APPENDIX E PLANT COMPARATIVE EVALUATION WITH THE PROPOSED AEC 70 DESIGN CRITERIA

TABLE OF CONTENTS

Section Page

E.1 Summary Description 2

E.2 Criterion - Conformance 2

E.2.1 Group I - Overall Plant Requirements 2
E.2.2 Group II - Protection by Multiple Fission Products Barriers 12
E.2.3 Group III - Nuclear and Radiation Controls 16
E.2.4 Group IV - Reliability and Testability of Protection Systems 23
E.2.5 Group V - Reactivity Control 28
E.2.6 Group VI - Reactor Coolant Pressure Boundary 33
E.2.7 Group VII - Engineered Safety Features 36
E.2.8 Group VIII - Fuel and Waste Storage Systems 56
E.2.9 Group IX - Plant Effluents 60

APPENDIX E PLANT COMPARATIVE EVALUATION WITH THE PROPOSED AEC 70 DESIGN CRITERIA

E.1 Summary Description

This appendix contains a comparative evaluation of the design basis of the Monticello Nuclear Generating Plant, Unit 1, with the 70 General Design Criteria for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission for public comment in July, 1967.

The comparative evaluation is made with each of the nine groups of criteria sent out in the July 1967 AEC release. As to each group, there is a statement of Northern States Power Company's current understanding of the intent of the criteria in that group and a discussion of the plant design conformance with the intent of the group of criteria. Following a restatement of the 70 proposed criteria is complete list of references to locations in this USAR where there is discussed subject matter relating to the intent of the particular criteria.

Based on its current understanding of the intent of the 70 proposed-criteria, the applicant believes that the Monticello Nuclear Generating Plant, Unit 1, is in conformance with the intent of such proposed criteria.

E.2 Criterion - Conformance

E.2.1 Group I - Overall Plant Requirements

The intent of the current draft of the proposed criteria for this group is to identify and record the adequacy of the quality control and assurance programs, the applicable codes or standards, the standards of design, fabrication and erection, and to assure protection against appropriate environmental phenomena. Test Procedures, and inspection acceptance levels of the reactor facility's essential components and systems are also identified. The influence of this sharing of common reactor facility components and systems along with the fire and explosion protection for all equipment is also to establish and described.

It is concluded that the design of this plant is in conformance with the criteria of Group I based on NSP's current understanding of the intent of these criteria.

The reactor facility's essential components and systems are designed, fabricated, erected, and perform in accordance with the specified quality standards which are, as a minimum, in accordance with applicable codes and regulations. These components and systems as well as applicable codes and standards have been identified in the report. Specific sections are included in the reference letter list following this group's discussion. Where components or system design exceeds code requirements it has been noted. A quality control and assurance program has been established to assure compliance with acceptable quality control specifications and procedures. These programs as well as applicable tests and inspections have been identified. Specific sections are included in the reference list. In planning and executing these control and assurance programs, particular attention was given to the quality control specifications and to their compliance by those systems, components, and structures which are important to the plant safety. (Criterion 1) The plant equipment which is important to safety is designed to permit safe plant operation and to accommodate all design basis

accidents for all appropriate environmental phenomena at the site without loss of their capability, taking into consideration historical data and suitable margins for uncertainties. (Criterion 2) Further design allowances are provided to minimize the occurrence of fire and explosions and their effects by the use of noncombustible and fire resistant materials through the plant. (Criterion 3) Records of design, fabrication, and construction for this facility are to be stored or maintained either under the applicant's control or available to the applicant for inspection. (Criterion 5) This reactor facility consists of a single BWR generating unit. (Criterion 4)

References to applicable sections of the USAR are given below for the individual criteria of this group.

<u>Criterion 1 - Quality Standards (Category A)</u> Those systems and components of reactor facilities which are essential to prevention of accidents which could affect the public health and safety or to mitigation to their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standard, quality assurance programs, test procedures, and acceptance levels used is required.

<u>Conformance 1 - Quality Standards (Category A)</u>

a. <u>General</u>

Section 1.2.1	▪	Principal Design Criteria - General Criteria
Section 1.3.1.3	▪	Summary Design Description and Safety Analysis - Geology
Section 1.3.1.4	▪	Summary Design Description and Safety Analysis - Hydrology
Section 1.3.1.5	▪	Summary Design Description and Safety Analysis - Regional and Site Meteorology
Section 1.3.1.6	▪	Summary Design Description and Safety Analysis - Seismology and Design Response Spectrum
Section 1.3.1.7	▪	Summary Design Description and Safety Analysis - Site Environmental Monitoring Program
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems

Section 1.3.5 ▪ Summary Design Description and Safety Analysis - Plant Instrumentation Control System

Section 1.3.6 ▪ Summary Design Description and Safety Analysis - Plant Fuel Storage and Handling Systems

Section 1.3.8 ▪ Summary Design Description and Safety Analysis - Plant Electrical Power Systems

Section 1.3.9 ▪ Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures

Section 1.3.10 ▪ Summary Design Description and Safety Analysis - Plant Radioactive Waste Control Systems

Section Appendix C ▪ Quality Assurance Program

b. <u>Containment Barriers</u>

Section 1.2.4 ▪ Principal Design Criteria - Plant Containment

Section 1.3.3 ▪ Summary Design Description and Safety Analysis - Plant Containment System

Section 1.3.11 ▪ Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety

<u>Fuel</u>

Section 1.3.2 ▪ Summary Design Description and Safety Analysis - Reactor System

Section 3.4.4 ▪ Fuel Mechanical Characteristics - Surveillance and Testing

<u>Fuel Cladding</u>

Section 3.2.3 ▪ Thermal and Hydraulic Characteristics - Design Criteria and Safety Limits

Section 3.4.1 ▪ Fuel Mechanical Characteristics - Design Basis

Section 3.4.2 ▪ Fuel Mechanical Characteristics - Description of Fuel Assemblies

Section 3.4.3	▪	Fuel Mechanical Characteristics - Design Evaluation
Section 3.4.4	▪	Fuel Mechanical Characteristics - Surveillance and Testing

Reactor Coolant System

Section 4	▪	Reactor Coolant System

Primary Containment System

Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.2	▪	Primary Containment System - Description
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.2.4	▪	Primary Containment System - Inspection and Testing

Secondary Containment System

Section 5.3.2	▪	Secondary Containment System - Design Basis
Section 5.3.5	▪	Secondary Containment System - Performance Analysis

Standby Gas Treatment System

Section 5.3.4.1	▪	Secondary Containment System - Standby Gas Treatment System (SGTS)
Section 10.3.2	▪	Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems

Plant Elevated Release Point

Section 9.3	▪	Gaseous Radwaste System

c. Plant Engineered Safeguards

Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 6.1	▪	Plant Engineered Safeguards - Summary Description

Control Rod Velocity Limiters

Section 6.4.3	▪	Control Rod Velocity Limiters - Performance Analysis
Section 6.4.4	▪	Control Rod Velocity Limiters - Inspection and Testing

Control Rod Drive Housing Supports

Section 6.5.3	▪	Control Rod Drive Housing Supports - Performance Analysis
Section 6.5.4	▪	Control Rod Drive Housing Supports - Inspection and Testing

Reactor Standby Liquid Flow Control System

Section 6.6.3	▪	Standby Liquid Control System - Performance Analysis
Section 6.6.4	▪	Standby Liquid Control System - Inspection and Training

Main Steam Line Flow Restrictors

Section 6.3.3	▪	Main Steam Line Flow Restrictions - Performance Analysis
Section 6.3.4	▪	Main Steam Line Flow Restrictions - Inspection and Testing

Emergency Core Cooling Systems (ECCS)

Section 6.2.4.3	▪	High Pressure Coolant Injection System (HPCI) - Performance Analysis
Section 6.2.5.3	▪	Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.2.3	▪	Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.3.3	▪	Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.6	▪	Emergency Core Cooling System (ECCS) - ECCS Performance Evaluation

Plant Structures and Shielding

Section 12.2	▪	Plant Principal Structures and Foundations
Section 12.3	▪	Shielding and Radiation Protection

Criterion 2 - Performance Standards (Category A) Those systems and components of reactor facilities which are essential to prevention of accidents which could affect the public health and safety or to mitigation to their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Conformance 2 - Performance Standards (Category A)

a. General

Section 1.2.1	▪	Principal Design Criteria - General Criteria
Section 1.3.1.3	▪	Summary Design Description and Safety Analysis - Geology
Section 1.3.1.4	▪	Summary Design Description and Safety Analysis - Hydrology
Section 1.3.1.5	▪	Summary Design Description and Safety Analysis - Site and Regional Meteorology
Section 1.3.1.6	▪	Summary Design Description and Safety Analysis - Seismology and Design Response Spectra
Section 1.3.1.7	▪	Summary Design Description and Safety Analysis - Site Environmental Monitoring Program
Section 1.3.8	▪	Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 1.3.9	▪	Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures

Section 1.3.10	▪ Summary Design Description and Safety Analysis - Plant Radioactive Waste Control Systems
Section 1.3.11	▪ Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 2.3	▪ Meteorology
Section 2.4	▪ Hydrology
Section 2.5	▪ Geology and Soil Investigation
Section 2.6	▪ Seismology
Section 2.7	▪ Radiation Environmental Monitoring Program (REMP)
Section 2.8	▪ Ecological and Biological Studies

b. Containment Barriers

Section 1.3.3	▪ Summary Design Description and Safety Analysis - Plant Containment System

Fuel Cladding

Section 1.3.6	▪ Summary Design Description and Safety Analysis - Plant Fuel Storage and Handling Systems
Section 3.2.1	▪ Thermal and Hydraulic Characteristics - Design Basis
Section 3.2.3	▪ Thermal and Hydraulic Characteristics -Design Criteria and Safety Limits
Section 3.3.1	▪ Nuclear Characteristics - Design Basis
Section 3.3.3	▪ Nuclear Characteristics - Nuclear Design Characteristics
Section 3.4.1	▪ Fuel Mechanical Characteristics - Design Basis
Section 3.4.3	▪ Fuel Mechanical Characteristics - Design Evaluation
Section 3.5.1	▪ Reactivity Control Mechanical Characteristics - Design Basis

Section 3.5.5	▪	Reactivity Control Mechanical Characteristics - Operation and Performance Analysis

Reactor Coolant System

Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 4 - Complete	▪	Reactor Coolant System

Primary Containment System

Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 12.2.1.1	▪	Plant Principal Structures and Foundations - Safety Categories
Section Appendix A	▪	Design Bases - Seismic Design and Analysis
Section 12.2.1.6	▪	Plant Principal Structures and Foundations - Wind Loads

Secondary Containment System

Section 5.3.2	▪	Secondary Containment System - Design Basis
Section 5.3.5	▪	Secondary Containment System - Performance Analysis
Section 12.2.1.1	▪	Plant Principal Structures and Foundations - Safety Categories
Section 12.2.1.6	▪	Plant Principal Structures and Foundations - Wind Loads
Section 12.2.1.7	▪	Plant Principal Structures and Foundations - Flooding

Standby Gas Treatment System

Section 5.3.4.1	▪	Secondary Containment System - Standby Gas Treatment System (SGTS)
Section 12.2.1.2	▪	Plant Principal Structures and Foundations - Class I Structures and Equipment

Plant Elevated Release Point

Section 9.3 ▪ Gaseous Radwaste System

c. Plant Engineered Safeguards

Section 1.2.3 ▪ Principal Design Criteria - Reactor Core Cooling

Section 1.3.4 ▪ Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems

Section 1.3.5 ▪ Summary Design Description and Safety Analysis - Plant Instrumentation Control System

Control Rod Velocity Limiters

Section 6.4.1 ▪ Control Rod Velocity Limiters - Design Basis

Section 6.4.3 ▪ Control Rod Velocity Limiters - Performance Analysis

Control Rod Drive Housing Supports

Section 6.5.1 ▪ Control Rod Drive Housing Supports - Design Basis

Section 6.5.3 ▪ Control Rod Drive Housing Supports - Performance Analysis

Reactor Standby Liquid Flow Control System

Section 6.6.1 ▪ Standby Liquid Control System - Design Basis

Section 6.6.3 ▪ Standby Liquid Control System - Performance Analysis

Main Steam Line Flow Restrictors

Section 6.3.1 ▪ Main Steam Line Flow Restrictions - Design Basis

Section 6.3.3 ▪ Main Steam Line Flow Restrictions - Performance Analysis

Emergency Core Cooling Systems (ECCS)

Section 6.2.1.1	▪ Emergency Core Cooling Systems (ECCS) - ECCS Design Basis
Section 6.2.4.3	▪ High Pressure Coolant Injection System (HPCI) - Performance Analysis
Section 6.2.5.3	▪ Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.2.3	▪ Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.3.3	▪ Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.6	▪ Emergency Core Cooling Systems (ECCS) - ECCS Performance Evaluation

Plant Structures and Shielding

Section 12.2	▪ Plant Principal Structures and Foundations
Section 12.3	▪ Shielding and Radiation Protection

Criterion 3 - Fire Protection (Category A) The reactor facility shall be designed (a) to minimize the probability of events such as fires and explosions and (b) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical through the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Conformance 3 - Fire Protection (Category A)

Section 1.2.1	▪ Principal Design Criteria - General Criteria
Section 10.3.1	▪ Plant Service Systems - Fire Protection Systems

Criterion 4 - Sharing of Systems (Category A) Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Conformance 4 - Sharing of Systems (Category A) This Plant is a single unit and does not share any system, component, or equipment with any other facility.

Criterion 5 - Records Requirements (Category A) Records of design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator (NSP) or under its control throughout the life of the reactor.

Conformance 5 - Records Requirements (Category A)

Section Appendix C	▪	Quality Assurance Program
Section 13.4	▪	Operational Procedures
Section 13.5	▪	Operational Records and Reporting Requirements

E.2.2 Group II - Protection by Multiple Fission Products Barriers

The intent of the current draft of the proposed criteria for this group is to assure that the plant has been provided with multiple barriers to protect against or to mitigate the effects of fission products prior to being released to the site environs and to establish that these barriers will remain intact under all operational transients caused by a single reactor operator error or equipment malfunction. It is the further intent of this group that proper barriers are made available for the design basis accidents.

It is concluded that design of this plant is in conformance with the Criteria of Group II Based on NSP's understanding of the intent of these criteria.

The plant containment barriers are the basic features which minimize release of radioactive materials and associated doses. A boiling water reactor provides seven means of containing and/or mitigating the release of fission products; (a) the high density ceramic UO_2 fuel, (b) the high integrity Zircaloy cladding, (c) the reactor vessel and its connected piping and isolation valves, (d) the drywell-suppression chamber primary containment, (e) the reactor building (secondary containment), (f) the reactor building standby gas treatment system utilizing high efficiency absolute and charcoal filters, and (g) the plant main stack. The primary containment system is designed, fabricated, and erected to accommodate without failure, the pressures and temperatures resulting from or subsequent to double-ended rupture or equivalent failure of any coolant pipe within the primary containment. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open for refueling operations. The two containment systems and such other associated engineered safety systems as may be necessary are designed and maintained so that off-site doses resulting from postulated design basis accidents are below the values stated in 10CFR100. (Criterion 10) The reactor core is designed so there is no inherent tendency for sudden divergent oscillation of operating characteristics of divergent power transient in any mode of plant operation. (Criterion 6, 7) The basis of the reactor core design, in combination with the plant equipment characteristics, nuclear instrumentation system, and the reactor protection system is, to provide margins to ensure that fuel damage will not occur in normal operation or operational transient caused by single reactor operator error or equipment malfunction. (Criterion 6, 7) The reactor core is designed so that the overall power coefficient in the power operating range is not positive. (Criterion 8) The reactor coolant system is

designed to carry its dead weight and specified live loads, separately or concurrently, such as pressure and temperature stress, vibrations, seismic loads as appropriately prescribed for the plant. Provisions are made to control or shutdown the reactor coolant system in the event of a malfunction of the operating equipment or excessive leakage of the coolant from the system. The reactor vessel and support structure are designed, within the limits of applicable criteria for low probability accident conditions, to withstand the forces that would be created by a full area flow from any vessel nozzle to the containment atmosphere with the reactor vessel at design pressure concurrent with the plant design earthquake loads. (Criterion 9)

References to applicable sections of the USAR are given below for the individual criteria of this group.

Criterion 6 - Reactor Core Design (Category A) The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of off-site power.

Conformance 6 - Reactor Core Design (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 3.2	▪	Thermal and Hydraulic Characteristics
Section 3.3	▪	Nuclear Characteristics
Section 3.4	▪	Fuel Mechanical Characteristics
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 4	▪	Reactor Coolant System
Section 8.4	▪	Plant Standby Diesel Generator Systems
Section 8.5	▪	D-C Power Supply Systems
Section 8.6	▪	Reactor Protection System Power Supplies

Section 10.2.5 ▪ Reactor Auxiliary Systems - Reactor Core Isolation Cooling System (RCIC)

Section 14.4.3 ▪ Transient Events Analyzed for Core Reload - Rod Withdrawal Error

Criterion 7 - Suppression of Power Oscillations (Category B) The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Conformance 7 - Suppression of Power Oscillations (Category B)

Section 1.2.2 ▪ Principal Design Criteria - Reactor Core

Criterion 8 - Overall Power Coefficient (Category B) The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Conformance 8 - Overall Power Coefficient (Category B)

Section 1.2.2 ▪ Principal Design Criteria - Reactor Core

Section 3.2 ▪ Thermal and Hydraulic Characteristics

Section 3.5 ▪ Reactivity Control Mechanical Characteristics

Criterion 9 - Reactor Coolant Pressure Boundary (Category A) The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Conformance 9 - Reactor Coolant Pressure Boundary (Category A)

Section 1.2.2 ▪ Principal Design Criteria - Reactor Core

Section 4 Complete ▪ Reactor Coolant System

Section 7.4 ▪ Reactor Vessel Instrumentation

Criterion 10 - Containment (Category A) Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary area, without loss of required integrity and, together with other engineered safety features as may be necessary to retain for as long as the situation requires the functional capability to protect the public.

Conformance 10 - Containment (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 1.3.3	▪	Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 4 Complete	▪	Reactor Coolant System
Section 5.1	▪	Containment System - Summary Description
Section 6.2	▪	Emergency Core Cooling Systems (ECCS)
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.5	▪	Control Rod Drive Housing Supports
Section 6.6	▪	Standby Liquid Control System
Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.3.2	▪	Secondary Containment System - Design Basis
Section 12 Complete	▪	Plant Structures and Shielding
Section 14.1.1	▪	Summary Description - General Safety Design Basis
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

E.2.3 Group III - Nuclear and Radiation Controls

The intent of the current draft of the proposed criteria for this group is to identify and define the instrumentation and control systems, necessary for maintaining the plant in a safe operational status. This, also includes determining the adequacy of radiation shielding, effluent monitoring, and fission process controls, and providing for the effective sensing of abnormal conditions and initiation of engineered safety features.

It is concluded that the design of this plant is in conformance with the criteria of Group III based on NSP's current understanding of the intent of these criteria.

The plant is provided with a centralized main control room having adequate shielding, fire protection, air conditioning and facilities to permit access and continuous occupancy under 10CFR20 dose limits during all design basis accident situations. However, if it is necessary to evacuate the main control room the design does not preclude the capability to bring the plant to a safe-cold shutdown from outside the main control room. (Criterion 11) The necessary plant controls, instrumentation, and alarms for safe and orderly operation are located in the main control room. These include such controls and instrumentation as the reactor coolant system leakage detection system. (Criterion 11, 13, 16) The performance of the reactor core and the indication of power level are continuously monitored by the in-core nuclear instrumentation system. (Criterion 13) The reactor protection system, independent from the plant process control systems, overrides all other controls to initiate any required safety action. The reactor protection system automatically initiates appropriate action whenever the plant conditions approach pre-established operational limits. The system acts specifically to initiate the emergency core and containment cooling systems as required. (Criterion 12, 13, 14, 15) The plant radiation and process monitoring systems are provided for monitoring significant parameters from specific plant process systems and specific areas including the plant effluents to the site environs and to provide alarms and signals for appropriate corrective actions. (Criterion 17, 18)

Reference to applicable sections of the USAR are given below for the individual criteria of this group.

<u>Criterion 11 - Control Room (Category B)</u> The facility shall be provided with a control room from which action to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control to the facility without radiation exposures of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access
the control room is lost due to fire or other causes.

Conformance 11 - Control Room (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.8	▪	Principal Design Criteria - Plant Shielding and Access Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrumentation and Control Systems
Section 1.3.9	▪	Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 7.2	▪	Reactor Control Systems
Section 7.3	▪	Nuclear Instrumentation System
Section 7.6	▪	Plant Protection System
Section 7.7	▪	Turbine-Generator System Instrumentation and Control
Section 12.3.3	▪	Shielding and Radiation Protection - Performance Analysis

Criterion 12 - Instrumentation and Control Systems (Category B) Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Conformance 12 - Instrumentation and Control Systems (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrumentation Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 7	▪	Plant Instrumentation and Control Systems
Section 7.2	▪	Reactor Control Systems

Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.5	▪	Plant Radiation Monitoring Systems
Section 7.6	▪	Plant Protection System
Section 7.7	▪	Turbine-Generator System Instrumentation and Control
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer

<u>Criterion 13 - Fission Process Monitors and Controls (Category B)</u> Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variation in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

<u>Conformance 13 - Fission Process Monitors and Controls (Category B)</u>

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrumentation Control Systems
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.6	▪	Standby Liquid Control System
Section 7.2	▪	Reactor Control Systems
Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.6	▪	Plant Protection System
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer

Criterion 14 - Core Protection Systems (Category B) Core protection systems together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Conformance 14 -Core Protection Systems (Category B)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrumentation and Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 3.3	▪	Nuclear Characteristics
Section 3.4	▪	Fuel Mechanical Characteristics
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.2	▪	Emergency Core Cooling System (ECCS)
Section 6.3	▪	Main Steam Line Flow Restrictions
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.5	▪	Control Rod Drive Housing Supports
Section 7.2	▪	Reactor Control Systems
Section 7.3	▪	Nuclear Instrumentation System
Section 7.6	▪	Plant Protection System
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer
Section 8 Complete	▪	Plant Electrical Systems
Section 14 Complete	▪	Plant Safety Analysis

Criterion 15 - Engineered Safety Features Protection Systems (Category B) Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Conformance 15 - Engineered Safety Features Protection Systems (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 6 Complete	▪	Plant Engineered Safeguards
Section 7.2	▪	Reactor Control Systems
Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.5	▪	Plant Radiation Monitoring Systems
Section 7.6	▪	Plant Protection System
Section 7.7	▪	Turbine-Generator Systems Instrumentation and Control
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer

Criterion 16 - Monitoring Reactor Coolant Pressure Boundary (Category B) Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Conformance 16 - Monitoring Reactor Coolant Pressure Boundary (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 5.2	▪	Primary Containment System
Section 7.1	▪	Plant Instrumentation and Control Systems - Summary Description

Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.6	▪	Plant Protection System

Criterion 17 - Monitoring Radioactivity Releases (Category B) Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs, for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Conformance 17 - Monitoring Radioactivity Releases (Category B)

Section 1.2.7	▪	Principal Design Criteria - Plant Radioactive Waste Disposal
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 5.3.4.1	▪	Secondary Containment System - Standby Gas Treatment System (SGTS)
Section 7.5	▪	Plant Radiation Monitoring Systems
Section 7.6.1	▪	Plant Protection System - Reactor Protection System
Section 9.2	▪	Liquid Radwaste System
Section 9.3	▪	Gaseous Radwaste System
Section 10.3.2	▪	Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems
Section 10.3.7	▪	Plant Service Systems - Plant Process Sampling System
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

Criterion 18 - Monitoring Fuel and Waste Storage (Category B) Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Conformance 18 - Monitoring Fuel and Waste Storage (Category B)

Section 7.5	▪	Plant Radiation Monitoring Systems
Section 7.6.1	▪	Plant Protection System - Reactor Protection System
Section 9.2.1	▪	Liquid Radwaste System - Design Basis
Section 9.2.2.1	▪	Liquid Radwaste System - General
Section 9.2.2.3	▪	Liquid Radwaste System - Instrumentation and Control of the Liquid Radwaste
Section 9.3.1	▪	Gaseous Radwaste System - Design Basis
Section 9.3.3	▪	Gaseous Radwaste System - Performance Analysis
Section 9.4.1	▪	Solid Radwaste System - Design Basis
Section 9.4.3	▪	Solid Radwaste System - Performance Analysis
Section 10.2.1.1	▪	Reactor Auxiliary Systems - Design Basis
Section 10.2.1.2	▪	Reactor Auxiliary Systems - Description
Section 10.2.2.1	▪	Reactor Auxiliary Systems - Design Basis
Section 10.2.2.3	▪	Reactor Auxiliary Systems - Performance Analysis

E.2.4 Group IV - Reliability and Testability of Protection Systems

The intent of the current draft of the proposed criteria for this group is to identify and establish the functional reliability, in-service testability, redundancy, physical and electrical independence and separation, and fail-safe design of the reactor protection instrumentation and control systems.

It is concluded that the design of this plant is in conformance with the criteria of Group IV based on NSP's current understanding of the intent of these criteria.

The reactor protection system automatically overrides the plant normal operational control system (that is, functions independently) to initiate appropriate action whenever the plant conditions monitored (neutron flux, containment, and vessel pressure, etc.) by the system approach pre-established limits. (Criterion 22) By means of a dual channel protection system with complete redundancy in each channel, no loss of the protection systems can occur by either component failure or removal from service. The reactor protection system acts to shutdown the reactor, close primary containment isolation valves and initiates the operation of the emergency core and containment cooling systems. The reactor protection system is designed so that a credible plant transient or accident is sensed by different parametric measurements (e.g., loss of coolant accident is detected by high drywell pressure and low-low reactor level monitors). (Criterion 20) Components of the redundant subsystems can be removed from service for testing and maintenance without negating the ability of the protection system to perform its protection functions (even when subjected to a single event, multiple failure incident) upon receipt of the appropriate signals. (Criterion 19, 20, 21) The design of the reactor protection system is such as to facilitate maintenance and trouble shooting while the reactor is at power operation without impeding the plant's operation or impairing its safety function. System faults are annunciated in the main control room. (Criterion 25) The system electrical power requirements are supplied from independent, redundant sources. (Criterion 24) The system circuits are isolated to preclude a circuit fault from inducing a fault in another circuit and to reduce the likelihood that adverse conditions, which might affect system reliability (1 of 2 x 2), will encompass more than one circuit. The system sensors are electrically and physically separated with both sensors in any one trip channel not allowed to occupy the same local area or to be connected to the same power source or process measurement line. The system internal wiring or external cable routing arrangement are such as to negate any external influence (a fire or accident) on the systems performance. (Criterion 23, 24) A failure of any one reactor protection system input or subsystem component will produce a trip in one of two channels, a situation insufficient to produce a reactor scram but readily available to perform its protective function upon another trip (either by failure or by exceeding the preset trip). (Criterion 26) This reactor protection system design includes allowance for single reactor operator error and equipment malfunction and still performs its intended function. (Criterion 21) References to applicable sections of the USAR are given below for the individual criteria of this group.

Criterion 19 - Protection Systems Reliability (Category B) Protection systems shall be designed for high functional reliability and in-service testability commensurate, with the safety functions to be performed.

Conformance 19 - Protection Systems Reliability (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.1	▪	Summary Design Description and Safety Analysis - Plant Site and Environs
Section 7.2	▪	Reactor Control Systems
Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.5.2	▪	Plant Radiation Monitoring systems - Process Radiation Monitoring Systems
Section 7.6	▪	Plant Protection System
Section 11.2	▪	Turbine-Generator System
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

Criterion 20 - Protection Systems Redundancy and Independence (Category B) Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true redundant instrumentation components.

Conformance 20 - Protection Systems Redundancy and Independence (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 7.1	▪	Plant Instrumentation and Control Systems - Summary Description
Section 7.3	▪	Nuclear Instrumentation System

Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.5.2	▪	Plant Radiation Monitoring Systems - Process Radiation Monitoring System
Section 7.6	▪	Plant Protection System
Section 11.2	▪	Turbine-Generator System
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

<u>Criterion 21 - Single Failure Definition (Category B)</u> Multiple failures from a single event shall be treated as a single failure.

<u>Conformance 21 - Single Failure Definition (Category B)</u>

Section 7.2	▪	Reactor Control Systems
Section 7.6	▪	Plant Protection System
Section 14.4	▪	Transient Events Analyzed for Core Reload

<u>Criterion 22 - Separation of Protection and Control Instrumentation Systems (Category B)</u> Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying requirements for protection channels.

<u>Conformance 22 - Separation of Protection and Control Instrumentation Systems (Category B)</u>

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 7.4.2	▪	Reactor Vessel Instrumentation - Description
Section 7.4.3	▪	Reactor Vessel Instrumentation - Inspection and Testing
Section 7.6.3	▪	Plant Protection System - Primary Containment Isolation System

Criterion 23 - Protection Against Multiple Disability for Protection Systems (Category B)
The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Conformance 23 - Protection Against Multiple Disability for Protection Systems (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 5.2.1.3	▪	Primary Containment System -Containment Penetrations
Section 7.1	▪	Plant Instrumentation and Control Systems - Summary Description
Section 7.3	▪	Nuclear Instrumentation System
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.5	▪	Plant Radiation Monitoring Systems
Section 7.6	▪	Plant Protection System
Section 11.2	▪	Turbine-Generator System

Criterion 24 - Emergency Power for Protection Systems (Category B) In the event of the loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Conformance 24 - Emergency Power for Protection Systems (Category B)

Section 1.2.6	▪	Principal Design Criteria - Plant Electrical Power
Section 1.3.8	▪	Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 7 Complete	▪	Plant Instrumentation and Control Systems
Section 8.3	▪	Auxiliary Power System
Section 8.4	▪	Plant Standby Diesel Generator Systems
Section 8.5	▪	D-C Power Supply Systems

Section 8.6	▪	Reactor Protection System Power Supplies
Section 10.3.8	▪	Plant Service Systems - Plant Communication System
Section 10.3.9	▪	Plant Service Systems - Plant Lighting System

<u>Criterion 25 - Demonstration of Functional Operability of Protection System (Category B)</u> Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

<u>Conformance 25 - Demonstration of Functional Operability of Protection System (Category B)</u>

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 7.3.5.5	▪	Nuclear Instrumentation System - Inspection and Testing
Section 7.4.3	▪	Reactor Vessel Instrumentation - Inspection and Testing
Section 7.5.2.1	▪	Plant Radiation Monitoring Subsystem - General
Section 7.5.2.4.2	▪	Plant Radiation Monitoring Systems - Description
Section 7.6.1.4	▪	Plant Protection System - Inspection and Testing
Section 7.6.3.4	▪	Plant Protection System - Inspection and Testing
Section 10.3.1.4	▪	Plant Service Systems - Inspection and Testing
Section 10.3.2.4	▪	Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems
Section 10.3.9	▪	Plant Service Systems - Plant Lighting System
Section 10.4	▪	Plant Cooling Systems

Criterion 26 - Protection Systems Fail-Safe Design (Category B) The protection systems shall be designed to fail into safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

Conformance 26 - Protection Systems Fail-Safe Design (Category B)

Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.6	▪	Principal Design Criteria - Plant Electrical Power
Section 1.3.5	▪	Summary Design Description and Safety Analysis - Plant Instrument Control Systems
Section 1.3.8	▪	Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 3.5.1	▪	Reactivity Control Mechanical Characteristics - Design Basis
Section 3.5.5	▪	Reactivity Control Mechanical Characteristics - Operation and Performance Analysis
Section 7.6	▪	Plant Protection Systems
Section 8.6	▪	Reactor Protection System Power Supplies
Section 10.3	▪	Plant Service Systems
Section 10.4	▪	Plant Cooling System

E.2.5 Group V - Reactivity Control

The intent of the current draft of the proposed criteria for this group is to establish the reactor core reactivity insertion and withdrawal rate limitations and the means to control the plant operations within these limits.

It is concluded that the design of this plant is in conformance with the criteria of Group V based on NSP's current understanding of the intent of these criteria.

The plant design contains two independent reactivity control systems of different principles. Control of reactivity is operationally provided by a combination of movable control rods, fixed control devices or curtains, and reactor coolant recirculation system flow. These subsystems accommodate fuel burnup, load changes, and long term reactivity changes. Reactor shutdown by the control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for all operating transients. A reactor standby liquid control system is provided as a redundant, independent shutdown system

to cover emergencies in the operational reactivity control system described above. This system is designed to shut down the reactor in about two hours. (Criterion 27, 28)

The reactor core is designed to have (a) a reactivity response which regulates or damps changes in power level and spatial distributions of power productions to a level consistent with safe and efficient operation, (b) a negative reactivity feedback consistent with the requirements of overall plant nuclear-hydrodynamic stability, and (c) have a strong negative reactivity feedback under severe power transient conditions. (Criterion 27, 31) The operational reactivity control system is designed such that under conditions of normal operation sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition, and means are provided for continuous regulation of the reactor core excess reactivity and reactivity distribution. (Criterion 29, 30) This system is also designed to be capable of compensating for positive and negative reactivity changes resulting from nuclear coefficients, fuel depletion, and fission product transients and buildup. (Criterion 29) The system design is such that control rod worths, and the rate at which reactivity can be added, are limited to assure that credible reactivity accidents cannot cause a transient capable of damaging the reactor coolant system, disrupt the reactor core, its support structures, or other vessel internals sufficiently to impair the emergency core cooling systems effectiveness, if needed. Acceptable fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or reactor operator error. (Criterion 29, 31, 32)

References to applicable sections of the USAR are given below for individual criteria of this group.

Criterion 27 - Redundancy of Reactivity Control (Category A) At least two independent reactivity control systems, preferable of different principles, shall be provided.

Conformance 27 - Redundancy of Reactivity Control (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 3.3.1	▪	Nuclear Characteristic - Design Basis
Section 3.3.3.3	▪	Nuclear Characteristic - Reactivity Control
Section 3.3.3.4	▪	Nuclear Characteristic - Control Rod Worth
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.6.3	▪	Standby Liquid Control System - Performance Analysis
Section 7.2	▪	Reactor Control Systems
Section 8.4	▪	Plant Standby Diesel Generator Systems

Criterion 28 - Reactivity Hot Shutdown Capability (Category A) At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Conformance 28 - Reactivity Hot Shutdown Capability (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 3.3.1	▪	Nuclear Characteristic - Design Basis
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.6	▪	Standby Liquid Control System
Section 7.2	▪	Reactor Control Systems

Criterion 29 - Reactivity Shutdown Capability (Category A) At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceedingly acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most efficient control rod when fully withdrawn shall be provided.

Conformance 29 - Reactivity Shutdown Capability (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.6	▪	Standby Liquid Control System
Section 7.2	▪	Reactor Control Systems

Criterion 30 - Reactivity Holddown Capability (Category B) At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Conformance 30 - Reactivity Holddown Capability (Category B)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 3.3.3.3	▪	Nuclear Characteristic - Reactivity Control
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.6	▪	Standby Liquid Control System
Section 7.2	▪	Reactor Control Systems

Criterion 31 - Reactivity Control Systems Malfunction (Category B) The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Conformance 31 - Reactivity Control Systems Malfunction (Category B)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 3.2	▪	Thermal and Hydraulic Characteristics
Section 3.3	▪	Nuclear Characteristic
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.6	▪	Standby Liquid Control System
Section 7.2	▪	Reactor Control Systems

Criterion 32 - Maximum Reactivity Worth of Control Rods (Category A) Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Conformance 32 - Maximum Reactivity Worth of Control Rods (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 3.3.3.3	▪	Nuclear Characteristic - Reactivity Control
Section 3.3.3.4	▪	Nuclear Characteristic - Control Rod Worth
Section 3.4	▪	Fuel Mechanical Characteristics
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 4 Complete	▪	Reactor Coolant System
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.5	▪	Control Rod Drive Housing Supports
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

E.2.6 Group VI - Reactor Coolant Pressure Boundary

The intent of the current draft of the proposed criteria for this group is to establish the reactor coolant pressure boundary design requirements and to identify the means used to satisfy these design requirements.

It is concluded that the design of this plant is in conformance with the criteria of Group VI based on NSP's current understanding of the intent of these criteria.

The inherent safety features of the reactor core design in combination with certain engineered safety features (control rod velocity limiters and control rod housing supports, etc.) and the plant operational reactivity control system are such that the consequences of the most severe potential nuclear excursion accident, caused by a single component failure within the reactivity control system (control rod drop accident) cannot result in damage (either by motion or rupture) to the reactor coolant system. (Criterion 33) The ASME and USASI Codes are used as the established and acceptable criteria for design, fabrication, and operation of components of the reactor primary pressure system. The reactor primary system is designed and fabricated to meet the following as a minimum: (Criterion 34)

(1) Reactor Vessel - ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection A

(2) Pumps - ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection C

(3) Piping and Valves - USASI-B-31.1, Code for Pressure, Power Piping

Protection against the brittle fracture or other failure modes of the reactor coolant pressure boundary system components is provided for all potential service loading temperatures. Control is exercised in the selection of materials and fabrication and design of equipment and components. It is intended that NDT testing be performed on all ferritic materials in the reactor coolant pressure boundary with appropriate modifications for material thickness of individual components. (Criterion 35)

The reactor coolant system will be given a final hydrostatic test at 1560 psig in accordance with Code requirements prior to initial reactor startup. A hydrostatic test, not to exceed system operating pressure, will be made on the reactor coolant system following each removal and replacement of the reactor vessel head. The reactor primary system will be checked for leaks and abnormal conditions will be corrected before reactor startup. The minimum vessel temperature during hydrostatic test shall at least be 60° F above the calculated NDT temperature prior to pressurizing the vessel. Extensive quality control assurance programs are being so followed during the entire fabrication of the reactor coolant system. (Criterion 36) Vessel material surveillance samples are located within the reactor primary vessel to enable periodic monitoring of material properties with exposure. The program will include specimens of the base metal, heat affected zone metal, and standards specimens. Leakage from the reactor coolant system is monitored during reactor operation. (Criterion 36)

References to applicable sections of the USAR are given on the following page for the individual criteria of this group.

Criterion 33 - Reactor Coolant Pressure Boundary Capability (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Conformance 33 - Reactor Coolant Pressure Boundary Capability (Category A)

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 3.3.3.3	▪	Nuclear Characteristic - Reactivity Control
Section 3.3.3.4	▪	Nuclear Characteristic - Control Rod Worth
Section 3.4	▪	Fuel Mechanical Characteristics
Section 3.5	▪	Reactivity Control Mechanical Characteristics
Section 4 Complete	▪	Reactor Coolant System
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.5	▪	Control Rod Drive Housing Supports
Section 7.8	▪	NUMAC Rod Worth Minimizer and Plant Process Computer
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevent (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties if materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loading, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Conformance 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A)

Section Appendix C	▪	Quality Assurance Program
Section 4 Complete	▪	Reactor Coolant System

Criteria 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of Ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F above the nil ductility transition (NDT) temperature of the component material if the resulting energy is expected to be absorbed within the elastic strain energy range.

Conformance 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)

Section 4.2.3	▪	Reactor Vessel - Design Evaluation
Section 4.3.1	▪	Recirculation System - Design Criteria
Section 4.3.3	▪	Recirculation System - Performance Evaluation
Section 4.4.3	▪	Reactor Pressure Relief System - Performance Analysis

Criteria 36 - Reactor Coolant Pressure Boundary Surveillance (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak tight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Conformance 36 - Reactor Coolant Pressure Boundary Surveillance (Category A)

Section 4.2.1	▪	Reactor Vessel - Design Basis
Section 4.3.1	▪	Recirculation System - Design Basis
Section 4.3.4	▪	Recirculation System - Inspection and Testing
Section 4.4.4	▪	Reactor Pressure Relief System - Inspection and Testing

E.2.7 Group VII - Engineered Safety Features

The intent of the current draft of the proposed criteria for this group is (a) to identify the engineered safety features (ESF), (b) to examine each ESF for independency, redundancy, capability, testability, inspectability, and reliability, (c) to determine the suitability of each ESF for its intended duty, and (d) justify that each ESFs capability-scope envelopes all the anticipated and credible phenomena associated with the plant operational transients or design basis accidents being considered.

It is concluded that the design of the plant is in conformance with the criteria of Group VII based on NSP's current understanding of the intent of these criteria.

The normal plant control systems maintain plant variables within narrow operating limits. These systems are thoroughly engineered and backed up a significant amount of experience in system design and operation. Even if an improbable maloperation or equipment failure including a reactor coolant boundary break up to and including the circumferential rupture of any pipe in that boundary assuming an unobstructed discharge from both sides allows variables to exceed their operating limits, an extensive system of engineered safety features (ESF) limit the transient and the effects to levels well below those which are of public safety concern.

These engineered safety features (ESF) include the normal protection systems (reactor core, reactor coolant system, plant containment system, plant and reactor control systems, reactor protection system, other instrumentation and process systems, etc.); those which offer additional protection against a reactivity excursion (reactor standby liquid control system, control rod velocity limiters, and control rod housing support, etc.); those which act to reduce the consequences of design basis accidents (main steam line flow restrictors, etc.); and those which provide emergency core and standby containment cooling in the event of a loss of normal cooling (emergency core cooling systems (ECCS), residual heat removal system (RHRS), high pressure coolant injection system (HPCIS), automatic depressurization system (ADS), and the standby coolant supply system). (Criterion 37)

The engineered safety features are designed to provide high reliability and ready testability. Specific provisions are made in each ESF to demonstrate operability and performance capabilities. (Criterion 38) Components of the ESF which are required to function after design basis accidents or incidents are designed to withstand the most severe forces and credible environmental effects, including missiles from plant equipment failures anticipated from the events, without impairment of their performance capability. (Criterion 40, 42, 43)

Sufficient off-site and redundant, independent and testable standby auxiliary sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The capacity of the power sources are adequate to accomplish all required engineered safety features functions under all postulated design basis accident conditions (Criterion 39).

The emergency core cooling systems (ECCS) are designed such that at least two different ECCSs of different phenomena are provided to prevent clad melt over the entire spectrum of postulated breaks. Such capability is available even with the loss of all off-site AC power. The ECCS (individual systems) themselves are designed to various levels of component redundancy such that no single active component failure in

addition to the accident will negate the necessary emergency core cooling capability (Criterion 41, 44). To further assure that the ECCS will function properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system (Criterion 46, 47, 48). Design provisions have also been made to enable physical and visual inspection of the ECCS components (Criterion 45).

The primary containment structure, including access openings and penetrations, is designed to withstand the peak transient pressure and temperatures which could occur due to the postulated design basis loss-of-coolant design accident. The containment design includes considerable allowance for energy addition from metal-water or other chemical reactions beyond conditions that would occur with normal operation of Emergency Core Cooling Systems (ECCS). The primary containment has a metal-water reaction capability approximately 55% (at 2 hr) which is 500 times the calculated metal water reaction for the design basis loss-of-coolant accident (Criterion 49). Plates, structural member, forgings, and pipe associated with the drywell have an initial NDT temperature of approximately 0° F when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30° F. Provisions are made for the removal of heat from within the plant containment system and to isolate the various process system lines as may be necessary to maintain the integrity of the plant containment systems as long as necessary following the various postulated design basis accidents. The plant containment is designed and maintained so that the off-site doses resulting from the postulated design basis accident will be below the values stated in 10CFR 100 (Criterion 50, 51, 54). All pipes or ducts, which penetrate the primary containment and which connect to the reactor coolant system or to the drywell, are provided with at least two isolation valves in series (Criterion 53). The plant design provides for preoperational pressure and leak rate testing of the primary containment system, and include the capability for leak testing at design pressure after the plant has commenced operation (Criterion 54, 55). Provisions are also made for demonstrating the functional performance of the plant containment system isolation valves and leak testing of selected penetrations (Criterion 56, 57).

The pressure suppression pool and the containment spray cooling system provide two different means to rapidly condense the steam portion of the flow from the postulated design basis loss-of-coolant accident so that the peak transient pressure shall be substantially less than the primary containment design pressure (Criterion 52). Demonstration of operability and the ability to test the functional performance and inspect the containment spray/cooling system are provided (Criterion 58, 59, 60, 61). The secondary containment standby gas treatment system is designed such that means are provided for periodic testing of the system performance including tracer injection and sampling (Criterion 64). The system may be physically inspected and its operability demonstrated (Criterion 62, 63, 65).

References to applicable sections of the USAR are given below for the individual criteria of this group.

Criterion 37 - Engineered Safety Features Basis for Design (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Conformance 37 - Engineered Safety Features Basis for Design (Category A)

Section 1.2.3	▪ Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪ Principal Design Criteria - Plant Containment
Section 1.2.5	▪ Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.6	▪ Principal Design Criteria - Plant Electrical Power
Section 1.3.2	▪ Summary Design Description and Safety Analysis - Reactor System
Section 1.3.3	▪ Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪ Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 1.3.5	▪ Summary Design Description and Safety Analysis - Plant Instrumentation Control Systems
Section 1.3.8	▪ Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 5 Complete	▪ Containment System
Section 6 Complete	▪ Plant Engineered Safeguards
Section 7 Complete	▪ Plant Instrumentation and Control Systems
Section 8 Complete	▪ Plant Electrical Systems
Section 10.3.8	▪ Plant Service Systems - Plant Communication System
Section 10.3.9	▪ Plant Service Systems - Plant Lighting System
Section 14.1.5	▪ Summary Description - Design Basis for Accidents

Criterion 38 - Reliability and Testability of Engineered Safety Features (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineering safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Conformance 38 - Reliability and Testability of Engineered Safety Features (Category A)

Section 1.2.2	▪ Principal Design Criteria - Reactor Core
Section 1.2.3	▪ Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪ Principal Design Criteria - Plant Containment
Section 1.2.5	▪ Principal Design Criteria - Plant Instrumentation and Control
Section 1.3.2	▪ Summary Design Description and Safety Analysis - Reactor System
Section 1.3.3	▪ Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪ Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 1.3.5	▪ Summary Design Description and Safety Analysis - Plant Instrumentation Control Systems
Section 5 Complete	▪ Containment System
Section 6 Complete	▪ Plant Engineered Safeguards
Section 7 Complete	▪ Plant Instrumentation and Control Systems
Section 8 Complete	▪ Plant Electrical Systems
Section 10.3.8	▪ Plant Service Systems - Plant Communication System
Section 10.3.9	▪ Plant Service Systems - Plant Lighting System

Criterion 39 - Emergency Power for Engineered Safety Features (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Conformance 39 - Emergency Power for Engineered Safety Features (Category A)

Section 1.2.6	▪	Principal Design Criteria - Plant Electrical Power
Section 1.3.8	▪	Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 8.2	▪	Transmission System
Section 8.3	▪	Auxiliary Power System
Section 8.4	▪	Plant Standby Diesel Generator Systems
Section 8.5	▪	D-C Power Supply Systems
Section 8.6	▪	Reactor Protection System Power Supplies

Criterion 40 - Missile Protection (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from the plant equipment failures.

Conformance 40 - Missile Protection (Category A)

Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.3.5	▪	Secondary Containment System - Performance Analysis
Section 12 Complete	▪	Plant Structures and Shielding

Criterion 41 - Engineered Safety Features Performance Capability (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Conformance 41 - Engineered Safety Features Performance Capability (Category A)

Section 1.2.3	▪ Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪ Principal Design Criteria - Plant Containment
Section 1.2.5	▪ Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.6	▪ Principal Design Criteria - Plant Electrical Power
Section 1.3.3	▪ Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪ Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 1.3.8	▪ Summary Design Description and Safety Analysis - Plant Electrical Power Systems
Section 5.2.1	▪ Primary Containment System - Design Criteria
Section 5.3.2	▪ Secondary Containment System - Design Basis
Section 6.2.1.1	▪ Emergency Core Cooling System (ECCS) - ECCS Design Basis
Section 6.2.4.3	▪ High Pressure Coolant Injection System (HPCI) - Performance Analysis
Section 6.2.5.3	▪ Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.2.3	▪ Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.3.3	▪ Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.6	▪ Emergency Core Cooling System (ECCS) - ECCS Performance Evaluation

Section 6.3	▪	Main Steam Line Flow Restrictions
Section 6.4	▪	Control Rod Velocity Limiters
Section 6.5	▪	Control Rod Drive Housing Supports
Section 6.6	▪	Standby Liquid Control System
Section 8.2	▪	Transmission System
Section 8.3	▪	Auxiliary Power Systems
Section 8.4	▪	Plant Standby Diesel Generator Systems
Section 8.5	▪	D-C Power Supply Systems
Section 8.6	▪	Reactor Protection System Power Supplies
Section 10.3.4	▪	Plant Service Systems - Plant Instrumentation and Service Air Systems
Section 10.3.8	▪	Plant Service Systems - Plant Communication System
Section 10.3.9	▪	Plant Service Systems - Plant Lighting System
Section 14.1.5	▪	Summary Description - Design Basis for Accidents

Criterion 42 - Engineered Safety Features Components Capability (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Conformance 42 - Engineered Safety Features Components Capability (Category A)

Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 1.2.5	▪	Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.6	▪	Principal Design Criteria - Plant Electrical Power
Section 3.6	▪	Other Reactor Vessel Internals

Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 6 Complete	▪	Plant Engineered Safeguards
Section 7.4	▪	Reactor Vessel Instrumentation
Section 7.6	▪	Plant Protection System
Section 12 Complete	▪	Plant Structures and Shielding
Section 14.1.5	▪	Summary Description - Design Basis Accident Analysis

Criterion 43 - Accident Aggravation Prevention (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse affects of the loss of normal cooling avoided.

Conformance 43 - Accident Aggravation Prevention (Category A)

Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 6.2.1.1	▪	Emergency Core Cooling System (ECCS) - ECCS Design Basis
Section 6.2.4.3	▪	High Pressure Coolant Injection System (HPCI) - Performance Analysis
Section 6.2.5.3	▪	Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.2.3	▪	Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.3.3	▪	Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.6	▪	Emergency Core Cooling System (ECCS) - ECCS Performance Evaluation
Section 6.3.1	▪	Main Steam Line Flow Restrictions - Design Basis
Section 6.4.1	▪	Control Rod Velocity Limiters - Design Basis

Section 6.5.1 ▪ Control Rod Drive Housing Supports - Design Basis

Section 6.6.1 ▪ Standby Liquid Control System - Design Basis

Criterion 44 - Emergency Core Cooling System Capability (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts of all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or components to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Conformance 44 - Emergency Core Cooling Systems Capability (Category A)

Section 1.2.3 ▪ Principal Design Criteria - Reactor Core Cooling

Section 1.3.4 ▪ Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems

Section 6.2.1.2 ▪ Emergency Core Cooling System (ECCS) - Description and Function of ECCS

Section 6.2.2.1 ▪ Reactor Core Spray Cooling System (CSCS) - Design Basis

Section 6.2.3.1 ▪ Residual Heat Removal System (RHR) - Design Basis

Section 6.2.4.1 ▪ High Pressure Coolant Injection System (HPCI) - Design Basis

Section 6.2.5.1 ▪ Automatic Depressurization System (ADS) - Design Basis

Section 6.2.6 ▪ Emergency Core Cooling System (ECCS) - ECCS Performance Evaluation

Section 14.1.5 ▪ Summary Description - Design Basis for Accidents

Criterion 45 - Inspection of Emergency Core Cooling Systems (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Conformance 45 - Inspection of Emergency Core Cooling Systems (Category A)

Section 3.6.1	▪	Other Reactor Vessel Internals - Design Basis
Section 6.2.2.4	▪	Reactor Core Spray Cooling System (CSCS) - Inspection and Testing
Section 6.2.3.4	▪	Residual Heat Removal System (RHR) - Inspection and Testing
Section 6.2.4.4	▪	High Pressure Coolant Injection System (HPCI) - Inspection and Testing
Section 6.2.5.4	▪	Automatic Depressurization System (ADS) - Inspection and Testing

Criterion 46 - Testing of Emergency Core Cooling Systems Components (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and require functional performance.

Conformance 46 - Testing of Emergency Core Cooling Systems Components (Category A)

Section 6.2.1.1	▪	Emergency Core Cooling System (ECCS) - ECCS Design Basis
Section 6.2.2.1	▪	Reactor Core Spray Cooling System (CSCS) - Design Basis
Section 6.2.2.3	▪	Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.2.4	▪	Reactor Core Spray Cooling System (CSCS) - Inspection and Testing
Section 6.2.4.1	▪	High Pressure Coolant Injection System (HPCI)- Design Basis
Section 6.2.4.3	▪	High Pressure Coolant Injection System (HPCI) - Performance Analysis

Section 6.2.4.4	▪	High Pressure Coolant Injection System (HPCI) - Inspection and Testing
Section 6.2.3.1	▪	Residual Heat Removal System (RHR) - Design Basis
Section 6.2.3.3	▪	Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.3.4	▪	Residual Heat Removal System (RHR) - Inspection and Testing
Section 6.2.5.1	▪	Automatic Depressurization System (ADS) - Design Basis
Section 6.2.5.3	▪	Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.5.4	▪	Automatic Depressurization System (ADS) - Inspection and Testing

Criterion 47 - Testing of Emergency Core Cooling Systems (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Conformance 47 - Testing of Emergency Core Cooling Systems (Category A)

Section 6.2.1.1	▪	Emergency Core Cooling System (ECCS) - ECCS Design Basis
Section 6.2.2.1	▪	Reactor Core Spray Cooling System (CSCS) - Design Basis
Section 6.2.2.3	▪	Reactor Core Spray Cooling System (CSCS) - Performance Analysis
Section 6.2.2.4	▪	Reactor Core Spray Cooling System (CSCS) - Inspection and Testing
Section 6.2.4.1	▪	High Pressure Coolant Injection System (HPCI)- Design Basis
Section 6.2.4.3	▪	High Pressure Coolant Injection System (HPCI) - Performance Analysis
Section 6.2.4.4	▪	High Pressure Coolant Injection System (HPCI) - Inspection and Testing

Section 6.2.3.1	▪	Residual Heat Removal System (RHR) - Design Basis
Section 6.2.3.3	▪	Residual Heat Removal System (RHR) - Performance Analysis
Section 6.2.3.4	▪	Residual Heat Removal System (RHR) - Inspection and Testing
Section 6.2.5.1	▪	Automatic Depressurization System (ADS) - Design Basis
Section 6.2.5.3	▪	Automatic Depressurization System (ADS) - Performance Analysis
Section 6.2.5.4	▪	Automatic Depressurization System (ADS) - Inspection and Testing

<u>Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling System (Category A)</u>

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

<u>Conformance 48 - Testing of Operational Sequence of Emergency Core Cooling System (Category A)</u>

Section 6.2	▪	Emergency Core Cooling System (ECCS)
Section 8 Complete	▪	Plant Electrical Systems
Section 8.2	▪	Transmission System
Section 8.3	▪	Auxiliary Power System
Section 8.4	▪	Plant Standby Diesel Generator Systems
Section 8.5	▪	D-C Power Supply Systems
Section 8.6	▪	Reactor Protection System Power Supplies
Section 10.4	▪	Plant Cooling System

<u>Criterion 49 - Containment Design Basis (Category A)</u>

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

<u>Conformance 49 - Containment Design Basis (Category A)</u>

Section 1.2.2	▪	Principal Design Criteria - Reactor Core
Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 1.3.3	▪	Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 1.3	▪	Summary Design Description and Safety Analysis
Section 5.1	▪	Containment System - Summary Description
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 5.3.2	▪	Secondary Containment System - Design Basis
Section 5.3.5	▪	Secondary Containment System - Performance Analysis
Section 5.3.6	▪	Secondary Containment System - Inspection and Testing
Section 6.2	▪	Emergency Core Cooling System (ECCS)
Section 6.6	▪	Standby Liquid Control System
Section 10.2.5	▪	Reactor Auxiliary Systems - Reactor Core Isolation Cooling System (RCIC)
Section 14.1.5	▪	Summary Description - Design Basis for Accident

Analysis

Criterion 50 - NDT Requirement for Containment Material (Category A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

Conformance 50 - NDT Requirement for Containment Material (Category A)
Section 5.2.2.2 - Primary Containment Construction Materials

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Conformance 51 - Reactor Coolant Pressure Boundary Outside Containment (Category A)

Section 1.2.1	▪ Principal Design Criteria - General Criteria
Section 1.2.4	▪ Principal Design Criteria - Plant Containment
Section 1.2.5	▪ Principal Design Criteria - Plant Instrumentation and Control
Section 1.2.6	▪ Principal Design Criteria - Plant Electrical Power
Section 1.3.2	▪ Summary Design Description and Safety Analysis - Reactor System
Section 1.3.3	▪ Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.5	▪ Summary Design Description and Safety Analysis - Plant Instrumentation Control Systems
Section 1.3.8	▪ Summary Design Description and Safety Analysis - Plant Electrical Power System
Section 1.3.11	▪ Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 2.2	▪ Site Description

Section 5.2	▪	Primary Containment System
Section 5.3	▪	Secondary Containment System
Section 6.3	▪	Main Steam Line Floor Restrictions
Section 7.5.2	▪	Plant Radiation Monitoring Systems - Process Radiation Monitoring System
Section 7.6.3	▪	Plant Protection System - Primary Containment Isolation System
Section 14.1.5	▪	Summary Description - Design Basis for Accident Analysis

Criterion 52 - Containment Heat Removal Systems (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Conformance 52 - Containment Heat Removal Systems (Category A)

Section 1.2.3	▪	Principal Design Criteria - Reactor Core Cooling
Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 1.3.2	▪	Summary Design Description and Safety Analysis - Reactor System
Section 1.3.3	▪	Summary Design Description and Safety Analysis - Plant Containment System
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling Systems
Section 5.2	▪	Primary Containment System
Section 6.2	▪	Emergency Core Cooling System (ECCS)
Section 10.2	▪	Reactor Auxiliary Systems
Section 10.4	▪	Plant Cooling System
Section 14.1.5	▪	Summary Description - Design Basis for Accident Analysis

Criterion 53 - Containment Isolation Valves (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Conformance 53 - Containment Isolation Valves (Category A)

Section 5.2.1.3	▪	Primary Containment System - Containment Penetrations
Section 5.2.2.5.3	▪	Primary Containment System - Isolation System
Section 5.2.3.7	▪	Primary Containment System - Penetrations
Section 5.2.3.6.2	▪	Primary Containment System - Isolation System
Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 7.6.3	▪	Plant Protection System - Primary Containment Isolation System

Criterion 54 - Containment Leakage Rate Testing (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Conformance 54 - Containment Leakage Rate Testing (Category A)

Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 5.3.2	▪	Secondary Containment System - Design Basis
Section 5.3.5	▪	Secondary Containment System - Performance Analysis
Section 5.3.6	▪	Secondary Containment System - Inspection and Testing

Criterion 55 - Containment Periodic Leakage Rate Testing (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Conformance 55 - Containment Periodic Leakage Rate Testing (Category A)

Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.3.2	▪	Secondary Containment System - Design Basis

Criterion 56 - Provisions for Testing of Penetrations (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at anytime.

Conformance 56 - Provisions for Testing of Penetrations (Category A)

Section 5.2.1	▪	Primary Containment System - Design Criteria
Section 5.2.3	▪	Primary Containment System - Performance Analysis
Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 5.3.5	▪	Secondary Containment System - Performance Analysis
Section 5.3.6	▪	Secondary Containment System - Inspection and Testing

Criteria 57 - Provisions for Testing of Isolation Valves (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Conformance 57 - Provisions for Testing of Isolation Valves (Category A)

Section 7.6.3.1	▪	Plant Protection System - Design Basis
Section 7.6.3.3	▪	Plant Protection System - Performance Analysis
Section 7.6.3.4	▪	Plant Protection System - Inspection and Testing
Section 7.5.2	▪	Plant Radiation Monitoring Systems - Process Radiation Monitoring System

Criterion 58 - Inspection of Containment Pressure-Reducing System (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Conformance 58 - Inspection of Containment Pressure-Reducing System (Category A)

Section 5.2.4	▪	Primary Containment System - Inspection and Testing
Section 6.2	▪	Emergency Core Cooling System (ECCS)

Criterion 59 - Testing of Containment Pressure-Reducing Systems Components (Category A)

The containment pressure-reducing systems shall be designed so that active components such as pumps and valves can be tested periodically for operability and required functional performance.

Conformance 59 - Testing of Containment Pressure-Reducing Systems Components (Category A)

Section 6.2.1.1	▪	Emergency Core Cooling System (ECCS) - Design Basis
Section 6.2	▪	Emergency Core Cooling System (ECCS)

Criterion 60 - Testing of Containment Spray Systems (Category A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzle as is practical.

Conformance 60 - Testing of Containment Spray Systems (Category A)

- Section 6.2.1.1 ▪ Emergency Core Cooling System (ECCS) - Design Basis
- Section 6.2 ▪ Emergency Core Cooling System (ECCS)

Criterion 61 - Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A)

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Conformance 61 - Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A)

- Section 5.2 Complete ▪ Primary Containment System
- Section 7.6.3.3 ▪ Plant Protection System - Performance Analysis
- Section 7.6.3.4 ▪ Plant Protection System - Inspection and Testing
- Section 6.2.1.1 ▪ Emergency Core Cooling System (ECCS) - Design Basis
- Section 6.2 ▪ Emergency Core Cooling System (ECCS)
- Section 8 Complete ▪ Plant Electrical Systems

Criterion 62 - Inspection of Air Cleanup Systems (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems such as ducts, filters, fans, and dampers.

Conformance 62 - Inspection of Air Cleanup Systems (Category A)

- Section 5.3.4.1 ▪ Secondary Containment System - Standby Gas Treatment System (SGTS)
- Section 5.3.5 ▪ Secondary Containment System - Performance Analysis

Section 5.3.6 ▪ Secondary Containment System - Inspection and Testing

Section 10.3.2 ▪ Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems

Criterion 63 - Testing of Air Cleanup Components (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans, dampers, can be tested periodically for operability and required functional performance.

Conformance 63 - Testing of Air Cleanup Components (Category A)

Section 5.3.4.1 ▪ Secondary Containment System - Standby Gas Treatment System (SGTS)

Section 5.3.5 ▪ Secondary Containment System - Performance Analysis

Section 5.3.6 ▪ Secondary Containment System - Inspection and Testing

Section 10.3.2 ▪ Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems

Criterion 64 - Testing of Air Cleanup Systems (Category A)

A capability shall be provided for insitu periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Conformance 64 - Testing of Air Cleanup Systems (Category A)

Section 5.3.4.1 ▪ Secondary Containment System - Standby Gas Treatment System (SGTS)

Section 5.3.5 ▪ Secondary Containment System - Performance Analysis

Section 5.3.6 ▪ Secondary Containment System - Inspection and Testing

Section 10.3.2 ▪ Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems

Criterion 65 - Testing of Operational Sequence Air Cleanup Systems (Category A)

A capability shall be provided to test under conditions close to design as practical the full operational sequence that would bring the air cleanup systems to action, including the transfer to alternate power sources and the design air flow delivery capability.

Conformance 65 - Testing of Operational Sequence Air Cleanup Systems (Category A)

Section 5.3.4.1	▪ Secondary Containment System - Standby Gas Treatment System (SGTS)
Section 5.3.5	▪ Secondary Containment System - Performance Analysis
Section 5.3.6	▪ Secondary Containment System - Inspection and Testing
Section 7.5.2	▪ Plant Radiation Monitoring Systems - Process Radiation Monitoring System
Section 7.6.1	▪ Plant Protection System - Reactor Protection System
Section 8.4	▪ Plant Standby Diesel Generator Systems
Section 8.5	▪ D-C Power Supply Systems
Section 8.6	▪ Reactor Protection System Power Supplies
Section 10.3.2	▪ Plant Service Systems - Plant Heating, Ventilating and Air Conditioning Systems

E.2.8 Group VIII - Fuel and Waste Storage Systems

The intent of the current draft of the proposed criteria for this group is to establish the safe fuel and waste storage systems design and to identify the means used to satisfy these requirements.

It is concluded that the design of this plant is in conformance with criteria of Group VIII based on NSP's current understanding of the intent of these criteria.

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality and to provide sufficient cooling for spent fuel. (Criterion 66, 67) The new fuel storage vault racks (located inside the secondary containment reactor building) are top entry, and are designed to prevent an accidental critical array, even in the event the vault becomes flooded. Vault drainage is provided to prevent possible water collection. (Criterion 66) The handling and storage of spent fuel, which takes place entirely within the reactor building (which provides containment), is done in the spent fuel storage pool. The pool has provisions to maintain water clarity, temperature control, and

instrumentation to monitor water level. Water depth in the pool will be such as to provide sufficient shielding for normal reactor building occupancy (10 CFR 20) by operating personnel. The storage racks in which spent fuel assemblies are placed are designed and arranged to ensure subcriticality in the storage pool. (Criterion 66, 67, 68, 69) The spent fuel pool cooling and demineralizer system is designed to maintain the pool water temperature (decay heat removal) to control water clarity (safe fuel movement), and to reduce water radioactivity (shielding and effluent release control). (Criterion 66, 67, 68) Accessible portions of the reactor and radwaste buildings shall have sufficient shielding to maintain dose rates within 10 CFR 20. (Criterion 68) The radwaste building is designed to preclude accidental release of radioactive materials to the environs. (Criterion 69) The spent fuel storage pool and racks are designed and constructed such that all credible missiles as a result of a design basis tornado and tornado itself, will not have radiological effects exceeding 10 CFR 100 guideline limitations.

References to applicable sections of the USAR are given below for the individual criteria of this group. (Criterion 67, 69)

Criterion 66 - Prevention of Fuel Storage Critically (Category B)

Critically in new and spent storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Conformance 66 - Prevention of Fuel Storage Critically (Category B)

Section 1.2.9	▪	Principal Design Criteria - Plant Fuel Handling and Storage
Section 1.3.6	▪	Summary Design Description and Safety Analysis - Plant Fuel Storage and Handling Systems
Section 6.6.3	▪	Standby Liquid Control System - Performance Analysis
Section 10.2.1.1	▪	Reactor Auxiliary Systems - Design Basis
Section 10.2.1.2	▪	Reactor Auxiliary Systems - Description

Criterion 67 - Fuel and Waste Storage Decay Heat (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Conformance 67 - Fuel and Waste Storage Decay Heat (Category B)

Section 1.2.7	▪	Principal Design Criteria - Plant Radioactive Waste Disposal

Section 1.2.9	▪	Principal Design Criteria - Plant Fuel Handling and Storage
Section 1.3.4	▪	Summary Design Description and Safety Analysis - Plant Auxiliary and Standby Cooling system
Section 1.3	▪	Summary Design Description and Safety Analysis
Section 6.2.1.2	▪	Emergency Core Cooling System (ECCS) - Description and Function of ECCS
Section 10.2.1	▪	Reactor Auxiliary Systems - Fuel Storage and Fuel Handling Systems
Section 10.2.2	▪	Reactor Auxiliary Systems - Spent Fuel Pool Cooling and Demineralizer System
Section 10.2.3	▪	Reactor Auxiliary Systems - Reactor Cleanup Demineralizer System
Section 10.2.4	▪	Reactor Auxiliary Systems - Reactor Shutdown Cooling System
Section 12 Complete	▪	Plant Structures and Shielding

Criterion 68 - Fuel and Waste Storage Radiation Shielding (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet requirements of 10 CFR 20.

Conformance 68 - Fuel and Waste Storage Radiation Shielding (Category B)

Section 1.2.8	▪	Principal Design Criteria - Plant Shielding and Access Control
Section 1.3.6	▪	Summary Design Description and Safety Analysis - Plant Fuel Storage and Handling Systems
Section 1.3.9	▪	Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures
Section 1.3.10	▪	Summary Design Description and Safety Analysis - Plant Radioactive Waste Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety

Section 12.3	▪	Shielding And Radiation Protection
Section 9.2.1	▪	Liquid Radwaste System - Design Basis
Section 9.2.3	▪	Liquid Radwaste System - Performance Analysis
Section 9.3.1	▪	Gaseous Radwaste System - Design Basis
Section 9.3.3	▪	Gaseous Radwaste System - Performance Analysis
Section 9.4.1	▪	Solid Radwaste System - Design Basis
Section 9.4.3	▪	Solid Radwaste System - Performance Analysis
Section 10.2.1.1	▪	Reactor Auxiliary Systems - Design Basis
Section 10.2.1.2	▪	Reactor Auxiliary Systems - Description
Section 10.2.1.3	▪	Reactor Auxiliary Systems - Performance Analysis

<u>Criterion 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category B)</u>

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

<u>Conformance 69 - Protection Against Radioactivity Release from Spent Fuel and Waste Storage (Category B)</u>

Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 1.2.8	▪	Principal Design Criteria - Plant Shielding and Access Control
Section 1.3.6	▪	Summary Design Description and Safety Analysis - Plant Fuel Storage and Handling Systems
Section 1.3.9	▪	Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures
Section 1.3.10	▪	Summary Design Description and Safety Analysis - Plant Radioactive Waste Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety

Section 5.1	▪	Containment System - Summary Description
Section 5.3	▪	Secondary Containment System
Section 9 Complete	▪	Plant Radioactive Waste Control Systems
Section 10.2.1	▪	Reactor Auxiliary Systems - Fuel Storage and Fuel Handling Systems
Section 10.2.2	▪	Reactor Auxiliary Systems - Spent Fuel Pool Cooling and Demineralizer System
Section 1.2.7	▪	Principal Design Criteria - Plant Radioactive Waste Disposal
Section 1.2.8	▪	Principal Design Criteria - Plant Shielding and Access Control
Section 14.7.6.4.2	▪	Refueling Accident Analysis - Radiological Consequences
Section 14.7.4	▪	Accident Evaluation Methodology - Fuel Loading Error Accident

E.2.9 <u>Group IX - Plant Effluents</u>

The intent of the current draft of the proposed criterion for this group is to establish the plant effluent release limits and to identify the means of controlling the releases within these guide limits.

It is concluded that the design of this plant is in conformance with the criteria of Group IX based on NSP's current understanding of the intent of these criteria.

The plant radioactive waste control systems (which include the liquid, gaseous and solid radwaste sub-systems) are designed to limit the off-site radiation exposure to levels below doses set forth in 10 CFR 20. The plant engineered safety systems (including the containment barriers) are designed to limit the off-site dose under various postulated "design basis" accidents to levels significantly below the limits of 10 CFR 100. The air ejector off-gas system is designed with sufficient holdup retention capacity so that during normal plant operation the controlled release of radioactive materials does not exceed the established release limits at the elevated plant stack. (Criterion 70)

References to applicable sections of the USAR are given for the individual criteria of this group.

Criterion 70 - Control of Release of Radioactivity to the Environment (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Conformance 70 - Control of Release of Radioactivity to the Environment (Category B)

Section 1.2.4	▪	Principal Design Criteria - Plant Containment
Section 1.2.7	▪	Principal Design Criteria - Plant Radioactive Waste Disposal
Section 1.2.8	▪	Principal Design Criteria - Plant Shielding and Access Control
Section 1.3.9	▪	Summary Design Description and Safety Analysis - Plant Shielding, Access Control, and Radiation Protection Procedures
Section 1.3.10	▪	Summary Design Description and Safety Analysis - Plant Radioactive Waste Control Systems
Section 1.3.11	▪	Summary Design Description and Safety Analysis - Summary Evaluation of Plant Safety
Section 2.2	▪	Site Description
Section 5 Complete	▪	Containment System
Section 12 Complete	▪	Plant Structures and Shielding
Section 7.5	▪	Plant Radiation Monitoring Systems
Section 8 Complete	▪	Plant Electrical Systems
Section 9 Complete	▪	Plant Radioactive Waste Control Systems
Section 10.3.6	▪	Plant Service Systems - Plant Equipment and Floor Drainage Systems

Section 10.3.7	▪	Plant Service Systems - Plant Process Sampling System
Section 11.3.2	▪	Main Condenser System - Main Condenser Gas Removal System
Section 13 Complete	▪	Plant Operations
Section 14 Complete	▪	Plant Safety Analysis