2.

Table 11.5-1 Gaseous Process and Effluent Monitors

Bottom-Justify in CELL

Monitor (-channel)	Detector Type[1]	Typical Sensitivity (μCi/cc)	Typical Range (μCi/cc)	Power Supply	Seismic Category	Automatic Function[2]	Configuration Location
Waste Gas - Gas - High Gas	 Beta G-M	 1E-6 (Xe-1[illegible])	 1E-6 - 1E+0 1E-2 - 1E+4	 Non-1E Non-1E	 None None	 Isolate Discharge Isolate Discharge	Online - downstream of charcoal beds
Unit Vent - Particulate - Iodine - Gas - High Gas	 Beta Gamma/SCA Beta SS	 7E-12 (Cs-137) 3E-11 (I-131) 5E-7 (Xe-133) 1E-3 (Xe-133)	 1E-11 - 1E-5 1E-12 - 1E-6 1E-07 - 1E-1 1E-03 - 1E+3	 Non-1E Non-1E Non-1E Non-1E	 None None None None	 None None None None	Offline- downstream of the last possible addition of gas
Unit Vent Post-Accident - High Gas	Ion	3E+0 (Xe-133)	1E+2 - 1E+5	Non-1E	None	None	Inline
Containment High Purge Exhaust - Gas	G-M	2 Times Background	1E-6 - 1E-2	Non-1E	None	Isolate Purge	Online
Containment Low Purge Exhaust - Gas	G-M	2 Times Background	1E-6 - 1E-2	Non-1E	None	Isolate Purge	Online
Main Condenser Evacuation System - Gas	Beta	5E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None	Online - downstream of condenser vacuum pumps

[1] "Beta" = Beta Scintillation Detector; "SS" = Solid State Detector
"Gamma" = Gamma Scintillation Detector; "Ion" = Ion Chamber Detector
"G-M" = Geiger-Mueller Tube; "SCA" = Single Channel Analyzer
(Note: Other types of high range detectors may be substituted for the Solid State Detector.)

[2] Automatic Functions for Gas Monitors are described in Section 11.5.1.2.3.1.

Table 11.5-3 Airborne Radiation Monitors (Cont'd.)

Monitor (-channel)	Detector Type[1]	Typical Sensitivity (μCi/cc)	Typical Range (μCi/cc)	Power Supply	Seismic Category	Automatic Function[2]	Configuration/ Location
Control Room Air Intake (4 monitors, 2 monitors/intake) - Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	1E	I	Isolate Most Contaminated Intake	Offline - downstream of intake, upstream of filter inlet
Reactor Building Annulus - Gas	Beta	5E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None	Offline - exhaust duct upstream of filter inlet
Reactor Building Subsphere Ventilation (2) - Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None	Offline - exhaust duct upstream of filter inlet
Portable Airborne - Particulate - Iodine - Gas	Beta Gamma/SCA Beta	5E-09 (Cs-137) 5E-10 (I-131) 1E-6 (Xe-133)	5E-09 - 5E-4 5E-10 - 5E-5 1E-06 - 1E-1	Non-1E Non-1E Non-1E	None None None	None None None	Portable
Emergency Operations Facility Ventilation - Gas	Beta	3E-7 (Xe-133)	1E-07 - 1E-1	Non-1E	None	None	Offline - upstream of filter inlet, downstream of intake

delete extra line

[1] "Beta" = Beta Scintillation Detector
"Gamma" = Gamma Scintillation Detector
"SCA" = Single ~~Chemical~~ Channel Analyzer

[2] Automatic Functions for ~~Liquid~~ Airborne Radiation Monitors are described in Section 11.5.1.2.3.4.

Table 12.4-1 PWR Reference Plant Data

	Oconee 3 Unit	McGuire 2 Unit	Catawba 2 Unit
Total Exposure (man-rem)	~~122~~ 684	620[1]	334
Average Per Unit (man-rem)	~~233~~ 228	310[1]	167
Number of Refueling Outage	3	1	1
Refueling Exposure (% of Total)	77	67	67
Breakdown by Task (% of Total)			
Routine Operation and Maintenance	21	19	22
Steam Generator Inspection & Maintenance	20	36[1]	22
Reactor Vessel Head Inspection & Maintenance	19	5	8
Valve Maintenance	11	18	21
General Entry & Surveillance	8	5	11
Nuclear Station Modifications	5	6	1
In-service Inspections	5	4	5
Reactor Coolant Pumps	3	3	5
Decontamination	8	4	5

[1] Impacted by Abnormal Occurrence, i.e., Steam Generator Tube Rupture.

3.2.3 Low-Low Lube Oil Pressure

3.2.4 Generator Voltage-Controlled Overcurrent

3.2.5 Low Pressure Turbo Oil

3.2.6 Low Pressure Lube Oil

3.2.7 High Pressure Crankcase

3.2.8 High Temperature Bearings

3.2.9 High Temperature Lube Oil Out

3.2.10 3.2.10 High - High Temperature Jacket Water

"Hi-Hi" for Trip ? 3.2.11 High Vibration

3.3 Demonstrate that the following parameters are correctly monitored in the Control Room and at the local panel:

3.3.1 Lube Oil Temperature and Pressures

3.3.2 Bearing Temperatures

3.3.3 Cooling Water Temperatures and Pressures

3.3.4 Speed

3.3.5 Starting Air Pressure

3.4 Demonstrate the operation of the following status indications:

3.4.1 Cooling water not available

3.4.2 Diesel Generator breaker racked out

3.4.3 Diesel Generator overspeed

3.4.4 Loss of control power

3.4.5 Generator fault

3.4.6 Low air and oil pressure

3.4.7 Maintenance mode

3.5 Demonstrate 35 consecutive starts capability.

3.6 Demonstrate full load capability.

ITAAC and will be similar to the methods in the Section 2.0 ITAAC for comparable/similar design characteristics.

Selection Criteria - The selection criteria listed in Section 14.3.2.1 were used to guide selection of interface requirements defined in Section 4.0 of the CDM (or in the Section 2.0 entries referenced from Section 4.0). The intent is that the interface requirements in the CDM define key, safety-significant design attributes and performance characteristics of the site-specific, out-of-scope portion of the plant which must be provided in order for the certified portions of the System 80+ standard plant to comply with the design commitments in the CDM. It is an objective of this section that it address interfaces between in-scope and out-of-scope portions of the plant that are unique to the System 80+ standard plant design; it is not intended that it be a comprehensive listing of design requirements applicable to the out-of-scope portions of the plant. A discussion of the design feature of out-of-scope portions of the plant will be provided for NRC review when the COL applicant submits a site-specific safety analysis report.

Selection Methodology - The interface requirements included in the CDM were selected from the interface requirements listed in the ADM for fully or partially out-of-scope systems. For example, Section 8.2 defines interface requirements for the Offsite Power Systems. These sections and similar interface requirement sections for other systems were reviewed, and CDM Section 4.0 entries selected using the criteria discussed above.

14.3.5 CDM Section 5.0: Site Parameters

This section of the CDM defines the site parameters which were used as a basis for the design defined in the System 80+ standard plant design certification application. These entries respond to the 10 CFR 52.47(a) (1) (iii) requirement that the design certification documentation include site parameter information. The plant must be designed and built based on the parametric information in Section 5.0. Furthermore, it is intended that applicants referencing the System 80+ standard plant design certification demonstrate that these parameters for the selected site are within the certification envelope or demonstrate that those site characteristics not bounded by the site parameters in Section 5.0 do not invalidate the certified design commitments in Sections 1.0, 2.0, 3.0, and 4.0 of the CDM.

Site-specific external threats that relate to the acceptability of the design (and not to the acceptability of the site) are not considered site parameters and are addressed as interface requirements in the appropriate system entry (e.g. toxic gas).

Section 5.0 of the CDM does not include any ITAAC and is limited to defining site parameters. This is an appropriate approach because compliance of the site with these parameters must be demonstrated by a COL applicant prior to issuance of the license.

Selection Criteria - Chapter 2, Table 2.0-1 provides the envelope of site design parameters used for the System 80+ standard plant design. The corresponding CDM Section 5.0 is based on using Table 2.0-1 in its entirety except as modified to meet the CDM content criteria previously discussed. For example, references in this table to specific Regulatory Guides have been deleted from the CDM table because of the guideline that the CDM does not contain direct references to codes and standards. Section 5 is limited to a tabular entry; no supporting text material is required.

14.3.6 Elements of Design Material Incorporated into the Certified Design Material

Tables 14.3-1 through 14.3-7 summarize the design material that has been incorporated into the CDM in the areas of 1) Design Bases Accident Analysis, 2) Probabilistic Risk Assessment, 3) Shutdown Risk,

4) Severe Accident Analysis, 5) Flood Protection, 6) Fire Protection, and 7) Anticipated Transients Without Scram (ATWS). PRA assumptions incorporated into these tables encompass elements of the system design and assumptions that were expressly included in Tier 1 due to their importance. Both types of PRA assumptions are included for completeness, but are not distinguished in the tables. CDM falling outside of the seven subject areas are intentionally not incorporated in these tables. However, the referenced ADM sections may contain more information than just that encompassed by these seven subject areas. Each table may also include design information (certified or non-certified) that is not directly related to the particular subject area. Further, the tables are not intended to include all system-specific CDM information that is provided in the ADM system descriptions.

Table 14.3-1 Design Basis Accident Analysis

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Paragraph	Assumption/Parameter Description	Value
Table 5.4.13-1	The primary safety valves pass the minimum flow rate of 525,000 1bm/hr-valve (saturated steam at 2575 psia).	
Table 5.4.13-1	Pressurizer Safety Valve: Set Pressure	2500 ±1% psia
Table 5.4.13-1	Pressurizer Safety Valve: Capacity at accumulation Pressure, each Valve	525,000 lb/hr minimum
5.4.13.2	A total MSSV capacity of $19x10^6$ lb/hr is required to maintain the peak secondary pressure below 110% of design.	
Table 6.2.1-3	Containment Shell: Containment Atmosphere Design Basis Peak Pressure	53 psia
Table 6.2.1-18	In-Containment Refueling Water Storage Tank: IRWST Water Volume	495,000 gal minimum
Table 6.2.1-19	Runout Flow Per Safety Injection Train	1232 gpm maximum
Table 6.2.1-19	Runout Flow Per Safety Injection Train	980 gpm minimum
Table 6.2.1-19	Containment Spray Pump: Flowrate Per Pump	6500 gpm maximum
— " —	Containment Spray Pump: Flowrate Per Pump	5000 gpm minimum
— " —	Containment Spray Heat Exchanger: Number of Heat Exchangers	2
— " —	Containment Spray Heat Exchanger: Shell Side Flow	8000 gpm minimum
Table 6.2.1-20	Delay Time from CSAS to Spray Delivery	68 sec maximum
6.2.1.3.3	Main Feed Water Isolation Valve: MFIV Design Closure Time	5.0 sec maximum
6.2.1.4	Main Steam Isolation Valve: MSIV Design Closure Time	5.0 sec maximum
6.2.3.2	Annulus Space: Negative Pressure	-0.25 in (water gauge) minimum

Table 14.3-1 Design Basis Accident Analysis (Cont'd.)

Paragraph	Assumption/Parameter Description	Value
6.2.6.1	The integrated containment leakage rate is less than 0.5% volume per day.	
6.2.6.1	Containment Vessel: Leak Rate	0.5 % volume/day maximum
6.3.3.2.2 [1]	The Engineered Safety Features System (ESFAS) sends a Safety Injection Actuation Signal (SIAS) to start the SIS pumps and open the SIS valves following a LOCA or transient. The SIAS is generated on low pressurizer pressure or high containment pressure.	
6.3.3.2.2	The SIS consists of four safety injection trains, each consisting of a safety injection pump and a safety injection tank.	
6.3.3.2.2	Diesel generators will provide power on LOCA.	
6.3.3.2.2	There are four direct vessel injection points.	
6.3.3.3.2	Delay Time for SI Flow to Reactor Vessel After SIAS	40 sec maximum
Table 6.3.3.4-1	Emergency Feedwater Storage Tank: Emergency Feedwater Storage Tank Capacity	350000 gal/tank minimum
6.3.3.4.2	SITs can be vented or isolated.	
6.3.3.4.4	Alignment of SIS for hot and cold injection is possible.	
Table 15.0-3	Reactor Vessel: Coolant Flow Rate (% of 445,600 gpm)	95% minimum
15.1.2.1	Steam Generator: Maximum Auxiliary Feedwater Flow to Each Steam Generator	800 gpm maximum
Table 15.1.5-10	Main Steam Line: Blowdown Area for Each Steam Line	1.283 sq ft maximum
Table 15.1.5-11	Core: 100% Core Power	3914 MWt
Table 15.1.5-12	Atmospheric Dispersion Factor, 0-2 Hrs at EAB for SLBFPD and SLBZPLOPD Events	1.0×10^{-3} sec/m^3
Table 15.1.5-12	Atmospheric Dispersion Factor, 0-8 Hr at LPZ for SLBFPD and SLBZPLOPD Events	1.35×10^{-4} sec/m^3
Table 15.2.3-1	Main Steam Safety Valve: Main Steam Safety Valves - Open	1212 psia

Table 14.3-1 Design Basis Accident Analysis (Cont'd.)

Paragraph	Assumption/Parameter Description	Value
Table 15.2.8-2	Emergency Feedwater Pump: Emergency Feedwater Flow Initiated to the Intact Steam Generators	500 gpm
Table 15.2.8-2	Main Steam Safety Valve: Main Steam Safety Valves - Open	1212 psia
Table 15.2.8-3	Dispersion Data (MDNBR) 2 hr EAB	1.0×10^{-3} sec/m^3
— " —	Dispersion Data (Over Pressure) 2 hr EAB	1.0×10^{-3} sec/m^3
— " —	Dispersion Data (MDNBR) 8 hr LPZ	1.35×10^{-4} sec/m^3
— " —	Dispersion Data (Over Pressure) 8 hr LPZ	1.35×10^{-4} sec/m^3
Table 15.3.1-1	Main Steam Safety Valve: MSSV Opening Pressure Setpoint	1212 psia maximum
15.3.1.3	Each of the EFW pumps can provide 100% of the required EFW flow.	
Table 15.3.3-1	Emergency feedwater is assumed to be automatically actuated on Steam Generator Low Level EFAS.	
— " —	An isolation valve (block valve) is located upstream of the ADV. The block valve can be closed manually from the control room.	
— " —	Main Steam Safety Valve: MSSV Opening Pressure Setpoint	1212 psia maximum
15.3.3.1	The ADVs are manually operated from the control room.	
15.3.3.1	Diesel generators provide power to the 4.16 kV safety buses.	
15.3.3.1	An ADV is located downstream of the MSSVs and upstream of the MSIV in each steam line.	
15.3.3.1	Each ADV discharges to the atmosphere.	
15.3.3.3.1	An isolation valve (block valve) upstream of the ADV exists to be closed in case of the stuck open ADV.	
15.3.3.3.1	Reactor trip causes the turbine generator trip.	
Table 15.4.8-1	Main Steam Safety Valve: Main Steam Safety Valves - Open	1212 psia

Table 14.3-1 Design Basis Accident Analysis (Cont'd.)

Paragraph	Assumption/Parameter Description	Value
Table 15.4.8-3	Atmospheric Dispersion Factors - LPZ (30 days)	2.2×10^{-5} sec/m^3
— " —	Atmospheric Dispersion Factors - LPZ (1-4 days)	5.4×10^{-5} sec/m^3
— " —	Containment Vessel: Leak Rate	0.5% volume/day maximum
Table 15.4.8-3	Atmospheric Dispersion Factors - EAB (0-2 hr)	1.0×10^{-3} sec/m^3
— " —	Atmospheric Dispersion Factors - LPZ (0 - 8 hr)	1.35×10^{-4} sec/m^3
Table 15.4.8-3	Atmospheric Dispersion Factors - LPZ (8 - 24 hr)	1.0×10^{-4} sec/m^3
15.5.1.1	A SIAS actuates safety injection pumps and opens the corresponding discharge valves.	
Table 15.5.2-1	Main Steam Safety Valve: MSSV Opening Pressure Setpoint	1212 psia maximum
Table 15.6.2-3	Double-ended letdown line break size is assumed (0.01556 ft^2).	
Table 15.6.2-3	Letdown Line: Letdown Line Double Ended Break Size	0.01556 sq ft
15.6.2.1	Three letdown line isolation valves in series are located within the containment.	
15.6.2.2	The letdown line orifices are located within the containment downstream of the letdown heat exchanger.	
15.6.2.2	The hardware for DIAS (Discrete Indication and Alarm System) is seismically and environmentally qualified.	
Table 15.6.3-7	Main Steam Safety Valve: MSSV Opening Pressure Setpoint	1212 psia maximum
15.6.3.2.2.1	The turbine/generator trips on reactor trip.	
15.6.3.2.2.2	The minimum capacity of each EFW storage tank of 350,000 gallons is more than enough to maintain the plant at hot standby for 8 hours.	
15.6.3.2.2.2	Each EFW storage tank is provided with an atmospheric vent to maintain atmospheric pressure inside the tank.	
15.6.3.2.3.1	Emergency feedwater is actuated automatically to recover steam generator water level.	

Table 14.3-1 Design Basis Accident Analysis (Cont'd.)

Paragraph	Assumption/Parameter Description	Value
15.6.3.3.3.1	The Emergency Feedwater Actuation signal (EFAS) is generated on low SG level.	
15.6.3.3.3.1	The EFW flow is actuated on EFAS to restore the SG level.	
15.6.3.3.3.2	The reactor trip automatically trips the turbine generator.	
15.6.3.3.4	Main Steam Safety Valve: Maximum Allowable Pressure	110% of design pressure maximum
15.6.5.2	Containment Leak Rate, Per Day, (during 1st 24 hours of LOCA) Expressed as a Percentage of Containment Volume Per Day	0.5% nominal
15.6.5.3	Containment Leak Rate, Per Day, (during 1st 24 hours of LOCA) Expressed as a Percentage of Containment Volume Per Day	0.5% nominal
Table 15A-10	Unfiltered Normal Air Intake Rate	2000 cfm maximum
— " —	Post Accident Iodine Filter: Post Accident Intake and Recirculating Iodine Filter Efficiency - Elemental	95% minimum
Table 15A-10	Post Accident Iodine Filter: Post Accident Intake and Recirculating Iodine Filter Efficiency - Organic	95% minimum
— " —	Post Accident Iodine Filter: Post Accident Intake and Recirculating Iodine Filter Efficiency - Particulate	99% minimum
— " —	Control Room: Pressurization	1/8 in (water gauge) Nominal

[1] PRA Assumption

Table 14.3-3 Shutdown Risk (Cont'd.)

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Paragraph	Assumption/Parameter Description/Value
19.8A.2.8.3.2.5.3	[1] Discrete indicators are provided on the Nuplex 80+[†] control room stations to provide the operator with information that (1) is frequently used to assess system level performance and (2) allows continued operation if the data processing system should become unavailable.
19.8A.2.8.3.2.5.3	[1] Discrete indicator displays to support shutdown cooling for key parameters are on the engineered safety feature panel. These include: Shutdown Cooling System (per train); Inlet Temperature; Outlet Temperature; Heat Exchanger Inlet Temperature; Heat Exchanger Outlet Temperature; Pump Motor Current; Flow; Pump Header Pressure; Reactor Coolant System; Pressurizer Level; Reactor Coolant System Level; Pressure; Core Exit Temperature; Refueling Cavity.
19.8A.2.13	[1] Flood barriers provide divisional and quadrant separation up to the 70' elevation. Failure of largest storage tank within a division will not flood above the 70' level.
19.8A.2.13	[1] Door closed sensors will be provided on flood doors with indications available at a monitored location.
19.8A.2.13.3	[1] Flood barriers provide divisional and quadrant separation up to the 70' elevation. Failure of largest storage tank within a division will not flood above the 70' level.
19.8A.2.13.3	[1] Door closed sensors will be provided on flood doors with indications available at a monitored location.
19.8A.2.13.3	[1] It was assumed that the primary means of flood control in the Nuclear Annex and Reactor Building is provided by the divisional wall which serves as a barrier between redundant divisions of safety related equipment.
19.8A.4.1.2	Emergency Feedwater Cavitating Venturi: Emergency Feedwater Flowrate = 800 gpm maximum
19.8A.7.2.7	[1] The integration evident in the Nuplex 80+ displays and Mode dependent alarms contributes to plant safety by reducing the historically common personnel errors during Mode changes and outages.

Nuplex 80+

[†] ~~System 80+~~ is a trademark of Combustion Engineering, Inc.

[1] PRA Assumption

Table 14.3-7 Anticipated Transients Without Scram (Cont'd.)

Paragraph	Assumption/Parameter Description/Value
7.3.1.1.10.1	[1] The ESFAS actuates the Engineered Safety Features (ESF) systems when demanded.
7.3.1.1.10.2	[1] The ESFAS actuates the Engineered Safety Features (ESF) systems when demanded.
7.3.1.1.10.3	[1] The ESFAS actuates the Engineered Safety Features (ESF) systems when demanded.
7.3.1.1.10.4	[1] The ESFAS actuates the Engineered Safety Features (ESF) systems when demanded.
7.3.1.1.10.5	[1] The ESFAS actuates the Engineered Safety Features (ESF) systems when demanded.
7.3.1.1.10.5	The interlock on the EFW isolation valves automatically closes the isolation valves on high SG levels when an Emergency Feedwater Actuation Signal is not present.
7.7.1.1.11	[1] The Alternate Protection System (APS) provides an alternate means of generating a reactor trip signal and an alternate feedwater actuation signal.
7.7.1.1.11	[1] The APS monitors the pressurizer pressure and generates a reactor trip signal if the RCS pressure exceeds a predetermined value. Similarly, an alternate feedwater actuation signal is generated if the steam generator level decreases below a predetermined value.
7.7.1.1.11	[1] The EFWS is actuated by an EFAS and an APS actuation signal (Low SG Water Level).
7.7.1.1.11	The DIAS and DPS provide for monitoring: - safety-related plant process display instrumentation, - reactor trip system status, - engineered safety feature system status, - CEA position, - post-accident monitoring of plant safety functions, - status of plant operating mode-related bypasses, - core cooling status prior to and following an accident, - PPS status information, - ESF-CCS status information, - PCS/P-CCS status information.
7.7.1.1.11	The digital equipment and software used in the PCS/P-CCS are diverse from those used in the PPS and ESF-CCS.

Table 15.1.5-12 Parameters Used in Evaluating the Radiological Consequences of Steam Line Breaks Outside Containment Opstream of MSIV

Parameter		Value	
		SLBFPLOPD (Case 5)	SLBZPLOPD (Case 6)
A.	Data and Assumptions Used to Evaluate the Radioactive Source Term		
1.	Power Level, MWt	3992	10
2.	Burnup, MWD/MT	28,000	28,000
3.	Percent of Fuel Assumed to Experience DNB, %	0.5	0
4.	Reactor Coolant Activity Before Event (based on 3992 MWt),	Tech Spec Appendix 15A	Tech Spec[1] 3.4.15 Appendix 15A
5.	Secondary System Activity Before Event	Tech Spec Appendix 15A	Tech Spec 3.7.6 Appendix 15A
6.	Primary System Liquid Inventory, lbm	638,000	638,000
7.	Steam Generator Inventory, lbm		
	- Affected Steam Generator	108,640	414,386
	- Intact Steam Generator	108,640	414,386
B.	Data and Assumptions Used to Estimate Activity Released from the Secondary System		
1.	Primary to Secondary Leak Rate, gpm	1.0 (total)	1.0 (total)
2.	Total Mass Release from the Affected Steam Generator, lbm (0-30 min)	390,050	570,520
3.	Total Mass Release from the Intact Steam Generator	1,350,990 (2 hrs) 2,885,400 (8 hrs)	1,351,950 (2 hrs) 2,885,440 (8 hrs)
4.	Percent of Core Inventory of Volatile Fission Products Assumed in the Gap	5	N/A
5.	Iodine/Cesium/Rubidium Decontamination Factor in the Affected Steam Generator	1.0	1.0

[1] Except for cases assuming pre-existing and concurrent iodine spike.

Table 15.3.3-3 Parameters used in Evaluating the Radiological Consequences of a Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power Resulting from Turbine Trip

Parameters	Value
A. Data and Assumptions Used to Evaluate the Event's Radioactive Source Term	
1. Core Power Level, MWt	3992
2. Burnup, MWD/T	28,000
3. Percent of Fuel Calculated to Experience DNB, %	1.2[1]
4. Reactor Coolant Activity Before Event	Appendix 15A
5. Secondary System Activity Before Event, μCi/gm	Appendix 15A
6. Primary System Liquid Inventory, lbm	605,000
7. Steam Generator Inventory:	
- Liquid, lbm per steam generator	197,000
- Steam, lbm per steam generator	15,160
B. Data and Assumptions Used to Estimate Activity Released from the Secondary System	
1. Primary-to-Secondary Leak Rate, gpm	1.0 (total)
2. Total Mass Release Through the Main Steam Safety Valves, lbm	Table 15.3.3-4
3. Total Mass Release Through the ADVs from 30 to 120 Minutes, lbm	Table 15.3.3-4
4. Percent of Core Fission Products Assumed Released to Reactor Coolant	See App. 15A
5. Iodine Decontamination Factor for the Unaffected Steam Generator	100[2]
6. Iodine Decontamination Factor for the Affected Steam Generator	100[2]
7. Credit for Radioactive Decay in Transit to Dose Point	No
8. Loss of Offsite Power	Yes
C. Atmospheric Dispersion Factors	
1. at EAB, 0-2 hr, sec/m^3	Table 2.3-1
2. at LPZ, 0-8 hr, sec/m^3	Table 2.3-1
D. Dose Data	
1. Method of Dose Calculation	Appendix 15A
2. Dose Conversion Assumptions	Appendix 15A

(Generated Iodine Spike)

[1] No failed fuel assumed for GIS doses.

[2] Also, applicable to Cesium and Rubidium for the failed fuel case (see Appendix 15A).

Table 15.3.3-4 Secondary System Mass Release to the Atmosphere for the Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power Resulting from Turbine Trip Event

Time (sec)	Minimum SG Liquid Mass		Cumulative Steam Releases (lbm)		
			MSSVs		
	Unaffected (lbm)	Affected (lbm)	Unaffected SG	Affected SG	Total (incl. ADV)
0.000	197021.	197021.	0.	0.	0.
11.200	205082.	219448.	0.	0.	0.
74.100	180933.	212331.	25977.	17918.	43895.
263.60	157739.	202082.	57455.	17918.	75373.
407.40	146831.	192151.	67292.	27689.	94981.
553.00	138161.	181875.	76411.	36938.	113349.
682.80	145915.	172793.	76411.	47032.	123443.
848.60	140345.	161723.	85395.	56016.	141412.
1038.2	140420.	149485.	94118.	64804.	158922.
1229.8	140556.	138010.	102579.	73265.	175844.
1503.2	139849.	139765.	111039.	81725.	192764.
1784.6[1]	140593.	139864.	119367.	90053.	209420.
1800.0[2]	141528.	140676.	119367.	90053.	209420.
3600.0	141528.	140676.	119367.	90053.	495800.
7200.0	141528.	140676.	119367.	90053.	980900.
28800.0	141528.	140676.	119367.	90053.	1961500.

[1] Main steam safety valves close.

[2] Operator begins cooldown utilizing the atmospheric dump valves. One atmospheric dump valve is assumed to stick open for the next 30 minutes.

Table 15.3.3-5 Radiological Consequences of a Postulated Single Reactor Coolant Pump Rotor Seizure with Loss of Offsite Power Resulting from Turbine Trip

Define "GIS"
200 μCi/g?

Location	Doses from Secondary System Steam Releases, rem	
	GIS[1]	Failed Fuel
Exclusion Area Boundary (0-2 hours)		
Thyroid	0.6	3.18
Whole-body	0.02	0.13
Low Population Zone (0-8 hours)		
Thyroid	1.33	2.18
Whole-body	0.01	0.04

[1] Generated Iodine spike

Table 15.4.8-3 Parameters Used in Evaluating the Radiological Consequences of a CEA Ejection Event

Parameter	Value
A. Data and Assumptions Used to Evaluate the Event's Radioactive Source Term	
1. General	
a. Core Power Level, MW	3992
b. Burnup, MWD/MT	28,000
c. Percent of Fuel Calculated to Experience DNB, %	6.8
d. Percent of Fuel Calculated to Experience Incipient Centerline Melt, %	0.0
e. Reactor Coolant Activity Before Event	Tech Spec 3.4.15 Appendix 15A
f. Secondary System Activity Before Event	Tech Spec 3.7.6 Appendix 15A
g. Primary System Liquid Inventory, lbm	605,000
h. Steam Generator Inventory	
- Liquid, lbm per steam generator	117,000
- Steam, lbm per steam generator	23,700
i. Average peaking factor	1.3
B. Data and Assumptions Used to Estimate Activity Released	
1. Containment Leakage	
a. Containment Volume, ft^3	3.34 E06
b. Containment Leak Rate, vol. %/day	Table 15.6.5-2 ~~Section 15.6.5~~
c. Percent of Core Fission Products Assumed Released to Containment	~~Refer to~~ Section 15.4.8.3
d. Natural Deposition in Containment	Yes $\lambda = 0.15\ hr^{-1}$ for particulate $\lambda = 2.89\ hr^{-1}$ for elemental iodine
e. Credit for Radioactive Decay	
- Hold up in Containment	Yes
- In Transit to Dose Point	No

Table 15.4.8-3 Parameters Used in Evaluating the Radiological Consequences of a CEA Ejection Event (Cont'd.)

Parameter			Value
2.	Activity Release from the Secondary System		
	a.	Primary-to-Secondary Leak Rate, gpm	1.0 (total)
	b.	Total Mass Release Through the Main Steam Safety Valves, lbm	317,100
	c.	Total Mass Release Through the ADVs from 30 minutes to 120 minutes, lbm	773,300
	d.	Total Mass Release Rate through the ADVs from 120 minutes to Shutdown Cooling Startup (480 minutes), lbm/hr	153,300
	e.	Percent of Core Fission Products Assumed Released to Reactor Coolant	Refer to Section 15.4.8.3
	f.	Iodine Carryover Fraction in the Steam Generators	Appendix 15A
	g.	Credit for Radioactive Decay in Transit to Dose Point	No
	h.	Loss of Offsite Power	Yes
C.	Atmospheric Dispersion Factors (from Table 2.3-1)		
1.	At EAB, 0-2 hr, sec/m^3		$1.0x10^{-3}$
2.	At LPZ, 0-8 hr, sec/m^3		$1.35x10^{-4}$
		8-24 hr	$1.0x10^{-4}$
		1-4 days	$5.4x10^{-5}$
		4-30 days	$2.2x10^{-5}$
D.	Engineered Safety Features		
1.	Containment Spray Credit		None
2.	Annulus Building Ventilation		After 30 minutes
3.	Containment Power Purge Isolation		
	a.	Isolation Time	40 seconds
	b.	Flowrate Prior to Isolation	1250 cfm
E.	Dose Data		
1.	Method of Dose Calculation		Appendix 15A
2.	Dose Conversion Assumptions		Appendix 15A

Table 15.6.3-1 Sequence of Events for the Steam Generator Tube Rupture

Time (Sec)	Event	Setpoint or Value
0.0	Tube Rupture Occurs	–
0.4	High Steam Generator Level Trip Signal Generated	–
0.55	Trip Breakers Open	–
0.55	Turbine Trip: Stop Valves Start to Close	–
5.4	Main Steam Safety Valves Open, psia	1212
5.75	Main Steam and Feedwater Isolation Valves Closed	–
8.77	Maximum Steam Generator Pressure, psia	1273
16.5	Backup Heaters Energized, psia	2325
623	Pressurizer Heaters Deenergize due to Low Pressurizer Liquid Volume, ft^3	297
1800	Operator Isolates the Damaged Steam Generator and Initiates Plant Cooldown at 100°F/hr for the 1.5 hour time period	–
28,800	Shutdown Cooling Entry Conditions are Assumed to be reached; RCS Pressure, psia / RCS Temperature, °F	330/350

Table 15.6.3-2 Assumptions and Initial Conditions for the Steam Generator Tube Rupture

Parameters	Assumed Value
Core Power Level, MWt	3876
Core Inlet Coolant Temperature, °F	563
Pressurizer Pressure, psia	2375
Core Mass Flow Rate, 10^6 lbm/hr	151.9
One Pin Integrated Radial Peaking Factor, with Uncertainty	1.46
Steam Generator Pressure, psia	1057
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho$/°F	0.0
Doppler Coefficient Multiplier	1.0
CEA Worth at Trip, % $\Delta\rho$ (most reactive CEA fully withdrawn)	-8.86
Doppler Reactivity Function	See Table 15.0-6

Table 15.6.3-5 Assumptions and Initial Conditions for the Steam Generator Tube Rupture with a Loss of Offsite Power

Parameter	Assumed Value
Core Power Level, MWt	3876
Core Inlet Coolant Temperature, °F	563
Pressurizer Pressure, psia	2375
Core Mass Flow Rate, 10^6 lbm/hr	151.9
One Pin Integrated Radial Peaking Factor, with Uncertainty	1.46
Steam Generator Pressure, psia	1057
Moderator Temperature Coefficient, 10^{-4} $\Delta\rho$/°F	0.0
Doppler Coefficient Multiplier	1.0
CEA Worth at Trip, % $\Delta\rho$ (most reactive CEA fully withdrawn)	-8.86
Doppler Reactivity Feedback Function	See Table 15.0-6

Table 15.6.3-6 Radiological Consequences of the Steam Generator Tube Rupture with a Loss of Offsite Power

Thyroid Inhalation Doses[1]			
Location		Offsite Doses[2] (rem)	
		GIS	PIS
1.	Exclusion Area Boundary; 0-2 hr, Thyroid	9.53	38.6
2.	Low Population Zone Outer Boundary; 0-8 hr, Thyroid	4.23	5.98
Whole-Body Doses[1]			
Location		Offsite Doses[2] (rem)	
		GIS	PIS
1.	Exclusion Area Boundary; 0-2 hr, Whole-Body	0.112	0.143
2.	Low Population Zone Outer Boundary; 0-8 hr, Whole-Body	0.019	0.02

[1] Radiological releases were determined for a core power of 3992 MWt by increasing the steam releases for the 3876 MWt core power case by 3%

[2] GIS - Generated Iodine Spike
PIS - Pre-accident Iodine Spike

Table 15.6.3-7 Sequence of Events for a Steam Generator Tube Rupture with a Loss of Offsite Power and Stuck Open ADV

Time	Event	Setpoint or Value
0.0	Tube Rupture Occurs	---
194.3	Backup Heaters Energized, psia	2325
1450	Pressurizer Heaters De-energized due to Low Pressurizer Liquid Volume, ft^3	297
1756.97	High Steam Generator Level Condition, % Narrow Range	95
1757.97	High Steam Generator Level Trip Signal Generated	---
1758.12	Trip Breakers Open	---
1758.12	Turbine Generator Trip	---
1761.12	Loss of Offsite Power	---
1762	LH Main Steam Safety Valves open, psia	1212
1762	RH Main Steam Safety Valves open, psia	1212
1764.88	Maximum Steam Generator Pressures Both Steam Generator, psia	1272
1783.16	Pressurizer Empties	---
1783.68	Steam Generator Water Level Reaches Emergency Feedwater Actuation Signal (EFAS) Analysis Setpoint in the Unaffected Generator, % wide range	26.9
1843.68	Emergency Feedwater Initiated to Unaffected Steam Generator	---
1849.26	Main Steam Safety Valves Closed, psia	1151.4
2178	Operator Initiates Plant Cooldown by Opening One ADV on each SG	---
2179	Operator Initiates Safety Injection Flow	---
3663	Operator Attempts to Isolate the Damaged Generator, RCS Temperature, °F	550
5463	Operator Closes the ADV Block Valve	---
5583	Operator Opens Pressurizer Gas Vent	---
8500	Operator Closes Pressurizer Gas Vent and Controls Backup Pressurizer Heater Output, and SI Flow to Reduce RCS Pressure and Control Subcooling, °F	20
28,800	Shutdown Cooling Entry Conditions Reached; RCS Pressure, psia / Temperature, °F	330/350

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTE--------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation device outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation device inside containment
B. ---------NOTE--------- Only applicable to those penetration flow paths with two containment isolation valves. ---------------------- One or more penetration flow paths with two containment isolation valves inoperable [except for purge valve leakage and shield building leakage not within limit]. — Redline	B.1 Isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. Add Blank Line	1 hour

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Verify by administrative means [Region 2] II fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

Region II

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.19.1 Verify the fuel storage pool boron concentration is within limit.	7 days

3.7 PLANT SYSTEMS

3.7.20 Spent Fuel Assembly Storage

LCO 3.7.20 The combination of initial enrichment and burnup of each spent fuel assembly stored in [Region 2] shall be within the acceptable [burnup domain] of Figure 3.7.20-1 [or in accordance with Specification 4.3.1.1].

APPLICABILITY: Whenever any fuel assembly is stored in [Region 2] of the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	--------NOTE---------- LCO 3.0.3 is not applicable. ---------------------- Initiate action to move the noncomplying fuel from [Region 2].	Immediately

Region II (4K)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.20.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.20-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly in [Region 2]

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC Electrical Power Sources shall be OPERABLE.

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Distribution System; and

b. Two diesel generators (DGs), each capable of supplying one division of the onsite Class 1E AC Distribution System.

c. Automatic load sequencers for Division 1 and Division 2.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	A.1 Perform SR 3.8.1.1 for the required OPERABLE offsite circuit.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	24 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s)
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3.2 Perform SR 3.8.1.2 for OPERABLE DG. <u>AND</u> B.4 Verify the combustion turbine generator (CTG) is functional by verifying the CTG starts and achieves steady state voltage and frequency within [2] minutes. <u>AND</u> B.5 Verify the CTG is capable of being aligned to the ESF buses associated with the inoperable DG. <u>AND</u> B.6 Restore required DG to OPERABLE status.	24 hours 72 hours 72 hours <u>AND</u> Once per 8 hours thereafter 14 days <u>AND</u> 15 days from discovery of failure to meet LCO
C. Two required offsite circuits inoperable.	C.1 Declare required feature(s) inoperable when its redundant required feature(s) is inoperable. <u>AND</u>	12 hours from discovery of Condition C concurrent with inoperability of redundant required features

(continued)

also pg 3.8-4

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2	Restore one required offsite circuit to OPERABLE status.	24 hours
D. One required offsite circuit inoperable. <u>AND</u> One required DG inoperable.	D.1	--------NOTE---------- Enter applicable Conditions and Required Actions of LCO 3.8.9. "Distribution Systems - Operating", when Condition D is entered with no AC power source to one division. --------------------- Verify the combustion turbine generator (CTG) is functional by verifying the CTG starts and achieves steady state voltage and frequency within [2] minutes.	12 hours
	<u>AND</u>		
	D.2	Verify the CTG is capable of being aligned to the ESF buses associated with the inoperable DG.	12 hours <u>AND</u> Once per 8 hours thereafter
	<u>AND</u> redline		(continued)

5.5 Reviews and Audits

5.5.1.1 Functions (continued)

c. Determine whether each item considered under Specifications 5.5.1.2.a through 5.5.1.2.d constitutes an unreviewed safety question as defined in 10 CFR 50.59; and

d. Notify the [Vice President - Nuclear Operations] of any safety significant disagreement between the [review organization or individual specified in Specification 5.5.1] and the [Plant Superintendent] within 24 hours. However, the [Plant Superintendent] shall have responsibility for resolution of such disagreements pursuant to Specification 5.1.1.

5.5.1.2 Responsibilities

The [plant review method specified in Specification 5.5.1] shall be used to conduct, as a minimum, reviews of the following:

a. All proposed procedures required by Specification 5.7.1.1 and changes thereto;

b. All proposed programs required by Specification 5.7.2 and changes thereto;

c. All proposed changes and modifications to unit systems or equipment that affect nuclear safety;

d. All proposed tests and experiments that affect nuclear safety;

e. Review and documentation of judgment concerning prolonged operation with protection channels placed in bypass since the last [plant review meeting] and the repair of these channels; and

f. All proposed changes to these Technical Specifications (TS), their Bases, and the Operating License.

5.5.2 <u>[Offsite] Review and Audit</u>

[The licensee shall describe the provisions for reviews and audits independent of the plant's staff (organization, reporting, and records) and the appropriate ANSI/ANS standards for personnel qualifications. These individuals may be located onsite or offsite provided organizational independence from plant staff is

(continued)

5.7 Procedures, Programs, and Manuals (continued)

~~5.7.2 Programs and Manuals (continued)~~

5.7.2.8 Radiological Environmental Monitoring Program

This program is for monitoring the radiation and radionuclides in the environs of the plant. The program shall provide representative measurements of radioactivity in the highest potential exposure pathways and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall be contained in the ODCM, shall conform to the guidance of 10 CFR 50, Appendix I, and shall include the following:

a. Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

b. A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of this census; and

c. Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

5.7.2.9 Component Cyclic or Transient Limit

This program provides controls to track the CESSAR-DC, Chapter 3 cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.7.2.10 Inservice Inspection Program

This program provides controls for inservice inspection of ASME Code Class 1, 2, and 3 components, including applicable supports. The program shall include the following:

a. Provisions that inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda, as required by 10 CFR 50.55a;

b. Provisions for safety-related snubbers in accordance with 10 CFR 50.55a. The only snubbers excluded from this requirement are installed on nonsafety related systems and then only if their failure, or failure of the system on

(continued)

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3.0 LCO APPLICABILITY

LCO 3.0.7 (continued)

effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCOs is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

Some of the STE LCOs require that one or more of the LCOs for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCOs). The Applicability, ACTIONS, and SRs of the specified normal LCOs, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist. There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCOs are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCOs must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCOs provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only tne failure to meet the ACTIONS of the STE LCO.

add Double Line !

BASES

1. Variable Overpower - High (continued)

LCO ~~(continued)~~

conservative including all applicable setpoint uncertainties.

The Variable Overpower trip is applicable in MODES 1 and 2 because the reactor can be critical in these modes. The trip is designed to take the reactor subcritical which assists (as described above) in mitigating the consequences of the particular accidents and AOOs listed.

In MODES 3, 4, and 5 the main concern is for a return to power event. The reactor is protected during this event by the High Log Power trip, and therefore, the above trip does not need to be OPERABLE.

2. Logarithmic Power Level - High

This LCO requires four CHANNELS of the Logarithmic Power Level - High to be OPERABLE in MODE 2, and in MODE 3, 4, or 5 when the RTCBs are closed and the CEA Drive System is capable of CEA withdrawal.

The MODES 3, 4, and 5 Condition is addressed in LCO 3.3.2.

The LCO on the Log Power Level - High trip ensures that violation of the Safety Limits for the reactor core and RCS is prevented during a continuous CEA withdrawal from low power levels event. Also, it ensures that the log power level CHANNELS are available to detect and alert the operator to a boron dilution event.

The allowable value setpoint is high enough to provide an operating envelope that prevents unnecessary Log Power Level - High reactor trips during normal plant operations. The setpoint is low enough for the system to maintain a margin to unacceptable fuel cladding damage should a CEA withdrawal event occur.

(continued)

BASES

LCO ~~(continued)~~ cc

2. Logarithmic Power Level - High (continued)

Only the Allowable Values are specified for each RPS trip function in the LCO. Each allowable value is specified such that the analytical limit assumed in the safety analysis is conservative including all applicable setpoint uncertainties.

The Logarithmic Power Level - High trip may be bypassed when THERMAL POWER is above [1E-4%] RTP to allow the reactor to be brought to power during a reactor startup. This bypass is automatically removed when THERMAL POWER decreases below [1E-4%] RTP. Above [1E-4%] RTP, the Variable Overpower - High and Pressurizer Pressure - High trips provide protection for reactivity transients.

The trip may be manually bypassed during physics testing pursuant to LCO 3.1.16, "RCS Loops - Test Exceptions." During this testing, the Variable Overpower - High trip and administrative controls provide the required protection.

3. Pressurizer Pressure - High

This LCO requires four CHANNELS of Pressurizer Pressure - High to be OPERABLE in MODES 1 and 2.

The Allowable Value is set below the nominal lift setting of the pressurizer code safety valves, and its operation avoids the undesirable operation of these valves during normal plant operation. In the event of a complete loss of electrical load from 100% power, this setpoint ensures the reactor trip will take place, thereby limiting further heat input to the RCS and consequent pressure rise. The pressurizer safety valves may lift to prevent overpressurization of the RCS.

4. Pressurizer Pressure - Low

This LCO requires four CHANNELS of Pressurizer Pressure - Low to be OPERABLE in MODES 1 and 2.

(continued)

ACTIONS (continued)

C.1, C.2.1, and C.2.2

Condition C applies to one automatic operating bypass removal function inoperable. If the inoperable bypass removal function for any TRIP CHANNEL cannot be restored to OPERABLE status within 1 hour, the associated RPS channel may be considered OPERABLE only if the bypass is not in effect. The operator must verify that the operating bypass is not in effect within one hour and every 12 hours thereafter; otherwise the affected RPS channel must be declared inoperable, as in Condition A, and the affected automatic TRIP CHANNEL placed in bypass or trip. The operating bypass removal function and the automatic TRIP CHANNEL must be repaired prior to entering MODE 2 following the next MODE 5 entry. The Bases for the Required Actions and Required Completion Times are consistent with Condition A.

The Required Action is modified by a Note stating that this LCO applies only to Functions 1, 5, and 6. This Note aids in identifying the applicable functions; Logarithmic Power Level - High, Reactor Coolant Flow - Low, LPD - High, and DNBR - Low.

D.1 and D.2

Condition D applies to two inoperable automatic operating bypass removal functions. If the operating bypass removal functions for two operating bypasses cannot be restored to OPERABLE status within 1 hour, the associated TRIP CHANNEL may be considered OPERABLE only if the operating bypasses are not in effect. The operator must verify that the operating bypass is not in effect within one hour and every 12 hours thereafter; otherwise the affected RPS channels must be declared inoperable, as in Condition B, and the operating bypasses either removed or one automatic TRIP CHANNEL placed in bypass and the other in trip within 1 hour. The restoration of one affected bypassed automatic trip channel must be completed prior to the next CHANNEL FUNCTIONAL TEST, or the plant must shut down per LCO 3.0.3 as explained in Condition B.

(continued)

BASES

2 spaces

BACKGROUND

LOGIC CHANNEL (continued)

logic either locally at the maintenance and test panels or remotely via the operator's module. The bypass status is available for display at the local maintenance and test panels, remote operators modules, and DPS.

ACTUATION LOGIC

The ESFAS Actuation Logic consists of a selective two-out-of-four logic for each ESFAS function.

The inputs to the ACTUATION LOGIC are the LCL outputs from the appropriate local coincidence logics. The initiation circuits also contain a time delay (TD). The TD functions as a noise filter. It accomplishes this filter action by monitoring the continuous presence of an input for a minimum period of time. If the signal is present for the required time, the signal is transmitted to the initiation relay. Test capability is also provided.

The initiation circuit is designed to fail-safe (i.e., in a trip condition). This will result in a partial trip (1 of 4) in the selective 2-out-of-4 ESFAS actuation logic. The partial trip will be alarmed the same as a full ESF trip and actuation and will be indicated by the DIAS and DPS; the partial trip cannot be bypassed. If the initiation circuit fails in an undesired condition the failure will be promptly detected and alarmed via the automatic test function. Since the actuation functions in the ESF-CCS work in a selective coincidence logic, this is considered a degraded condition and a technical specification LCO will apply. CESSAR-DC Section 7.3 (Ref. 1) describes ACTUATION LOGIC in detail.

COMPONENT CONTROL LOGIC

The COMPONENT CONTROL LOGIC is used to actuate the individual ESF components which are actuated to mitigate the consequences of the occurrence that caused the actuation.

The ESFAS actuation and component control logics are physically located in four independent and geographically separate ESF-CCS cabinets.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.3 (continued)

2. Electrical and mechanical instrument and equipment uncertainties must be considered. However, harsh environment uncertainties need not be included for the AFAS setpoint analysis.

[CHANNEL CALIBRATION shall find measurement errors are within the acceptance criteria specified in Reference 3.]

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an [18] month calibration interval for the determination of the magnitude of equipment drift.

REFERENCES

1. 10 CFR 50, Appendix A
2. Section 7.7
3. [Setpoint Report]
4. Sections 15.2, 15.4, and 15.5

BASES

LCO (continued)

of the vessel to become more pronounced), and the consequences also depend on the existences, sizes and orientations of flaws in the vessel material. Although vessel failure is not an expected outcome of a violation, the possibility for failure exists.

APPLICABILITY

The RCS P/T limits provides a definition of acceptable operation for prevention of non-ductile failure that is in accordance with 10 CFR 50 Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup and cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for non-ductile failure. At all times is defined to be any condition with fuel in the reactor vessel. The limits do not apply to the pressurizer.

However, during MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement the P/T limits. These other LCOs include LCO 3.4.2, "RCS Minimum Temperature for Criticality," and LCO 3.4.1, "RCS Pressure, Temperature, and Flow Limits." SL 2.1, safety limits for pressure and temperature and maximum pressure, also provides operational restrictions. In MODE 6, with the reactor vessel head detensioned or removed, the capability for violating the P/T curves does not exist, however the potential for violating the temperature rate-of-change limit remains.

Furthermore, in MODES 1 and 2, operation is above the temperature range of concern for non-ductile failure. As such, stress analyses have been developed in accordance with normal maneuvering profiles such as power ascension.

The actions of this LCO consider the premise that a violation of the limits occurred during normal plant maneuvering. Severe violations caused by abnormal transients, which may be accompanied by equipment failures, may also require additional actions based on emergency operating procedures.

(continued)

CAPS (all of B 3.5)

1

BACKGROUND (continued)

interlocked with the pressurizer pressure instrumentation channels to ensure the valves will automatically open as RCS pressure is increased above SIT pressure and to prevent inadvertent closure prior to an accident. The valves also receive a Safety Injection Actuation Signal (SIAS) to open. These features ensure the valves meet the requirements of IEEE Std 279-1971 (Ref. 1) for "operating bypasses" and that the SITs will be available for injection without reliance on operator action.

The SIT gas and water volumes, gas pressure, and outlet pipe size are selected to allow three of the four SITs to partially recover the core before significant clad melting or zirconium-water reaction can occur following a LOCA. The need to ensure that three SITs are adequate for this function is consistent with LOCA analysis assumption that the entire contents of one SIT will be lost via the break during the blowdown phase of a LOCA.

APPLICABLE SAFETY ANALYSES

The SITs are taken credit for in both the large and small break LOCA analysis at full power (Ref. 3). These are the Design Basis Accidents (DBAs) that establish the acceptance limits for the SITs. Reference to the analyses for these DBAs is used to assess changes to the SITs as they relate to the acceptance limits.

In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. These assumptions include signal generation time, equipment starting times, and delivery time due to system piping. In the early stages of a LOCA with a loss of offsite power, the SITs provide the sole source of makeup water to the RCS. (The assumption of a loss of offsite power is required by regulations). This is because the safety injection pumps cannot deliver flow until the diesel generators (DGs) start, come to rated speed, and go through their timed loading sequence. In cold leg breaks, the entire contents of one SIT are assumed to be lost through the break during blowdown, even through the SITs discharge their contents directly to vessel downcomer via the direct vessel injection nozzle.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.7.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. Chapter 8.
2. Chapter 6.
3. Chapter 15.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two cases were analyzed: (1) a TLOFW event with one RDF bleed path open, two SI pumps operable, and immediate operator action to open the RDF bleed path after the primary safety valves (PSVs) open, and (2) a TLOFW event with both RDF bleed paths operable, four SI pumps operable, and an operator delay to open the RDF paths after the PSVs open. The analysis shows that case 2 is the worst case, which requires larger RDF bleed valves, each sized to meet the acceptance criteria.

The RDS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires the RDF to be OPERABLE. Both vent paths shall be closed for all design basis events. The RDF is OPERABLE when a vent path can be established form the pressurizer to the IRWST.

APPLICABILITY

In MODES 1, 2, 3, and 4, at least one vent path is required to be operable, and both vent paths closed. The RDF is for use in beyond-design-bases events such as a TLOFW, and for mitigating severe accidents such as a core melt.

ACTIONS

A.1, A.2, and A.3

With inoperable components, such that both vent paths are inoperable, one of the two vent paths must be returned to OPERABLE status within 72 hours. If at least one RDF vent path cannot be made OPERABLE within 72 hours, then the plant must be in MODE 3 within an additional 6 hours, and then in MODE 5 within an additional 36 hours. The 72 hour Completion Time is based on the extremely low probability of the beyond-design-basis event (TLOFW) that the RDF is designed for and reflects an adequate time allotted for return of redundant safety grade systems to OPERABLE status.

(continued)

Maintenance Rule Integration -- The plant owner/operator should consider the integration or interface of operations reliability assurance process and the requirements of 10 CFR 50.65 which require the operator to develop a maintenance program for risk significant SSCs or SSCs that could produce trips or transients.

The plant owner's operations reliability assurance process should address the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance and procurement of replacement equipment.

17.3.11 D-RAP Implementation

An example of implementation of the D-RAP is given for the Component Cooling Water System (CCWS). This system was selected as an example because it was a support system and was found in the earlier System 80 PRA to contain risk-significant components. Because of this finding, and through the D-RAP organization described in Section 17.3.5, the design was changed. The design and analytical results, as presented in this chapter, is presented only as a D-RAP example and does not necessarily correspond to the current System 80+ design.

17.3.11.1 CCWS Function

The Component Cooling Water System (CCWS) is a closed loop system that provides cooling water flow to remove heat released from plant systems, structures, and components. The CCWS functions to cool the safety-related and non-safety-related reactor auxiliary loads.

Heat transferred by these components to the CCWS is rejected ~~by~~ to the Station Service Water System (SSWS) via the CCWS heat exchangers.

17.3.11.2 Earlier CCWS Design

The System 80+ Design is an evolutionary plant and improvements were included with input from the earlier System 80 PRA. The earlier CCWS design is shown in Figure 17.3-8 and described in more detail in section 5.3.19 of Reference 17.3-2. It consisted of two independent, closed loop, safety trains. Each train contained one pump that was on standby. One of the major insights of the System 80 PRA (Section 8.2 of Reference 17.3-2) was that loss of the CCWS was a dominant cause of front-line system failure. Failure of the CCWS pumps to start and run was one of the dominant failure modes.

17.3.11.3 System Redesign

To more easily meet the desired CDF for the ALWR, the CCWS required a redesign using the process identified in Figure 17.3-2. This redesign was also helped by design review meetings where the Project Manager for the RAP and PRA discussed with the designers the PRA results, including failure modes and importance of support systems to front line safety systems. An example of an improved CCWS design is given in Figure 17.3-9 and an example of analytical results are presented in Tables 17.3-1 and 17.3-2. Details of the actual System 80+ CCWS design and reliability analysis are given in the System 80+ PRA and do not necessarily correspond to the example presented here.

contributed by failure of

The improved CCWS design contains two trains (only one is shown in Figure 17.3-9). Each train contains two pumps and one pump is kept running at all times. This design eliminated the important failure mode of the CCWS pump failing to start which was observed in the earlier design. Table 17.3-1 gives an example of the components importance for internal events for an ALWR. The Fussell-Vesley Importance is the fraction of the CDF that ~~the~~ component ~~failure contributes to~~. ~~In this example, the components in the CCWS are underlined~~. The first CCWS component is only ranked 59th in importance based on this measure. The components in the improved CCWS meet the criteria that they have a small impact on risk (bottom of Figure 17.3-2) and can be considered in an operations reliability assurance process. e

49

17.3.11.4 Failure Mode Identification

Figure 17.3-3 gives two methods for operations reliability assurance process evaluation, using failure history or analytical methods. For this example, an analytical method as represented in Figure 17.3-5 was used. Figure 17.3-10 gives an example of the upper level fault tree to analyze failure modes for Train 1 of an improved CCWS. Table 17.3-2 gives an example of the ranking of the risk significant SSCs for Train 1. There is also a second train not evaluated in this table. Because this CCWS design is an evolutionary design using standard components, a search of the operational data bases for component failure rates and operations experience is also possible using Figure 17.3-4, but was not used in this example.

Following the flow chart of Figure 17.3-5, the designer would determine more details about each failure mode, including pieceparts most likely to fail and the frequency of each failure mode category or piecepart failure. This would result in a list of the dominant failure modes to be considered in the operations reliability assurance process. ASME Section XI requirements for inspection and other mandated inspections and tests would be identified, as indicated in Figure 17.3-6.

Examples of the types of failure modes that could impact reliability of these identified components are shown in Table 17.3-3. The example is not a complete listing of the important failure modes, but is intended to indicate the types of failures that would be considered.

17.3.11.5 Identification of Maintenance Requirements

For each identified failure mode, the appropriate maintenance tasks will be identified to assure that the failure mode will be (1) avoided, (2) rendered insignificant, or (3) kept to an acceptably low probability. The type of maintenance and the maintenance frequencies are both important aspects of assuring that the equipment failure will be consistent with that assumed for the PRA. As indicated in Figure 17.3-7, the designer would consider periodic testing, performance testing or periodic preventive maintenance as possible operations reliability assurance process activities to keep failure rates acceptable.

For the CCWS, one pump in each train is in operation and all the valves in that flow path are open. An example of the possible maintenance and testing follows and is summarized in Table 17.3-3. Minor PM on the pumps will be performed based on the recommendations of the vender (8000 hrs of operation, for example) and a major overhaul would be performed every 50,000 hrs of operation. Only maintenance on one pump will be performed at a time during Modes 1 through 4. The most frequent surveillance requirement for the CCWS might be to verify that each CCW manual, power-operated or automatic valve in the flow path servicing essential equipment, that is not locked, sealed, or otherwise secured in position, is in its correct position. This test is performed every 31 days. Additionally, there is a surveillance requirement that every 18 months, it must be demonstrated that each CCW automatic valve actuates and each CCW pump starts on an actual or simulated actuation signal. Example of maintenance activities and

Table 17.3-2 Example of Risk-Significant Ranking of SSCS for the CCWS Train 1

Rank[1]/Component Name		Description
1)	CVNDCC-1316	Manual Valve CC-1316 Fails to Remain Open
2)	CPBKCCWP1A	Component Cooling Water Pump 1A Fails to Run
	CPBVCCWP1B	CCW Pump 1B Unavailable Due to Maintenance
3)	CPBJCCWP1B	CCW Pump 1B Fails to Start
	CPBKCCWP1A	Component Cooling Water Pump 1A Fails to Run
4)	CHFLCC-1305	Valve CC-1305 not Opened Due to Pre-existing Maint. Error
	CPBKCCWP1A	Component Cooling Water Pump 1A Fails to Run
5)	CHWEHX1A	CCW/SW Heat Exchgr. 1A Fails While Operating
	CVMACC-107	MOV CC-107 Fails to Open
6)	CHWEHX1A	CCW/SW Heat Exchgr 1A Fails While Operating
	CVMACC-109	MOV CC-109 Fails to Open
7)	CHWEHX1A	CCW/SW Heat Exchngr 1A Fails While Operating
	CVMASW-123	MOV SW-123 Fails to Open
8)	CHWEHX1A	CCW/SW Heat Exchngr 1A Fails While Operating
	CVMASW-121	MOV SW-121 Fails to Open
9)	CHFFSTBHX1B	Operator Fails to Open CCW HX 1B Isolation Valves
	CHWEHX1A	CCW/SW Heat Exchngr 1A Fails While Operating
10)	CBDBCCWP1B	4.16 Kv Circuit Breaker 1B Fails to Close
	CPBKCCWP1A	Component Cooling Water Pump 1A Fails to Run
11)	CPBVCCWP1B	CCW Pump 1B Unavailable Due to Maintenance
	CVCDCC-1302	Check Valve CC-1302 Fails to Remain Open
12)	CBDQCCWP1A	4.16 Kv Circuit Breaker 1A Trips Spuriously
	CPBVCCWP1B	CCW Pump 1B Unavailable Due to Maintenance
	CVCACC-1303	Check Valve CC-1303 Fails to Open
14)	CPBJCCWP1B	CCW Pump 1B Fails to Start
	CVCDCC-1302	Check Valve CC-1302 Fails to Remain Open

13 CPBKCCWP1A Component Cooling Water Pump 1A Fails to Run

- Alarm Representations

Visual alarm information in NUPLEX 80+ is identified by a unique hue, yellow. Different hues were not used to differentiate priorities because this limited the number of hues available for other purposes and using one hue (yellow) for alarms reduced search time for the existence of alarms (more important information than alarm priority). Position coding was not feasible. NUREG 0700 provides recommendation that shape is an acceptable coding mechanism and was thus chosen. Three levels of intensity have been assigned to differentiate the state of the alarms. New alarms fast flash with a high intensity yellow, existing alarms are solid (i.e. not flashing) with a medium intensity yellow, cleared alarms slow flash with a low intensity yellow. This approach allows all alarm conditions to be quickly and uniquely recognized by the hue yellow and allows the alarm state to be determined uniquely by the intensity of yellow.

Shape coding is used to identify alarm priority; i.e. 1, 2, or 3. The shape coding used for identifying alarm priorities uses representational features of decreasing levels of salience. Shape coding of alarm priorities also allows retention of priority information for Return to Normal conditions. Two borders have been defined around a descriptor or alarm tile as an enhancement, (not a code) that increases brightness and saliency of the coding between the different intensities of yellow used to distinguish unacknowledged new alarms (most important) from unacknowledged cleared and acknowledged existing alarm intensities. These borders, typically three pixels thick, define a spatial difference between the new and existing alarms. New alarms will flash the existing area and the additional three pixel area giving the effect of the alarm jumping out at the operator. This enhancement increases brightness and salience of the coding between the different intensities of yellow used to distinguish unacknowledged new alarms (most important) from unacknowledged cleared and acknowledged existing alarm hues. Cleared alarms will be shown using the same area as existing alarms but with a different uneven flash duty cycle (see Figure 18.7.1.8). The following provides the format for alarm representations in Nuplex 80+.

1. Priority 1 alarms

   Alarm tiles, mimic diagram component descriptors, symbols, process parameter descriptors, and directory/display page option fields have their descriptor presented in reverse video image using the alarm hue coding. On the ~~CRT~~ the descriptor is presented grey for static data and in blue for dynamic data to provide good contrast for readability. In addition, the alarm tile and alarm list status fields on the ~~CRT~~ use the same representation. VDU

2. Priority 2 alarms

   Alarm tiles, mimic diagram parameter descriptors, component descriptors, and menu/display page options have a thin box using the alarm hue code around their descriptor.

3. Priority 3 alarms

   Alarm tiles, mimic diagram parameter descriptors, component descriptors, and menu/display page options have brackets around their descriptors.

ee

7. *All five (reactivity control, reactor core cooling and heat removal from the primary system, reactor core cooling and heat removal from the primary system, reactor coolant system integrity, radioactivity control, and containment conditions) of the safety function elements are included in the DPS Critical Functions Monitoring hierarchy which forms the basis of the Nuplex 80+ SPDS function.*

8. *The System 80+ Critical Functions Monitoring function (SPDS) is developed in a complementary (parallel) fashion with the development of System 80+ Emergency Operations Guidelines. Generic emergency procedure guidelines are used during the design process.*]][1]

Attachment 2 of Reference 2 provides a more detailed explanation of how Nuplex 80+ complies with NUREG-0737 Supplement 1 requirements.

18.7.1.8.2 Critical Function and Success Path Monitoring

Critical Function and Success Path (availability and performance) information is integrated throughout the Nuplex 80+ information hierarchy. The critical function and success path monitoring application programs in conjunction with the continuous IPSO display and the DPS VDUs meet SPDS requirements for Nuplex 80+ without using stand-alone monitoring and display systems. Alarms provide guidance to unexpected deviations in critical functions as well as success path unavailability or performance problems.

Alarm priorities are assigned based on the proximity of the alarm setpoint to a significant operator action condition. Section 18.7.1.1.4 provides a more detailed explanation of alarm priorities.

IPSO continuously provides overview information that is most useful for operator assessment of the critical functions, see Section 18.7.1.1. Each box within the matrix highlights the presence of alarms that threaten the specific critical function. The display format for the box indicates the highest priority of all related alarm conditions. Supporting information relating to critical function alarm conditions is available by using the alarm tiles on the critical functions section of the DPS display page hierarchy.

The critical function section of the display page hierarchy contains the following information:

- Level 1 Display Page

  This "Critical Function" overview page provides more detail on the critical function matrix presented on IPSO. More detail is provided on alarm conditions (descriptor) to help guide the operators to appropriate Level 2 Critical Functions display pages. A typical Level 1 Critical Functions display page is shown in Figure 18.7.1-13.

- Level 2 Display Page

  A second level page exists for each of the twelve critical functions. Each page contains:

  1. The critical function information provided on the 1st level display page that is associated with the critical function.

---

1 NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

4. A label titled "INJ FLOW CNTL" is placed above the Seal Injection Flow Controller.

5. A label titled "INJ TEMP CNTL" is placed above the Seal Injection Temperature Controller.

18.7.3.4 Alarm Layout

The alarm tiles for the RCS panel are contained on two flat panel display modules. The module used for the 16 Seal Bleed System alarms is located above the RCP and seal bleed system functional groups to the left of the DPS VDU. Figure 18.7.3-39 illustrates its location, and Figure 18.7.3-37 illustrates the detailed layout of the alarm tiles. This location places the alarms at the highest level within the RCP/Seal Bleed functional groups to enhance its attention getting function. The module used for the RCS and Operator Established, Alarms tiles which is located above the Pressurizer functional group to the right of the DPS VDU. Figure 18.7.3-39 illustrates its location, and Figure 18.7.3-36 illustrates the detailed layout of the alarm tiles.

- Identification of Functional Alarm Groups

  The RCP functional alarm group is identified by a label titled "RCP", placed above the RCP/Seal Bleed System Alarm Module. The Seal Injection System functional alarm group is identified by a label titled "Seal/Bleed", placed above the RCP Seal/Bleed Alarm Module. The RCS functional alarm group is identified by a label titled "RCS", placed above the RCS alarm module.

- Layout of Alarms Within Each Functional Group

  The alarms for the RCS are located on the right alarm module and arranged as shown on Figure 18.7.3-36. For cases where high and low alarm tiles exist for the same process (i.e., Pressurizer Pressure Hi and Pressurizer Pressure Lo), the high alarm tile is placed above the low alarm tile.

  The alarms for the RCP are located on the left alarm module and arranged as shown on Figure 18.7.3-37. They are placed within functional groups identical to the RCP indication and controls. There are four columns of RCP alarms, RCP1A, RCP1B, RCP2A and RCP2B. These columns of alarms are located left to right, as are the indication and controls. To help identify problems common to more than one RCP, similar alarms (i.e., cooling system) for the four RCPs are located in the same row.

  There are three alarm tiles for the RCP Seal Injection System. These are shown in Figure 18.7.3-37.

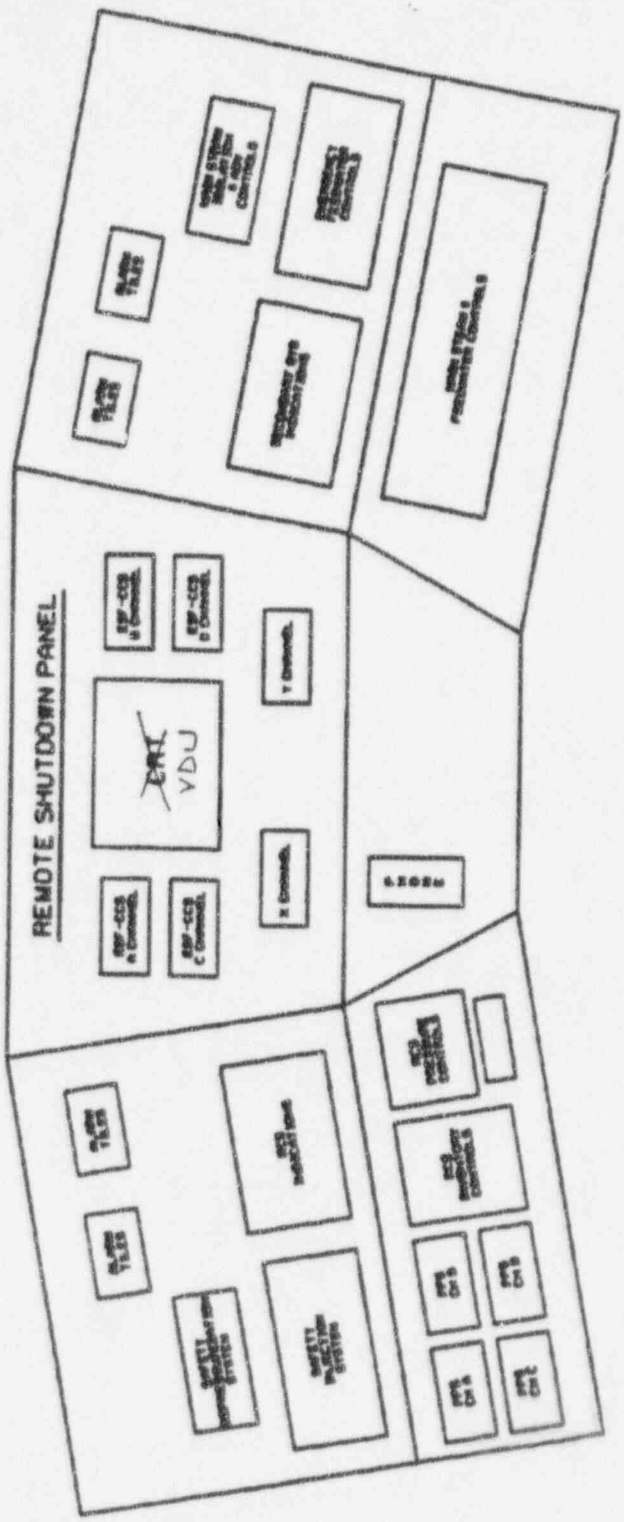
18.7.3.5 DPS VDU Layout

The ~~CRT~~ VDU is located near the center and close to the bottom of the vertical panel section, as specified in Section 18.7.2.3.1 (Figure 18.7.3-39 illustrates its location).

18.7.3.6 Miscellaneous Controls Layout

A lamp test switch is located in the upper center area of the apron section of the RCS panel. (Figure 18.7.3-39 illustrates its location).

The Operator Aid Alarm tile is provided on the vertical section of the RCS panel.



Replace

illegible

Remote Shutdown Panel Layout	Figure 18.8-1

Improvements have been made in the instrumentation of the SCS to provide the operator with more information about critical points in the system. The intent is to provide the operator with detailed system parameters so appropriate actions can be taken before the loss of DHR occurs. If a loss of the SCS does occur these parameters will aid in the correct and timely evaluation of the initiator thus decreasing SCS recovery time. Major new instruments which have been included in the System 80+ design are suction and discharge pressure indicators and SCS pump motor current indication. These instruments are all indicated in the main control room.

Suction piping arrangements have been simplified and improved. Several incidents have been attributed to the presence of loop seals in the suction lines that allow air to collect and lead to the reduction of NPSHA and air binding. System 80+ arrangements for the suction lines do not have loop seals and thereby enhance the ability of the pumps to survive low NPSHA conditions.

Improvements in AC power reliability are discussed in sub-section 2.4.3.2.

The System 80+ SCS design features presented above, and summarized in Table 2.4-1, provide a way to minimize a loss of the SCS. These design features also address initiators which are known to have defeated DHR systems in currently operating plants. A summary of these initiators and corresponding SCS design features are provided in Tables 1-1 and 1-2. |

1.4-1 1.4-2

2.4.3.1.3 Recovery from Initiators

Recognizing that some initiators may defeat the SCS, the System 80+ design will require that both SCS trains and one division of AC power be operable during Modes 5 and 6. This allows safety injection and containment spray equipment in the redundant division to undergo maintenance activities as necessary.

1.4-2 1.4-2

Table ~~1-2~~ provides a detailed listing of events that have resulted in the loss of shutdown cooling. Table 1-1 summarizes design features incorporated into System 80+ to prevent, detect and mitigate the effects of the events listed in Table 1-2. Consequently, a detailed listing of all potential initiators will not be provided in this section. Instead, initiators that result in the loss of DHR are categorized into four groups. This categorization is structured primarily to simplify the discussion but may also aid in constructing diagnostic loss of SCS procedures. These groups relate the initiator to a location in the system with respect to the SC pump. The instrumentation provided for monitoring the pump's performance identify whether the failure is in the suction line, discharge line, the pump itself, or a power failure. With proper diagnostic information from these groups, the operator can perform appropriate recovery actions to restore DHR. Table 2.4-2 identifies the groups, some representative initiators in each group, a brief description of the event and the instrumentation available to detect the event. The discussion that follows examines how DHR can be recovered using this information assuming a loss of a SCS train.

1.4-1

2.4.3.1.3.1 Group I Initiators

Group I initiators include a failure in the suction side of the SC pump. Suction line initiators are the most common during the midloop operation. These would include air ingestion, inadvertent closure of a valve in the suction line, failure of a relief valve to close, leakage from the system and procedural errors. The result of any of these initiators is to reduce the NPSHA for the SC pump.

Information provided to the operator in the control room for detecting and diagnosing these events include various alarms and indicators. The SCS includes an alarm for a low flow condition during shutdown cooling. This will be the initial indication of a possible suction line initiator since its set point is above

Procedural guidance will be provided to the COL applicant via the Emergency Operations Guidelines to ensure that the CVCS and BAMU pumps will not be made unavailable at the same time during Modes 5 and 6 reduced inventory to further enhance the capability to provide alternate RCS makeup.

Figure 2.4-2

Figure 2.4-3

Figure 2.4-4

Figures 2.4-2, 2.4-3, 2.4-4;
Safety Injection Piping and Instrumentation Diagrams.

(For these Figures, see Chapter 6, Figures 6.3.2-1A, 6.3.2-1B and 6.3.2-1C.)

2.5 Primary/Secondary Containment Capability and Source Term

2.5.1 Issue

This section addresses the ability of the containment to protect the public from the consequences of a release of radiation during the time the containment is open.

This issue is related to events initiated in Mode 5 or 6 which have the potential for radiological release. The events which will be considered are the loss of decay heat removal capability initiated by either a loss of shutdown cooling or by a loss of coolant caused by either operator error or a pipe break.

Following a loss of decay heat removal not the result of a pipe break, a radiological release from an open containment can occur when the time for the core to reach saturation is less than the time to restore RCS cooling and, failing this, the additional time to evacuate, close and isolate the containment. The time for the coolant to reach saturation is a function of plant conditions at the time the event is initiated.

Time to restore includes the time to detect that decay heat removal has been lost plus the time to restore either shutdown cooling or initiate alternate means of cooling. Time to detect depends on the instrumentation available to detect that Primary System cooling has been lost. The time to restore decay heat removal depends on the available systems and procedures.

Once Primary System cooling has been lost measures must be taken to evacuate and seal the containment before the system begins to boil. The time to close and isolate the containment depends on:

- Design, operation, condition and status of equipment to close penetrations, equipment hatches and personnel air-locks,
- Procedures for routing material and lines through these openings,
- Training of personnel
- Conditions of pressure, temperature and radiation within the containment as the core uncovers.

2.5.2 Acceptance Criteria

The following acceptance criteria apply to the issue addressed in this section:

1. Radiological exposure of the public to any event resulting in a loss of decay heat removal shall be limited to a small fraction of the limits stated in 10CFR100.

2. RCS Level below reduced inventory

Entry into or out of these operational conditions is controlled by procedures and Technical Specifications and require verification by the Senior Reactor Operator.

2.5.3.2.1.2.1 RCS Level Above Reduced Inventory

There are no Technical Specification requirements on containment integrity in Mode 5 when not in reduced inventory. Therefore proceeding from Mode 4 to Mode 5 does not require compliance with Technical Specifications dealing with containment integrity. It is during this mode of operation that equipment for maintenance and refueling outages, and support personnel, are moved into and out of containment through the one equipment hatch and two personnel locks. Also during this mode the surveillance testing of containment penetrations is completed and verified in accordance with site specific procedures.

2.5.3.2.1.2.2 RCS Level Below Reduced Inventory

In Mode 5 the RCS may be drained to facilitate installation of the steam generator nozzle dams as well as other maintenance items. Draining the RCS to a reduced inventory level (>3 feet below the reactor flange) requires (through the Containment Penetrations Technical Specifications) monitoring for any leakage of radiation through the penetrations. To maintain containment integrity, the equipment hatch and one of the two doors on each of the personnel locks must be closed during core alterations.

2.5.3.2.1.3 Mode 6

The potential for fuel handling accidents in Mode 6 establish the requirement that containment integrity be maintained. Thus the equipment hatch and one of the two doors on each of the personnel locks must be closed during core alterations.

Entry from Mode 5 to Mode 6 may require verification of containment penetration status. Since Mode 5 has two possible operational states, containment configuration must be satisfied and verified by the Senior Reactor Operator. Entry into Mode 6 from Mode 5 reduced inventory operation requires monitoring of containment penetrations for radiation leakage. Entry into Mode 6 from Mode 5, not at reduced inventory,requires verification of penetration status.

2.5.3.2.2 System 80+ Containment Features

2.5.3.2.2.1 Building Arrangement and Ventilation

The containment openings are surrounded by the Nuclear Annex Building. Therefore, there are no direct openings to the outside environment, and all leakage and air flow from containment openings (personnel locks, equipment hatch, open penetrations) is exhausted into the Nuclear Annex.

The Nuclear Annex Ventilation System (a non-safety grade system) draws air from various points in the Nuclear Annex and exhausts to the unit vent. If high radiation levels are detected by the system radiation monitor, the exhaust flow automatically aligns to a filter train. The filter train consists of particulate filters and carbon absorbers to remove radioactive material prior to exhausting into the unit vent.

The paths defined for the CVCS, if established, would be manageable with available make up sources during Modes 2, 3, 4, 5 or 6. This discharge would be slow enough such that a loss of primary coolant to the hot leg level should not occur before detection and mitigation have been accomplished. No new design features, technical specification or procedural guidance are identified for paths associated with the CVCS.

2.12.3.2.4 Potential Drainage Paths from the RCS to the SS

The potential drain paths from the RCS through the SS are shown in Figure 2.12-4. The potential drain paths presented in Figure 2.12-4 are normal sampling paths. A major opening in the sampling lines would need to occur for a net loss of primary coolant to occur.

The paths defined for the SS, if they were to occur, would be manageable with available make up sources during Modes 2, 3, 4, 5 or 6. The discharge would be slow enough such that a loss of primary coolant to the hot leg level should not occur before detection and mitigation have been accomplished. No new design features, technical specifications or procedural requirements are identified for paths associated with the SS.

2.12.4 Resolution

The shutdown risk issue of the potential for draining the System 80+ RCS is resolved primarily by design features, technical specifications and procedural guidance to prevent a drainage event from occurring and to allow the operator to recover in a timely manner if such an event occurs.

The vast majority of potential paths reviewed were judged to be minor such that the drain flow rate can be compensated using available detection and mitigating systems or are otherwise insignificant. System 80+ design features, technical specifications and procedural guidance are sufficient for such paths.

An examination of the potential drainage paths for various System 80+ plant arrangements and operating configurations has provided candidate paths, that if assumed to be opened, could lead to a rapid loss of primary coolant. The candidate paths primarily involve those opened by misoperation or misalignment of one or multiple valves by the operator. The importance of such potential major drainage paths to a shutdown risk scenario has led to procedural guidance (see Section 2.1) being specified to aid the operator in addressing these paths. Procedural guidance will be provided to the COL applicant via the Emergency Operations Guidelines in order to assure the availability of the *
The issue of potential RCS drain paths is ultimately resolved, from a core uncovery prevention perspective, by the use of System 80+ design features to detect, mitigate and recover from postulated loss of shutdown cooling events as described in Section 2.4 of Appendix 19.8A.

2.13 Flooding and Spills

2.13.1 Issue

Essential systems may be at higher risk for failure due to flooding and spills during shutdown because of the varied and interrelated maintenance activities that may be in progress simultaneously. Past events have involved, for example, spills from the component cooling water system, service water system, condensers, and refueling pool seals. The issue addressed here is the potential for loss of decay heat removal as a consequence of spills and internal flooding that may disable components of the shutdown cooling system.

* SCS, CSS and BAMU pumps to respond to a reactor cavity seal failure event.

Since the minimum transient DNBR for the events to which the events of Section 15.1.2 were referenced was 1.24, the consequences of events in shutdown modes are no more adverse than those of the events presented in this Safety Analysis Report. Document.

4.1.3 Increased Main Steam Flow

As noted in Section 15.1.3, the steam flow due to an increased main steam flow event is the same as (or less than) that due to an inadvertent opening of a steam generator relief or safety valve event. Further, there are no other differences between these events which affect their consequences. Therefore, the conclusions of Section 4.1.4 of this appendix also apply to an increased main steam flow event.

4.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Evaluation of the inadvertent opening of a steam generator relief or safety valve events postulated to be initiated in a shutdown mode shows that the results are less adverse than those of the events documented in Section 15.1.4.

This evaluation was completed as integral to the study of steam system piping failures postulated to be initiated in a shutdown mode. The transient caused by an inadvertent opening of a steam generator relief or safety valve is identical to that caused by a steam line break of area equal to that of a relief or safety valve. The study presented below covered a full spectrum of break sizes and initial conditions, including all and less than all reactor coolant pumps running. As documented below, the minimum transient DNBR for a steam system piping failure in a shutdown mode is greater than 2. The minimum transient DNBR for inadvertent opening of a steam generator relief or safety valve events postulated to be initiated in a shutdown mode is, therefore, also greater than 2. Since the minimum transient DNBR for the events of Section 15.1.4 was 1.24, the consequences of the events presented in Chapter 15 bound those of events in shutdown modes.

4.1.5 Steam System Piping Failures Inside and Outside Containment

Evaluation of steam system piping failures inside and outside containment, postulated to be initiated in a shutdown mode, shows that the results are less adverse than those of the events documented in Section 15.1.5.

The evaluation presented in this section focuses on fuel performance as measured by the departure from nucleate boiling ratio (DNBR) for verification of fuel integrity. Steam system piping failures are characterized by decreasing reactor coolant system (RCS) and steam generator pressures. Therefore the RCS pressure remains well below 110% of design pressure and the pressure-temperature limits for brittle fracture, ensuring that the integrity of the RCS is maintained, and the steam generator pressure remains below 110% of design pressure ensuring that the integrity of the secondary system is maintained. The two-hour inhalation dose at the EAB (Exclusion Area Boundary) is also examined to confirm that, if such an event were postulated to be initiated in a shutdown mode, the value meets the Standard Review Plan Acceptance Criteria of Reference 6.

For shutdown modes the reactor core is subcritical with power being generated only by decay heat. Substantial margins to DNB exist at the time of a postulated event initiation. Steam system piping failures would cause a decrease in the temperature of the reactor coolant and in the RCS and steam generator pressures. The decrease in the reactor coolant temperature would result in an increase in core reactivity due to the negative moderator temperature coefficient. If the cooldown were sufficiently large, a return to criticality followed by an increase in core power might occur. This could create a potential for

9. According to SRP Section 9.1.1, the failure of non-seismic Category I systems or structures located in the vicinity of the new fuel storage racks should not cause an increase in k_{eff} beyond the maximum allowable. Provide analysis that this condition is met or include in your application a commitment to the above condition as a design criterion.

Response 410.103

1. Although Regulatory Guide 1.13 pertains to the design of spent fuel storage racks, it is also used for the design of the new fuel racks. The applicable portions of the Regulatory Guide that are met are defined ~~by Paragraphs~~ in Section 9.1.1.1.~~A and 9.1.1.1.C.~~

2. The "facilities" associated with new fuel storage consist of the storage vault and the rack restraint system. The seismic category of other building components associated with handling fuel assemblies is noted in Table 3.2-1 (see response to NRC RAI 210.1).

3. The L-insert slots are provided in the wall of the fuel rack cavity (box) to permit the L-insert to be locked to the fuel cavity by its locking tab after it has been installed. The design of the locking tab and slot is such that the L-inserts can be remotely removed from the fuel racks, if required.

   The cell blockers are installed in the fuel racks before the fuel assemblies are placed in the fuel rack and before the pool is flooded. The design is basically two concentric tubes with end restraints that limit the engagement of the tubes in the rack cavity wall (to avoid protrusion into an adjacent fuel rack cavity). The tubes are collapsed, installed into the fuel rack cavity, expanded into the holes in the fuel rack cavity wall, then locked together with a captured pin. In this manner the cell blockers are positively locked to the fuel racks but can be remotely removed if desired.

   The new fuel inspection area is provided for the inspection of new fuel assemblies after they have been removed from their shipping container and before they have been placed in the fuel racks. It will contain a Seismic Category II inspection device to ascertain if the fuel assemblies meet the dimensional requirements for installation into the reactor vessel. Visual inspections will also be performed to check for shipping damage and to ensure that protective wrapping material has been removed.

4. The number of new fuel assemblies required for a 12 month refueling cycle, an 18 month refueling cycle, and a 24 month refueling cycle was evaluated. This evaluation disclosed that a 24 month cycle is controlling from the standpoint of the maximum number of new fuel assemblies required, i.e., 108. Since the rack structure is square, the minimum array to accommodate 108 fuel assemblies at a density of 50% is two 11 x 11 fuel rack modules or 121 fuel assemblies.

5. The fuel handling equipment located in the new fuel storage area meets the requirements of ANS 57.1. The new fuel racks meet the requirements of ANS 57.3.

6. The lifting capacity of the overhead crane that is used to remove new fuel assemblies from the new fuel rack is restricted by either adjusting the motor stall torque or using load limiting devices. (See ~~paragraphs~~ Sections 9.1.1.3.1.1.D and 9.1.4.2.1.7.B).

7. The new fuel racks are located at the opposite end of the fuel building from the spent fuel pool to eliminate the possibility of moving heavy loads near the new fuel storage area. (See response

to Question i). Using procedural guidance provided by the plant designer in Section 2.1 of Appendix 19.8A, the owner-operator will develop administrative controls to limit the size of the load that can be carried over the new fuel racks so that the design impact energy that the racks can absorb without affecting k_{eff} will not be exceeded.

8. The reference section of Section 9.1.1.3.1.1 that discusses potential moderators should be 9.1.1.3.1.2.~~B instead of 9.1.1.3.1.2.D. This change will be incorporated in the next submittal.~~

9. The new fuel storage racks are located in a concrete vault at the opposite end of the fuel building from the spent fuel pool area to preclude passage of the spent fuel shipping cask overhead crane over the racks during the handling operations associated with spent fuel inspection, handling, and shipping.

   This location minimizes the number of systems or structures located in the vicinity of the new fuel storage facility. All systems or structures in the vicinity will be designated as Seismic Category II to preclude their failure and entry into the new fuel storage area.

   The new fuel storage racks will be designed to limit the rack k_{eff} to 0.98 based on the postulated accident conditions and assumptions of ~~Paragraph~~ Section 9.1.1.3.1. Using the procedural guidance provided by the plant designer in Section 2.1 of Appendix 19.8A, the owner-operator will develop administrative controls to prohibit carrying any load over the loaded fuel racks whose impact energy, if dropped, will exceed the impact energy of the postulated dropped fuel handling tool or the combination of the dropped fuel handling tool and fuel assembly. The maximum impact energy shall be limited such that dropped loads do not change the K_{eff} of the fuel array to more than 0.98.

SPLB Question 410.104(d)

(d) According to SRP Section 9.1.2, the design of the spent fuel storage facility is acceptable if the integrated design is in accordance with General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety functions. Acceptance for meeting this criterion is based on conformance with Paragraph 5.4 of ANS 57.2. Provide the design features which satisfy GDC 63 and discuss compliance with Paragraph 5.4 of ANS 57.2.

Response 410.104(d)

(d) The design of the spent fuel storage facility meets the intent of Paragraph 5.4 of ANS 57.2. As an example, the facility incorporates monitoring systems to verify pool water temperature to ensure adequate fuel assembly cooling, radiation detectors to determine if radiation levels exceed predetermined setpoints and alarms to notify plant personnel of abnormal conditions.

SPLB Question 410.107

1. Evaluate the structural design features of the refueling cavity water seal that would preclude a leak or failure from occurring. Include the possibility of a fuel assembly or other structure dropping on the seal.

SRXB Question 440.37

Standard Review Plan 15.4.6 requires redundancy of alarms that alert the operator to an unplanned boron dilution event. Describe the redundant alarms available in each operating mode.

Response 440.37

The following pre-trip alarms are available for operational Modes 1 and 2: a high power or, under certain conditions, a high pressurizer pressure pre-trip alarm in Mode 1 or a high logarithmic power pre-trip alarm in Mode 2. Furthermore, a high RCS temperature alarm may also occur prior to trip. In operational Modes 3 through 6, a single failure proof boron dilution alarm will alert the operator to an unplanned boron dilution event.

In addition to the above mentioned alarms, there are also sampling and boronometer indications which would provide information in the case of a boron dilution event.

SRXB Question 440.49

Provide a discussion of the procedures and plant systems used to take the plant from normal operating conditions to cold shutdown conditions. This discussion should include, heat removal, depressurization, flow circulation, and reactivity control.

Response 440.49

The principal systems utilized in taking the plant from Mode 1, Power Operation, to Mode 5, Cold Shutdown are:

Reactor Coolant System
Feedwater System
Feedwater Control System
Reactivity Control System
Boron Control System
Chemical & Volume Control System
Shutdown Cooling System
Pressurizer Level Control System
Steam Bypass Control System
Pressurizer Pressure Control System
Liquid and Gaseous Waste Management Systems
Main Steam System
Condensate System

Reactivity control capability is discussed in Sections 7.7.1.1.1; 7.7.1.1.7; 9.3.4.1.3.3; 9.3.4.2.1 (last paragraph), and 9.3.4.2.3C. Power is reduced by increasing the boron concentration in the RCS to reduce k-effective to < 0.99. At low power the rods are inserted. The operator borates to the cold shutdown boron concentration consistent with the Technical Specifications prior to the beginning of cooldown. This margin is maintained throughout cooldown by making up shrinkage volume by means of the CVCS with water at the cold shutdown margin boron concentration.

12. Pressurizer water temperature should exceed RCS water temperature by no more than 350°F and no less than 50°F whenever there is a bubble in the pressurizer.

13. Auxiliary pressurizer spray is utilized to reduce pressurizer pressure whenever normal spray is inadequate or not available.

14. When the pressurizer pressure is approximately 400 psia and the RCS temperature decreases to 350°F, cooldown is transferred to the Shutdown Cooling System (SCS). Cooldown from this point is fully described in Section 5.4.7.2.6A. Steaming and feed may be terminated.

SRXB Question 440.54

Discuss the alarms and indications which would inform the operators that the SCS suction line isolation valve has closed while the plant is in shutdown cooling? Is there any common mode failure which would result in isolation valves in both trains being closed while in shutdown cooling? Are there any manual maintenance valves whose closure could isolate the SCS suction, if so, describe procedures and controls to restrict this possibility?

Response 440.54

The SCS suction line isolation valves are SI-651, SI-652, SI-653, SI-654, SI-655, and SI-656, as shown on ~~on CESSAR-DC~~ Figure 6.3.2-1C. Valves SI-651, SI-653, and SI-655 are in line with shutdown cooling pump 1. Valves SI-652, SI-654, and SI-656 are in line with shutdown cooling pump 2. There are several alarms and indications to inform the operators that a SCS suction valve has closed while the plant is in shutdown cooling:

- Valve position indications is provided in the main control room.
- If a suction valve were to close, a drop in SCS pump flow for the affected train would actuate a low shutdown cooling flow alarm.
- SCS pump current and pressure indicators in the control room would indicate a loss of flow.
- Valves SI-651, SI-652, SI-653, and SI-654 are alarmed when not fully open with concurrent low RCS temperature (below the LTOP enable temperature).

There is no interlock to automatically close the SCS suction valves if RCS pressure increases during shutdown cooling operation. The suction valve interlock with pressurizer pressure described in Section 5.4.7.2.3.A.2 only provides a permissive open signal to allow the operator to open the valves when aligning the SCS.

Electrical power assignments are as follows (all valves can be powered from the emergency diesel generators or the alternate AC power source):

Valves SI-651 and SI-655 are on electrical train A.
Valve SI-653 is on electrical train C.
Valves SI-652 and SI-656 are on electrical train B.
Valve SI-654 is on electrical train D.

SRXB Question 440.146

How did the plant design incorporate the consideration that outage and maintenance activities require only minimal isolation of important systems and components?

Response 440.146

The System 80+ Design utilizes divisional separation of important systems and components to ensure the ability of the design to comply with the following recommendations of NRC Generic Letter 88-17, "Loss Of Decay Heat Removal:"

(3) Equipment

- Assure that adequate operating, operable, and/or available equipment of high reliability is provided for cooling the RCS and for avoiding a loss of RCS cooling.
- Maintain sufficient existing equipment in an operable or available status so as to mitigate loss of DHR or loss of RCS inventory should they occur. This should include at least one high pressure injection pump and one other system. The water addition rate capable of being provided by each equipment item should be at least sufficient to keep the core covered.

["Reliable equipment is equipment that can be reasonably expected to perform the intended function."]

To further enhance divisional separation, the System 80+ Design utilizes four safety injection pumps to ensure the availability of at least one high pressure injection pump, as specified in (b) above. Additionally, the SIS design allows full flow pump testing without injecting water to the core, which minimizes RCS perturbations. The System 80+ Design also allows shutdown cooling system pumps to be interchangeable with containment spray pumps, thus allowing further redundancy and flexibility in shutdown operations.

Because of the complete divisional separation of important systems and equipment in the System 80+ Design, proper administrative control of maintenance activities together with the plant technical specifications will ensure that sufficient equipment and systems remain operational and available as required to maintain safe shutdown and refueling conditions.

SRXB Question 440.147

What design measures have been taken to ensure that demands on equipment during shutdown operations are consistent with equipment design operating range? You should consider the possibility of additional equipment which could be potentially useful during shutdown conditions if its control monitoring and operating parameters were appropriately selected during design.

Response 440.147

The System 80+ Design fulfills the regulatory requirements of NRC Generic Letter 88-17, "Loss Of Decay Heat Removal," regarding level and temperature instrumentation compatibility with shutdown and refueling conditions and parameters. Instrument ranges will be such as to ensure sufficient accuracy for plant shutdown/refueling modes. The following statement has been added (part added underlined) to Section 5.4.7.2.2.B, to clarify item "a."

"a. Two independent, highly reliable instruments are provided for RCS level management. These instruments function to monitor RCS level, to preclude SCS suction line vortexing and air entrainment. Level instrument ranges are optimized to encompass all reduced RCS inventory conditions."

One aim of the System 80+ Design is to utilize equipment for specific and unique purposes. An example of this is the use of the Shutdown Cooling Pumps for shutdown cooling alone, instead of using the pumps for shutdown cooling and low head safety injection. However, in instances where equipment serves both plant operating and non-operating functions, the equipment will be specified for procurement so that the full range of design requirements is met. Additionally, operating guidance will be provided to the plant owners to ensure cognizance of the operating envelope of this equipment.

In addition, the System 80+ design includes the following features to facilitate continued SCS operations during reduced RCS inventory:

- Two independent, highly reliable instrument systems (dP based and HJTC based) are provided for RCS level measurement. These instruments function to monitor RCS level, to preclude SCS suction line vortexing and air entrainment. Level instrument ranges are optimized to encompass all reduced RCS inventory conditions.

- Two independent thermocouples are provided to measure core exit temperature, with a large range optimized for SCS and refueling modes.

- Instruments which will monitor the state of SCS performance (such as pump suction pressure, vortexing monitoring equipment, flow instrumentation and/or pump motor current) are provided. These instruments function to sufficiently eliminate SCS pump loss events by monitoring the formation of vortexing and subsequent air entrainment.

- SCS suction isolation valves are not automatically closed in the event of an RCS pressurization during shutdown cooling. This precludes a loss of shutdown cooling by automatic closure of the isolation valves.

- The plant design provides other means of initiating alternate cooling for loss of SCS events. The plant design also ensures that a vent pathway is available to prevent pressurization.

SRXB Question 440.148

Will there be any maintenance activities for the System 80+ that will require isolation of IRWST pump suction inlets (or allow foreign material in the sump with potential for blockage)? If so, this would preclude operation of safety systems. What guidance can be provided to minimize this potential risk? Have TSs been provided limiting such maintenance activities?

Response 440.148

No maintenance activities that will require isolation of the IRWST pump suction inlets are possible because the inlets will be submerged during all modes of operation. Maintenance in the IRWST is only possible during mode 6, when IRWST inventory has been transferred to the refueling pool. During refueling operations, the Shutdown Cooling System pumps utilize the IRWST ECCS suction connections to fill the refueling cavity. Due to NPSH and vortexing considerations, the suction inlets are sufficiently submerged to protect the SCS pumps while the pumps are in operation.

leakage rate, the refueling cavity water level is maintained at its full height above the reactor pressure vessel flange by the available makeup capacity.

In the event that no refueling cavity makeup is available, the water level in the refueling cavity will decrease to the reactor vessel flange elevation in approximately four (4) hours. This is the minimum time based on the transfer tube valve being closed. If the valve is open, the drain time is substantially longer.

- Potential Consequences to Spent Fuel Being Transferred

The time to drain down the nine (9) feet of water over the top of an active fuel assembly being transferred with the refueling machine is approximately eighty (80) minutes. To preclude uncovering the fuel assembly, the assembly must be lowered below the reactor pressure vessel flange level in this time.

Loss of water depth in the refueling cavity is determined by the refueling cavity level alarm that is set two (2) inches below the nominal water level. The level monitoring system provides an indication of the water level down to the reactor pressure vessel flange elevation.

The fuel assembly may be either lowered into the reactor vessel or the end of the refueling cavity containing the transfer system upender and core support barrel (CSB) storage stand. Both of these locations provide sufficient water depth below the pool seal elevation to maintain water coverage over the fuel assembly. These two (2) areas are separated by a section of the refueling cavity that is at the elevation of the reactor pressure vessel flange. The raised section is about eleven feet long.

The refueling machine transit time over this area is less than thirty (30) seconds. The refueling machine can lower the fuel assembly below the reactor pressure vessel flange in approximately three (3) minutes in the slow speed range of the hoist. Therefore the eighty (80) minute drain down time (assuming no water makeup capability) is adequate to ensure the fuel assembly being transferred can be kept underwater in the event the pool seal develops the maximum credible leak.

Additional information on the permanent pool seal is contained in the Response to DSER Item 9.1.3-3.

SRXB Question 440.185

Appendix B, Reduced Inventory Operational Guidance, Operational Guidance 4.4, provides RCS/SCS system parameters monitored during reduced inventory. This operational guidance does not address SCS pump suction and discharge pressure as part of system parameters monitored during reduced inventory as mentioned in Section 2.8. Please clarify.

Response 440.185

As described in Section 5.4.7.2.2 Paragraph B.3, the SCS pump suction and discharge pressure can be monitored from the Control Room. These indications were inadvertently missing from Appendix B of the EOGs, Section 5.4.4 but will be used to provide information on the operation of the Shutdown Cooling System.

SRXB Question 440.194

Section 2.6, Rapid Boron Dilution, discusses some potential boron dilution events. NUREG/CR-0105, Vol. 2,"Seventeenth Water Reactor Safety Information Meeting", identifies several potential PWR boron dilution events that ABB-CE has not discussed in Section 2.6 of the submittal. Please provide the below discussions which emphasis in the use of design features, detection, mitigation, and prevention capability and relate these to resolutions in Table 2.6-1:

1. addition of diluted accumulator water during shutdown due to slow leakage or blowdown through single valve,

2. LOCA with diluted ECCS water from more than one accumulator or IRWST,

3. LOCA with sump water diluted,

4. uncontrolled boron dilution from CVCS during shutdown and the event of demineralized water from the purification system entered the core via the SCS system (the Belgians study),

5. rod ejection accident.

Response 440.194

A reassessment of the possible boron dilution events that need to be addressed in the System 80+ Shutdown Risk Analysis has been done to consider the five events in question. NUREG/CR-5368, Reactivity Accidents, provides the logic and assumptions that were used in a PRA that identified and quantified these five events along with several other events. The following responses are based on information found in NUREG/CR-5368 and on the System 80+ design. See also the response to Question 440.171.

1. The PRA done in NUREG/CR-5368 indicated that a critical boron dilution event due to the injection or leakage of diluted accumulator water as "incredible" events. This finding remains applicable to the System 80+ design. The design of the system used in the PRA is similar to the System 80+ design. The applicable Technical Specifications for the System 80+ design equal or exceed those assumed in the PRA analysis. The PRA assumed the shutdown margin could be as low as 1% while System 80+ design requires at least 6.5% shutdown margin. The PRA used 1380 ppm as the critically low boron value. Table 4.3-1 of Chapter 4 indicates that the System 80+ design will have a lower critical boron value of 1084 ppm.

   The PRA stated that the undetected filling of an accumulator with diluted water was not likely and this event would most likely be due to the leakage of water from the RCS to the accumulator. The System 80+ design would require nearly a 50% dilution of the accumulator in order to reach the low critical boron concentration. The Technical Specifications allow a maximum of 25% free volume in the accumulator. A 50% dilution of an accumulator would require the lifting of the relief valve on the accumulator. The dilution of an accumulator to a critically low boron concentration would require willful neglect of at least four parameters controlled by Technical Specifications. These parameters are: SIT pressure, SIT level, SIT level change and RCS leakage control. This boron dilution path is in Table 2.6-1. ~~Section A.1.b.~~

2. Review of NUREG/CR-5368 indicates that the LOCA with diluted ECCS water is a MODE 1 issue. Diluted water in the accumulator or IRWST during shutdown conditions was covered in

the PRA discussed in response (1). The LOCA with diluted ECCS water is not a dilution path that needs to be addressed in the shutdown risk analysis.

3. Review of NUREG/CR-5368 indicates that the LOCA with diluted sump water is an accident related long term cooling issue. The LOCA with diluted sump water is not a dilution path that needs to be addressed in the shutdown risk analysis.

4. The injection of pure water into the RCS via the CVCS is mentioned in Table 2.6-1 ~~Section D.e.~~ The dilution path of concern for this event would be from the reactor makeup water storage tank (RMWST) to the RCS.

   The PRA done in NUREG/CR-5368 indicates that boron dilution events associated with the purification system result in acceptable consequences. No resolution is required for this path.

5. Review of NUREG/CR-5368 indicates that the rod ejection is a reactivity accident but unrelated to boron dilution. The rod ejection accident is not a dilution path that needs to be addressed in the shutdown risk analysis.

SRXB Question 440.195

NUREG-1449 indicates that loss of coolant can result from the SCS pump suction relief valve opening. Please provide a discussion to address how your spring-loaded relief valve would not subject System 80+ to this vulnerability.

Response 440.195

The loss of coolant described by NUREG-1449 occurred at Braidwood 1 when an RHR pump suction relief valve opened at a pressure below its setpoint. The RCS was pressurized to 350 psig at the time. The event was made more severe due to an improper valve blowdown setting; this prevented the valve from reclosing.

In the System 80+ design, the SCS suction relief valves are used to provide low temperature overpressure protection (LTOP) for the RCS. Inservice testing and inspection of the valves will be conducted periodically in accordance with ASME XI, "Rules for Inspection of Nuclear Power Plant Components", and the ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants". This testing is intended to periodically reaffirm that the valves open within an acceptable tolerance of the specified setpressure. In addition, the plant designer will provide guidance through the Operational Support Information (OSI) Program (see Section 2.1.3 of Appendix 19.8A) to have the relief valves visually inspected during these test intervals to affirm that the blowdown setting is correct.

Each shutdown cooling system train in the System 80+ design is provided with its own suction line. There are no cross connections between the two trains. An LTOP relief valve is located in each shutdown cooling suction line, downstream of two remotely actuated isolation valves. An interlock with pressurizer pressure prevents these isolation valves from being opened by the operator when RCS pressure exceeds the shutdown cooling entry pressure; i.e., the pressure that would cause the LTOP relief valves to open. System 80+ Technical Specifications require that during low temperature conditions, both LTOP relief valves must be aligned to the RCS.

1.0 Introduction

This Appendix presents the basic steps used in determining the System 80+ reactor cavity ultimate static and dynamic pressure capacity including the corbels which support the reactor vessel. The loads used to determine an initial design and subsequently the ultimate capacity of the reactor cavity are obtained from the Ex-Vessel Steam Explosion event in Section 19.11.4.1.2.2.

2.0 Calculation Methodology

The following steps outline the approach and procedure used in determining the System 80+ reactor cavity ultimate static and dynamic pressure capacity including the corbels which support the reactor vessel.

1. Given a static ultimate capacity of 225 psi~~g~~d, the reactor cavity stresses, forces and moments are determined.

2. Reinforcing patterns are determined using ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures."

3. The actual ultimate static capacity is determined based on actual reinforcing steel provided: 235 psi~~g~~d

4. The actual design static pressure capacity based on actual reinforcing steel is determined using a typical load factor of 1.25, neglecting any other load combinations: 235/1.25=188 psi~~g~~d

5. The maximum resistance of the structure R_m, is determined by calculating a Dynamic Increase Factor (DIF) based on ACI 349, Appendix C. The reinforcing steel is assumed to take all of the pressure load. DIF = 1.1 R_m = 235(1.1) = 259 psi~~g~~d

6. The ductility ratio, μ_d, is determined using ACI 349, Appendix C. μ_d= 3.0

7. The reactor cavity is then checked to ensure that ~~then~~ when actual ductility can reach at least μ_d = 3.0. The reinforcing is carrying the total load and ~~a typical reinforcing is carrying the total load and~~ a typical reinforcing stress-strain curve is used.

8. The natural period of the structure, T, is determined by modal analysis using a finite element code. The concrete is assumed to be created below the reactor vessel support corbels, elevation 62'+0" to 73' +6".Above elevation 73' +6", the reactor cavity forms a complete hoop and is assumed to be uncracked. This is appropriate for the dynamic pressure loading since the pressure pulse in the water filled lower cavity will strike the walls before any loading is realized in the upper cavity area. T = 0.0114 seconds.

9. The dynamic pressure capacity of the reactor cavity, F_1, is determined by the following expression:

$$F_1 = \frac{R_M}{X}$$

Reference 1. [Biggs]

F_1 = Dynamic Pressure Capacity

R_m = Maximum Resistance of the Structure, 259 psi~~g~~d

X = Dynamic Load Factor (DLF), 0.90

The Dynamic Load Factor, X, is determined from charts of ductility ratio, μ, plotted against t_d/T. t_d is the load duration which was given to be 0.005 seconds. A rectangular shaped forcing function was given.

F_1 = 288 psi

10. The impulse capacity, I_{co}, of the cavity is determined based on a rectangular shaped forcing function.

$I_{co} = F_1 \times t_d$ = (288psi) x 0.005 sec = 1.44 psi - sec

11. The static pressure capacity of the reactor vessel support corbels is determined. The predicted reactor cavity water level is at elevation 79'+0" which is above the bottom of the corbels but below the reactor vessel. The pressure loadings are only a concern for the submerged structure where an in-liquid shock wave would propagate outward from the Fuel Coolant Interaction (FCI) steam explosion event. Above the water surface, the shock waves would propagate much slower and would be a lower magnitude. Therefore, the pressures on the structures and components above the water surface are not considered.

12. Forces and moments are determined on the corbels considering dead weight of the reactor vessel and the 235 psi~~g~~d static ultimate pressure capacity of the lower cavity applied to the bottom of the corbels.

13. Reinforcing patterns are determined using ACI-349. Additional reinforcing is required in the bottom of the corbels due to the Severe Accident upward forces only. This reinforcing is included in the System 80+ design.

14. The actual ultimate static capacity is determined based on actual reinforcing steel provided. 1,057 psi~~g~~d

15. The actual design static pressure capacity based on actual reinforcing steel is determined using a typical load factor of 1.25, neglecting any other load combinations. 1057/1.25 = 846 psi~~g~~d

16. The dynamic capacity of the corbels is determined in the same manner described for the reactor cavity.

t_d = duration of the event 0.005 seconds

R_m = Maximum Resistance of Corbels

= DIF(1,057 psi~~g~~d) = 1.1(1,057) = 1,163 psi~~g~~d

μ = 2.3 Average of ductility for concrete in shear (1.6) and reinforcing bars (3.0) in ACI 349

20.0 Closure of Unresolved and Generic Safety Issues

This chapter documents the technical resolution for all Unresolved Safety Issues (USIs) and Medium- and High-Priority Generic Safety Issues (GSIs) that are relevant to the System 80+™ (†) Standard Design as required by 10 CFR Part 52.47 for Design Certification.

After the System 80+ applicable issues were identified, a methodology for the documentation of the technical resolution of each issue was developed. Each applicable issue is comprised of three sections: ISSUE, ACCEPTANCE CRITERIA, and RESOLUTION. A fourth section, REFERENCES is provided where appropriate. The ISSUE statement section consists of a brief summary description of the safety issue. This is followed by the ACCEPTANCE CRITERIA section. These criteria are taken from NUREG-0933 in most cases, and, in the absence of a formal NRC resolution, developed from accepted industry codes, guidelines, standards and/or good engineering practice. The RESOLUTION section contains the technical resolution of the safety issue which is based upon the System 80+ Standard Design as described in this report or other pertinent documentation as listed in the REFERENCES section. This structure is intended to establish a clear and concise technical resolution for each safety issue.

20.1 NRC List of Unresolved Safety Issues and Generic Safety Issues

Unresolved and Generic Safety Issues were evaluated for their applicability to the System 80+ Standard Design based on the review of NRC and industry documentation (e.g., NUREG-0933 "A Prioritization Of Unresolved and Generic Safety Issues" and NUREG-1197, "Advanced Light Water Reactor Program, Management Review Methodology"). Section (a)(1)(iv) of 10 CFR 52.47 requires technical resolutions of those USIs and medium- and high-priority GSIs identified in the version of NUREG-0933, current on the date six months prior to application and that are technically relevant to the design. All USIs and GSIs identified through Supplement 15 of NUREG-0933, issued in April 1993, were reviewed in accordance with the guidance of NUREG-0933.

All USIs and GSIs that have been reviewed are listed in Table 20.1-1; the results of the review are presented for each issue as either Category 1 (not relevant) or Category 2 (relevant to the System 80+ Standard Design).

Issues were eliminated as not relevant to the System 80+ Standard Design if it met one or more of the following criteria:

- The issue is prioritized in NUREG-0933 as DROPPED or LOW, or the issue has not yet been prioritized. (Category 1a)
- The issue is specific to another design (e.g., General Electric BWR, Westinghouse, Babcock and Wilcox). (Category 1b)
- The NRC identified the issue as resolved with no new requirements and no references to old requirements. (Category 1c)
- The NRC identified the issue as either an operational, environmental, licensing, or NRC internal issue. (Category 1d)

† System 80+ is a trademark of Combustion Engineering, Inc.

Table 20.1-1 Listing of Unresolved Safety Issues and Generic Safety Issues (Cont'd.)

Number	Title	Type	Category
II.J.3.1	Organization and Staffing to Oversee Design and Construction	GSI	2
II.J.3.2	Management for Design and Construction - Issue Reg. Guide	GSI	1e
II.J.4.1	Revise Deficiency Report Requirements	GSI	1d
II.K.1 (1,2,4[a-c],7,8,11-13,17-23)	Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents - IE Bulletins 11,12,	GSI	1
II.K.1 (3,4d,5,6,9,10,14-16,24-28)	Measures to Mitigate Small Break LOCAs and Loss of Feedwater Accidents - IE Bulletins 13-16	GSI	2
II.K.2 (1-15,18,20,21)	Commission Orders on B&W Plant to Mitigate Accidents	GSI	1b
II.K.2 (16,17,19)	Commission Order on B&W Plants to Mitigate Accidents	GSI/TMI	1b
II.K.3 (1,3,4,7,9-24,26-29,32-54,56,57)	Final Recommendations of Bulletins and Orders Task Force to Mitigate Accidents	GSI	1
II.K.3 (2,5,6,8,25,30,31,55)	Final Recommendations of Bulletins and Orders Task Force to Mitigate Accidents	GSI/TMI	2
III.A.1.1 (1,2)	Upgrade Emergency Preparedness	GSI	1g
III.A.1.2 (1-3)	Upgrade Licensee Emergency Support Facilities	GSI/LI	2
III.A.1.3 (1)	Maintain Supplies of Thyroid Blocking Agent	GSI	1g
III.A.1.3 (2)	Maintain Supplies of Thyroid Blocking Agent	GSI	1g
III.A.2.1 (1-4)	Amendment to 10 CFR 50 and Appendix E	GSI	1d
III.A.2.2	Development of Guidance and Criteria	GSI	1g
III.A.3.1 (1-5)	Emergency Preparedness--NRC Role in Responding to Nuclear Emergencies	GSI/LI	1d
III.A.3.2	Emergency Preparedness--Improve Operations Centers	GSI/LI	1d
III.A.3.3 (1,2)	Emergency Preparedness--Communications	GSI/LI	1d
III.A.3.4	Nuclear Data Link	GSI	1d
III.A.3.5	Emergency Preparedness--Training, Drills & Tests	GSI/LI	1d

Resolution

The System 80+ Standard Design, as described in Chapter 8.0, does not have direct manual or automatic ties between the two Class 1E 4160 VAC power systems. Also, double breakers are provide to maintain independence between the Class 1E and the Permanent Non-Safety 4160 VAC buses. d

These breakers are provided for abnormal scenarios such as Loss-Of-Offsite Power and Station Blackout when it is necessary to isolate the Division I & II 4160 VAC buses from the Permanent Non-safety buses. The Reserve Auxiliary Transformers are capable of supplying power to their respective safety divisions only, thereby maintaining required independence. No single failure can prevent operation of the minimum number of required safety loads. (See Sections 8.3.1.2.1, 8.3.1.2.3 and 8.3.1.2.5 for a discussion of compliance). Operating and Quality Assurance procedures governing the engagement/disengagement of the tie breakers are the responsibility of the Owner-operator. The electrical systems meet the intent of the guidelines identified in Regulatory Guide 1.6 and IEEE Standard 308. As required by 10 CFR 50, Appendix A (GDC 17), the design of the power systems provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature systems. Since the guidance and requirements are met, this issue is resolved for the System 80+ Standard Design.

References

1. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission.

2. 10 CFR 50 Appendix A, "General Design Criteria."

20.2.13 Improving the Reliability of Open Cycle Service Water Systems

Issue

Generic Safety Issue 051 identified the susceptibility of the Station Service Water System (SSWS) to fouling which leads to plant shutdowns and reduced power operation for repairs.

The SSWS cools the Component Cooling Water System (CCWS) through the Component Cooling Water Heat Exchangers and rejects the heat to the ultimate heat sink during normal, transient, and accident conditions. The CCWS in turn provides cooling water to those safety-related components necessary to achieve a safe reactor shutdown, as well as to various non-safety reactor auxiliary components.

Acceptance Criteria

The acceptance criterion for the resolution of GSI 051 is that the design of the SSWS shall minimize the potential for fouling of the piping and heat exchangers. This minimization is achievable by: (1) reducing the number of components which are directly cooled by the SSWS; (2) employing site-specific corrosion-resistant materials and filtration systems which are consistent with the site water chemistry and treatment; (3) using heat exchangers with an enhanced thermal margin.

Resolution

The System 80+ Standard Design SSWS and CCWS are described in, Sections 9.2.1 and 9.2.2. The SSWS is designed to serve one Nuclear Steam Supply System (NSSS), and each NSSS on a multi-unit site will have its own SSWS.

the maximum flow to a steam generator to 800 gallons per minute, permit a longer operator response time to manually prevent overfill in the System 80+ Standard Design compared to current plants (see Sections 10.4.9.1 and 10.4.9.2).

Finally, the main steam lines are designed for a water filled load under static loading conditions, preventing failure of the lines and supports in the remote event that overfill does occur (see Section 10.3.2.3). The System 80+ Standard Design conforms to existing NRC requirements and guidelines (including the resolutions to the issues listed for Task 3) related to the avoidance of steam generator and steam line overfill. Additional design features such as the Safety Depressurization System, automatic termination of feedwater on high steam generator water level, and longer allowable operator response time minimize the probability of overfill in a SGTR event. In summary, because the plant capabilities are consistent with the goals of the NRC's tasks in GSI 135, this issue is therefore resolved for the System 80+ Standard Design.

References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition," U.S. Nuclear Regulatory Commission.

20.2.46 Leakage through Electrical Isolators in Instrumentation Circuits

Issue

Generic Safety Issue 142 concerns electronic isolators used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, thereby preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class 1E safety-related systems are transmitted to non-Class 1E control or display equipment.

qualify

There are a number of devices which may quality as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible fault applied in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuit (Class 1E side) below an acceptable level.

Recent observations have shown instances in which isolation devices subjected to failure voltages and/or currents less than maximum credible fault levels passed significant levels of voltage or current, but the same devices performed acceptably at maximum credible levels. The safety system on the Class 1E side of the isolation device may be affected by the passage of small levels of electrical energy, depending upon the design and function of the safety system.

Acceptance Criteria

The assumed solution to this issue would require the staff to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. A NRC bulletin to all licensees to provide input on these questions would be necessary. Assuming that the staff determines from the licensee responses to the proposed bulletin that a potential problem exists, a research program consisting of two major objectives would have to be initiated to develop the solution to this issue. The first objective would be to develop test procedures and acceptance criteria for isolators

Acceptance Criteria

The acceptance criteria for the resolution of these sub-issues identified in GSI II.K.1 are encompassed by the other Generic and Unresolved Safety Issues which are given in NUREG-0933. In general, these criteria provide that plant design and operation adequately address both small-break LOCAs and loss-of-feedwater events in accordance with the guidance given in NUREG-0737.

Resolution

GSI II.K.1 is a comprehensive issue covering a broad range of safety aspects of both plant design and emergency procedures. Each sub-issue applicable to the System 80+ Standard Design is identified in the following list and cross-referenced to the particular USI and/or GSI which specifically addresses the sub-issue. The remaining sub-issues of GSI II.K.1 are not applicable to the System 80+ Standard Design.

II.K.1(5): This sub-issue was resolved for the System 80+ design by the post-TMI upgrade of ABB-CE Emergency Operating Guidelines (CEN-152), by incorporating CEN-152 into the System 80+ EOGs, and by reviewing those EOGs in light of current design basis safety analysis and severe accident analysis.

II.K.1(10): This sub-issue was resolved for the System 80+ design during development and review of the surveillance requirements and corresponding actions in the Technical Specification (Chapter 16.

II.K.1(13): This issue was resolved during development and NRC review of the System 80+ Technical Specifications.

Resolutions for the applicable sub-issues of GSI II.K.1 are subsumed by the individual USIs and/or GSIs, each of which is separately resolved and included in this chapter. Therefore, Issue II.K.1 is resolved for the System 80+ Standard Design.

The following list cross-references GSI II.K.1 sub-issues to other GSI/USIs which are applicable to the System 80+ Standard Design.

GSI II.K.1 Sub-Issue	GSI/USI Cross-References Applicable to System 80+
3	I.C.1
4d	II.F.2
6	II.E.4.2
9	II.E.4.2
14	II.E.4.1, II.F.1
15	II.E.1.2
16	I.C.1, II.D.3
24, 25	I.C.1
26	I.C.1
27	I.C.1, II.F.2
28	II.K.3(5)

The RCS temperature response during natural circulation will usually be slow 5-15 minutes as compared to a normal forced flow system response time of 6-12 seconds, since the coolant loop cycle time will be significantly longer.

When single phase circulation is established in at least one loop, the RCS indicates all of the following conditions:

a. Loop ΔT (T_H - T_c) less than normal full power ΔT,

b. Cold leg temperatures constant or decreasing,

c. RCS at least subcooled (verifies single phase flow),

d. No abnormal differences between T_H RTDs and core exit thermocouples. Hot leg RTD temperature should be consistent with the core exit thermocouples. Adequate natural circulation flow ensures that core exit thermocouple temperatures will be approximately equal to the hot leg RTD temperatures.

If the criteria listed above are not satisfied, then the operators should ensure that RCS pressure and inventory, and SG steaming and feeding, are being controlled properly.

7. RCS Heat Removal is next in priority because the parameters associated with it are concerned mostly with steam generators, which are the primary means of removing heat from the RCS. Furthermore, steam generator level and pressure also have the potential for rapid change. Instruction step a) is to ensure the presence of an operable steam generator for removing heat. The steam generator level may briefly transit below the narrow range steam generator level indication. Emergency feedwater flow will be initiated automatically from either the Plant Protection System or the Alternate Protection System on low steam generator level or can be initiated manually by the operator. RCS average loop temperature (criterion b) in the range of [551 to 562°F] is indicative that steam generators are adequately removing heat. Instruction Step c) also ensures an operable steam generator for controlled removal of heat. The steam generator pressure range given provides the expected range maintained by the steam bypass control system. The upper steam generator pressure limit, [1150 psia] is below the MSSVs setpoint and the lower limit, [1050 psia] would be indicative of an excessive cooldown. The contingency actions relate to feed and or steam flow to the steam generator under one of two conditions:

a. RCS heat removal is <u>NOT</u> sufficient (RCS T_{ave} > [562°F])

<u>or</u>

b. RCS heat removal is excessive (RCS T_{ave} < [551°F])

If RCS Heat removal is not sufficient (e.g., RCS T_{ave} > [562°F] due to loss of condenser vacuum), the operator is provided several items to check in order to re-establish RCS heat removal. Feedwater must be supplied to at least one steam generator in order to ensure adequate heat removal will be maintained. If the turbine bypass system is not functioning properly in automatic, the operator should attempt to take manual control to restore RCS T_{ave} to [551 to 562°F]. If manual control of the turbine bypass system is not possible or condenser vacuum is lost, then the atmospheric dump valves are operated to control T_{ave} between [551 and 562°F].

SAFETY FUNCTION STATUS CHECK
REACTOR TRIP RECOVERY

Safety Function	Acceptance Criteria
1. Reactivity Control	1. a. Reactor power decreasing and b. Negative Startup Rate and c. Maximum of 1 CEA NOT fully inserted or RCS borated per Tech Specs.
2. Maintenance of Vital Auxiliaries (AC and DC Power)	2. a. All vital Division I [4.16 kV AC], [125 V DC], and [120 V AC] Distribution Centers energized, or All vital Division II [4.16 kV AC], [125 V DC], and [120 V AC] Distribution Centers energized. and b. Non-safety load [13.8 KV] Bus X energized or Non-safety load [13.8 KV] Bus Y energized and c. Non-safety load [4.16 KV] Bus X energized or Non-safety load [4.16 KV] Bus Y energized and

add Title to continuation each page

SAFETY FUNCTION STATUS CHECK
REACTOR TRIP RECOVERY (Continued)

Safety Function	Acceptance Criteria
2. (Continued)	d. Permanent Non-safety load [4.16 KV] Bus X energized <u>or</u> Permanent Non-safety load [4.16 KV] Bus Y energized
3. RCS Inventory Control	3. a. Pressurizer level is [2% to 78%] <u>and</u> b. Charging and letdown are restoring pressurizer level to [33% to 52%] <u>and</u> c. The RCS is subcooled <u>and</u> d. No reactor vessel voiding as indicated by the HJTC RVLMS.
4. RCS Pressure Control	4. a. Pressurizer pressure is: i) [2160-2370 psia] <u>and</u> ii) trending to [2225 to 2300 psia] <u>and</u> b. Pressurizer heaters and spray are controlling pressure within P-T limits of Figure 4-1.
5. Core Heat Removal	5. a. T_H - T_C is less than [3°F] <u>and</u> b. The RCS is subcooled.

SAFETY FUNCTION STATUS CHECK BASES
REACTOR TRIP RECOVERY (Continued)

	Safety Function	Acceptance Criteria	Bases
4.	RCS Pressure Control	a. Pressurizer pressure is: i) [2160-2370 psia] <u>and</u> ii) trending to [2225-2300 psia] <u>and</u> b. Pressurizer heaters and spray are controlling pressurizer pressure within the P-T limits of Figure 4-1.	The lower value of [2160 psia] corresponds to the RCS low pressure alarm setpoint. The higher value of [2370 psia] is the high pressure alarm setpoint. Pressurizer pressure for an uncomplicated reactor trip is expected to fall within this range. Operation of pressurizer heaters and spray should be capable of maintaining pressurizer pressure within [2225-2300 psia] and within the Post Accident P-T limits of Figure 4-1.
5.	Core Heat Removal	a. The RCS loop $\Delta T(T_H-T_C)$ is less than [3°F] <u>and</u> b. The RCS is subcooled based on T_H RTD temperature.	Best estimate analysis shows that SG ΔT will be less than [3°F] in the steaming loop with RCPs running. Subcooled margin assures adequate core cooling while also accounting for temperature variations in the RCS.
6.	RCS Heat Removal	a. i) At least one steam generator has level within normal level band with feedwater available to maintain level <u>or</u>	Adequate RCS heat removal will be maintained if at least one steam generator is available for removing heat (capable of steam flow and feed flow). The value of [500 gpm total feedwater flow] is sufficient feed flow to remove decay heat (approximately 2% rated thermal power) from the core. Decay heat levels may not be high enough to require 500 gpm] feed flowrate. In this case, steam generator levels in the normal band satisfies RCS heat removal.

OPERATOR ACTIONS
LOSS OF COOLANT ACCIDENT (Continued)

Instructions	Contingency Actions
55. (Continued)	
c. RCS pressure ≤ [450 psia]	c. If depressurization of the RCS to the SCS entry pressure is still not possible, and voiding is suspected to exist in the steam generator tubes, then attempt to eliminate the voiding by: i) cool the suspected steam generator (by steaming and/or blowdown, and feeding) to condense the steam generator tube void, and ii) monitor pressurizer level for trending RCS inventory.
d. RCS T_H ≤ [400°F], e. RCS activity level within plant specific limits Then exit this guideline and initiate SCS operation per operating instruction. Include any special precautions or procedure modifications from the Plant Technical Support Center or Plant Operations Review Committee.	d. If depressurization of the RCS to the SCS entry pressure is still not possible, then attempt to eliminate the voiding by: i) operate the pressurizer vent or the Reactor Coolant Gas Vent System to clear trapped non-condensible gases, and ii) monitor pressurizer level and/or the HJTC RVLMS for trending of RCS inventory.

The LOCA Guideline has accomplished its purpose if the plant is in a condition where all of the Safety Function Status Check acceptance criteria are being satisfied, and the RCS is either in long term core cooling (i.e., recirculation through the SIS), the break has been isolated, or SCS entry conditions are satisfied. Further recovery actions must be identified by the [Plant Technical Support Center].

END

OPERATOR ACTIONS
EXCESS STEAM DEMAND (Continued)

Instructions		Contingency Actions
*17.	(Continued) and e. take steps to have the H_2 recombiners made available and aligned for use.	
*18.	If containment spray system is operating and containment pressure is less than [5.5 psig], Then containment spray may be terminated. Upon termination, the CSS must be aligned and reset for automatic operation or manual restart.	18. Continue containment spray system operation.
*19.	If the containment hydrogen concentration is greater than or equal to 0.5%, Then operate the hydrogen recombiners.	19.
*20.	If containment hydrogen concentration is less than [0.5%], Then terminate operation of hydrogen recombiners.	20.
*21.	If no RCPs are operating, Then verify single phase natural circulation flow in at least one loop by ALL of the following: a. loop ΔT (T_H - T_c) less than normal full power ΔT, b. hot and cold leg temperatures constant or decreasing, c. RCS is subcooled based on Representative CET temperature (Figure 7-1), d. no abnormal difference [greater than 10°F] between T_H RTDs and representative CET temperature.	21. Ensure proper control of steam generator feeding and steaming (refer to steps 10 and 11) and RCS inventory and pressure control (refer to steps 14 and 15).

* Step Performed Continuously

SAFETY FUNCTION STATUS CHECK
FUNCTIONAL RECOVERY GUIDELINE (Continued)

4. RCS Pressure Control Safety Function

Success Path Currently In Use	Acceptance Criteria
PC-1: Pressurizer Heaters and Spray	Pressurizer pressure is within the Post Accident P-T limits of Figure 11-1.
PC-2: CVCS	Pressurizer pressure is within the Post Accident P-T limits of Figure 11-1.
PC-3: SIS	The available charging pump is operating and the SIS pump is injecting water into the RCS per Figure 11-3 (unless SIS termination criteria met.
PC-4: Forced Circulation with Controlled Steaming	Pressurizer pressure is within the Post Accident P-T T limits of Figure 11-1.
PC-5: Natural Circulation with Controlled Steaming	Pressurizer pressure is within the Post Accident P-T T limits of Figure 11-1.
PC-6: SDS	a. Pressurizer pressure is: i) less than [2360 psia] and constant or decreasing and ii) within the Post Accident P-T limits of Figure 11-1].[1]

Keep together

[1] RCS Subcooling is NOT applicable when RDS valves are open.

OPERATOR ACTIONS
REACTIVITY CONTROL SAFETY FUNCTION

Success Path: Boration using SIS, RC-3

Instructions	Contingency Actions
1. Maintain RCS temperature constant (if possible), until reactivity control is established, in order to prevent power increases following the initial transient.	1.
* 2. If pressurizer pressure ≤ SIAS setpoint or containment pressure ≥ [2.7 psig], Then verify an SIAS is actuated.	2. If pressurizer pressure ≤ SIAS setpoint or containment pressure ≥ [2.7 psig] and an SIAS has NOT been initiated automatically, Then manually initiate an SIAS.
* 3. Ensure maximum safety injection and charging flow to the RCS by the following:	3. If safety injection and charging flow not maximized, Then do the following as necessary:
a. start idle SIS pumps and verify SIS flow in accordance with Figure 11-3 and	a. ensure electrical power to valves and pumps,
b. start charging pump.	b. ensure correct SIS valve lineup,
	c. ensure operation of necessary auxiliary systems.
* 4. If high RCS pressure is preventing SIS pump injection of boric acid, Then attempt to cooldown/depressurize to obtain adequate SIS flow (Refer to the Pressure Control and Heat Removal success paths in use).	4.
* 5. If the SIS is operating, Then the SIS may be throttled or stopped, one pump at a time, if All of the following are satisfied:	5. Continue SIS operation
a. Reactor power less than [$10^{-(8)}$%] and constant or decreasing,	No Parenthesis

* Step Performed Continuously

RC-3

OPERATOR ACTIONS
REACTIVITY CONTROL SAFETY FUNCTION (Continued)

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Success Path: Boration using SIS, RC-3 (Continued)

Instructions	Contingency Actions
*5. (Continued)	
or	
Adequate shutdown margin established per Technical Specifications and reactor power constant or decreasing,	
and	
b. RCS subcooled based on Representative CET temperature (Figure 11-1),	
and	
c. pressurizer level greater than [14.3%] and not decreasing,	
and	
d. at least one steam generator is available for removing heat from the RCS (ability for feed and steam flow),	
and	
e. the HJTC RVLMS indicates a minimum level at the top of the hot leg nozzles.	
* 6. If criteria of step 5 can NOT be maintained after SIS pumps throttled or stopped, Then appropriate SIS pumps must be restarted and full SIS flow restored.	6.

Bases for RCS Pressure Control

The purpose of maintaining RCS Pressure Control is to maintain the RCS inventory in a subcooled condition to provide an adequate cooling medium for the core, and to prevent the loss of inventory out of a relief valve with subsequent release of radioactive liquid to the containment and possibly to the atmosphere. Controlling RCS pressure within the Post Accident P-T limits of Figure 11-1 is also desirable to minimize the potential for pressurized thermal shock.

There are many conditions that could cause a loss of pressure control. A breach in the RCS piping, a stuck open relief valve, failure of the PPCS, loss of heat sink, or the failure of CEAs to insert during a reactor trip condition are some examples of ways that RCS pressure control can be lost. Pressure Control is closely related to RCS Inventory Control, and RCS and Core Heat Removal. Changes in inventory will generally result in RCS pressure changes and excessive RCS pressure may prevent introduction of makeup water to the RCS. Similarly, the maintenance of an adequate cooling medium around the core for core heat removal is dependent on maintaining subcooling. If there is a conflict between maintaining adequate core cooling and complying with the pressure/temperature limits of Figure 11-1, then maintaining of adequate core cooling will be given the higher priority. Subcooling of 20°F has precedence over PTS considerations. Pressure control may be accomplished by any of the following methods:

PC-1: RCS Pressure Control via Pressurizer Heaters and Spray

PC-2: RCS Pressure Control via CVCS

PC-3: RCS Pressure Control via SIS

PC-4: RCS Pressure Control via Forced Circulation with Controlled Steaming

PC-5: RCS Pressure Control via Natural Circulation with Controlled Steaming

PC-6: RCS Pressure Control via Rapid Depressurization System

PC-7: RCS Pressure Control via Rapid Depressurization System during SGTR.

The bases for the recovery actions required for implementing each of the methods listed above are detailed ~~as follows~~: on the following pages.

OPERATOR ACTIONS
RCS AND CORE HEAT REMOVAL SAFETY FUNCTION (Continued)

Success Path: Forced Circulation, No SIS Operation; HR-1 (Continued)

Instructions	Contingency Actions
22. (Continued)	d. attempt to establish an alternate, low pressure feedwater source to at least one SG.
*23. <u>Verify</u> adequate RCS heat removal via the SGs by: a. at least one SG has wide range level [> 0%], <u>and</u> b. RCS T_c temperatures are stable or decreasing.	23. a. <u>When</u> at least one primary safety valve has opened following a SG dryout, <u>Then</u> go to RCS and Core Heat Removal success path HR-4 and initiate once-through-cooling.
*24. If main, startup or emergency feedwater is restored, <u>Then</u> modulate feedwater flow rate as necessary to restore and maintain SG water level in the normal band.	24.

Acceptance Criteria for Success Path HR-1:

1. RCS and Core Heat Removal is satisfied if:

   a. At least one SG (or the unisolated SG) has level:

      i) within the normal level band with feedwater available to maintain level.

      <u>or</u>

      ii) being restored by main, startup or emergency feedwater flow and SG level is increasing

   <u>and</u>

   b. $T_H - T_c < [3°F]$ and not increasing

   <u>and</u>

   c. $T_{ave} < [562°F]$ and not increasing

* Step Performed Continuously

Bases for RCS and Core Heat Removal

The purpose of the RCS and Core Heat Removal safety function is to remove the decay heat generated in the core and transfer it to the RCS fluid, where it can be transferred to the secondary system or to some other heat sink.

To achieve control of RCS and Core Heat Removal, and to continually provide a heat sink for residual heat removal, the following methods are available:

HR-1: RCS and Core Heat Removal via Forced Circulation, No SI Operation
HR-2: RCS and Core Heat Removal via Natural Circulation, No SI Operation
HR-3: RCS and Core Heat Removal via SG Heat Sink with SI Operating
HR-4: RCS and Core Heat Removal via Once-Through-Cooling
HR-5: RCS and Core Heat Removal via Shutdown Cooling System

The bases for the operator actions required to implement the above success paths are detailed ~~as~~ on the follow~~s~~ing pages.

Appendix A

Severe Accident Management Guidance

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Appendix B

Lower Mode Operational Guidance

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1.0 Introduction

This Appendix provides the basis for the Combined Operating License (COL) applicant to develop plant specific procedures for responding to events initiated from the shutdown modes. It is based upon the evaluations in Appendix 19.8A on shutdown risk. Insights and guidance developed from those evaluations were presented in Appendix 19.8A. This information is collected and summarized in this Appendix as Lower Mode Operational Guidance (LMOG).

The LMOGs include guidance for responding to events initiated from Modes 1 through 4. Two levels are provided. Optimal Recovery Guidelines (ORGs) apply when a specific event is identified and for which specific recovery sequences have been formulated. Functional Recovery Guidelines (FRGs) apply when a specific recovery sequence cannot be identified or when it becomes ineffective.

The EOGs may be entered from the critical Modes 1 and 2. They may also be entered from shutdown Modes 3 and 4 when SIAS has not been blocked and LTOP has not been initiated. The EOGs are typically exited when the safety functions are satisfied and after shutdown cooling has been initiated.

This LMOG may be entered from a shutdown mode following an event initiated from shutdown. It would not typically be entered directly from the EOGs because exit from the EOGs requires that LTOP has been initiated and that would imply that shutdown cooling has been established. An exception might be a situation where following an event initiated from a higher mode all success paths have been accomplished, but the event has resulted in some deficiency in components or systems for shutdown cooling that can be mitigated by the guidance in this Appendix.

2.0 Interface Between LMOG and EOGs

INSERT # Attached. ADD Redline

3.0 Content of LMOG Appendix

The content of this Appendix is generally consistent with the intent of the Safety Functions that are typically dominant during shutdown events. They are:

1. Reactivity Control - events that reduce boron concentration or cause CEA withdrawal.
2. RCS Inventory Control - events that drain the RCS or that cause loss of control of RCS inventory (such as during midloop operations).
3. RCS Heat Removal - events that cause loss of the shutdown cooling system capability.
4. Containment Integrity - events that cause radiological release directly out of an open containment, as during an outage, or indirectly through systems that interface with the RCS.

The particular events that challenge these safety functions may be somewhat different in detail than events initiated from the critical power modes. In the shutdown risk evaluations reported in Appendix 19.8A the shutdown specific topics are identified. A summary of the procedural guidance from Appendix 19.8A is given here in Table B-1. It lists seven topics for which procedural guidance related to shutdown operations is provided. For each topic, Table B-1 lists significant aspects that are addressed and also lists the relevant sections of Appendix 19.8A where there is additional information. These topics are expanded in the following sections of this LMOG.

To assure proper response to mitigate releases of radioactivity to the outside atmosphere, the plant operator will:

2.0 INTERFACE BETWEEN LMOG AND EOGs

The Emergency Operations Guidelines (EOGs) provide guidance for responding to events initiated from Modes 1 through 4. Two levels are provided. Optimal Recovery Guidelines (ORGs) apply when a specific event is identified and for which specific recovery sequences have been formulated. Functional Recovery Guidelines (FRGs) apply when a specific recovery sequence cannot be identified or when it becomes ineffective.

The EOGs may be entered from the critical Modes 1 and 2. They may also be entered from shutdown Modes 3 and 4 when SIAS has not been blocked and LTOP has not been initiated. The EOGs are typically exited when the safety functions are satisfied and after shutdown cooling has been initiated.

This LMOG may be entered from a shutdown mode following an event initiated from shutdown. It would not typically be entered directly from the EOGs because exit from the EOGs requires that LTOP has been initiated and that would imply that shutdown cooling has been established. An exception might be a situation where following an event initiated from a higher mode all success paths have been accomplished, but the event has resulted in some deficiency in components or systems for shutdown cooling that can be mitigated by the guidance in this Appendix.

locations provide sufficient water depth below the pool seal elevation to maintain water coverage over the fuel assembly. These two (2) areas are separated by a section of the refueling cavity that is at the elevation of the reactor pressure vessel flange. The raised section is about eleven feet long.

The refueling machine transit time over this area is less than thirty (30) seconds. The refueling machine can lower the fuel assembly below the reactor pressure vessel flange in approximately three (3) minutes in the slow speed range of the hoist. Therefore the eighty (80) minute drain down time (assuming no water makeup capability) is adequate to ensure the fuel assembly being transferred can be kept underwater in the event the pool seal develops the maximum credible leak.

IP Procedures should ensure the availability of the SCS, CSS and BAMU pumps to respond to a reactor cavity seal failure event.

Table B-1 Summary of Procedural Guidance Related to Shutdown Operations (Cont'd.)

Topic	Procedural Guidance	Appendix 19.8A Section
Unplanned Draining of the Reactor Coolant (Cont'd.)	<u>MITIGATION</u> (Immediate Operator Action) Identify leakage path. Isolate leakage path. Make up losses. a. Safety Injection b. SCS via IRWST c. Containment spray from IRWST via SCS lines d. Charging pumps [1] e. BAST f. Safety Injection Tanks g. BAMU pumps [1]	2.3.3.4, 2.12.3, 2.12.4, 2.12.2.3(2) Appendix B
<u>Heavy Loads</u> 1) Drop of transported equipment. 2) Drop of fuel bundle 3) Refueling pool seal integrity. 4) Loads over ICI table.	Restrictions specified for: a. Lift Height b. Travel Directions c. Systems Lineup (Specified in Chapter 9 and Plant Designer's "Heavy Load Guides")	2.11.3
<u>Outage Maintenance</u>	Strategy for Shutdown Operations a. Define operating and operational divisions. b. Limit maintenance activities to components and systems not included in a).	2.4.3.2.2 Appendix B
Fire Protection	Administratively require fire protection systems to remain operable in shutdown modes. Procedurally Control: a. Combustible materials b. Housekeeping c. Hot work Pre-Fire Plan a. Outline fire fighting strategies b. Monitor status of fire barriers	2.7.3.2, and 2.7.3.3 2.7.3.2

Table B-1 Summary of Procedural Guidance Related to Shutdown Operations (Cont'd.)

Topic	Procedural Guidance	Appendix 19.8A Section
RCS Cooling Using Feed and Bleed (other systems not available)	RCS Pressurized 1. Start SI pump. 2. Reduce pressure through Safety Depressurization System (SDS) venting to IRWST. (Maintain subcooled temperatures in RCS). 3. Secure operating RCPs (if applicable) 4. Cycle SI feed and SDS bleed to reduce RCS pressure and temperature. 5. When depressurized, open SDS and Run SI continuously. 6. Align SDC heat exchanger for IRWST cooling. 7. Restore Normal SDC systems. RCS Depressurized 1. Start SI 2. Open SDS 3. Secure RCP's (if RCS not vented) 4. Align SDC heat exchanger for IRWST Cooling. 5. Restore normal SDC Systems	2.4.3.1.3.1.1 2.4.3.1.3.2.1
SG Tube Rupture	Include in Emergency Procedure Guides a requirement to maintain a positive primary to secondary pressure differential.	Table 2.6-1 Section C(a)
Lockout of main feedwater pumps in shutdown modes with RTCBs closed.	Administratively lockout main feedwater pumps if subcritical.	4.1.1 and 4.1.2

[1] Procedures should ensure that these pumps are not made unavailable at the same time.