APPENDIX F

CONFORMANCE TO AEC PROPOSED GENERAL DESIGN CRITERIA

		PAGE
1.0	SUMMARY DESCRIPTION	F-1-1
2.0	CRITERION CONFORMANCE	F-2-1
2.1	Group I -- Overall Plant Requirements (Criteria 1-5)	F-2-1
2.1.1	Criterion 1 -- Quality Standards	F-2-1
2.1.2	Criterion 2 -- Performance Standards	F-2-1
2.1.3	Criterion 3 -- Fire Protection	F-2-2
2.1.4	Criterion 5 -- Records Requirement	F-2-2
2.2	Group II -- Protection by Multiple Fission Product Barriers (Criteria 6-10)	F-2-2
2.2.1	Criterion 6 -- Reactor Core Design	F-2-3
2.2.2	Criterion 7 -- Suppression of Power Oscillations	F-2-3
2.2.3	Criterion 8 -- Overall Power Coefficient	F-2-3
2.2.4	Criterion 9 -- Reactor Coolant Pressure Boundary	F-2-4
2.2.5	Criterion 10 -- Containment	F-2-4
2.3	Group III -- Nuclear and Radiation Controls (Criteria 11-18)	F-2-4
2.3.1	Criterion 11 -- Control Room	F-2-5
2.3.2	Criterion 12 -- Instrumentation and Control Systems	F-2-5
2.3.3	Criterion 13 -- Fission Process Monitors and Control	F-2-5
2.3.4	Criterion 14 -- Core Protection Systems	F-2-6
2.3.5	Criterion 15 -- Engineered Safety Features Protection Systems	F-2-6
2.3.6	Criterion 16 -- Monitoring Reactor Coolant Pressure Boundary	F-2-6
2.3.7	Criterion 17 -- Monitoring Radioactivity Releases	F-2-6
2.3.8	Criterion 18 -- Monitoring Fuel and Waste Storage	F-2-7
2.4	Group IV -- Reliability and Testability of Protection Systems (Criteria 19-26)	F-2-7
2.4.1	Criterion 19 -- Protection Systems Reliability	F-2-7
2.4.2	Criterion 20 -- Protection Systems Redundancy and Independence	F-2-7
2.4.3	Criterion 21 -- Single Failure Definition	F-2-8
2.4.4	Criterion 22 -- Separation of Protection and Control Instrumentation Systems	F-2-8
2.4.5	Criterion 23 -- Protection Against Multiple Disability for Protection Systems	F-2-8
2.4.6	Criterion 24 -- Emergency Power for Protection Systems	F-2-9
2.4.7	Criterion 25 -- Demonstration of Functional Operability of Protection Systems	F-2-9
2.4.8	Criterion 26 -- Protection Systems Fail-Safe Design	F-2-9
2.5	Group V -- Reactivity Control (Criteria 27-32)	F-2-10
2.5.1	Criterion 27 -- Redundancy of Reactivity Control	F-2-10
2.5.2	Criterion 28 -- Reactivity Hot Shutdown Capability	F-2-10
2.5.3	Criterion 29 -- Reactivity Shutdown Capability	F-2-11
2.5.4	Criterion 30 -- Reactivity Holddown Capability	F-2-11
2.5.5	Criterion 31 -- Reactivity Control Systems Malfunction	F-2-11
2.5.6	Criterion 32 -- Maximum Reactivity Worth of Control Rods	F-2-11
2.6	Group VI -- Reactor Coolant Pressure Boundary (Criteria 33-36)	F-2-12

2.6.1 Criterion 33 -- Reactor Coolant Pressure Boundary Capability F-2-12
2.6.2 Criterion 34 -- Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention F-2-12
2.6.3 Criterion 35 -- Reactor Coolant Pressure Boundary Brittle Fracture Prevention F-2-13
2.6.4 Criterion 36 -- Reactor Coolant Pressure Boundary Surveillance F-2-14
2.7 Group VII -- Engineered Safety Features (Criteria 37-65) F-2-14
2.7.1 General Requirements for Engineered Safety Features (Criteria 37-43) F-2-15
2.7.1.1 Criterion 37 -- Engineered Safety Features Basis for Design F-2-15
2.7.1.2 Criterion 38 -- Reliability and Testability of Engineered Safety Features F-2-15
2.7.1.3 Criterion 39 -- Emergency Power for Engineered Safety Features F-2-15
2.7.1.4 Criterion 40 -- Missile Protection F-2-16
2.7.1.5 Criterion 41 -- Engineered Safety Features Performance Capability F-2-16
2.7.1.6 Criterion 42 -- Engineered Safety Features Components Capability F-2-16
2.7.1.7 Criterion 43 -- Accident Aggravation Prevention F-2-17
2.7.2 Emergency Core Cooling Systems (Criteria 44-48) F-2-17
2.7.2.1 Criterion 44 -- Emergency Core Cooling Systems Capability F-2-17
2.7.2.2 Criterion 45 -- Inspections of Emergency Core Cooling Systems F-2-17
2.7.2.3 Criterion 46 -- Testing of Emergency Core Cooling Systems Components F-2-18
2.7.2.4 Criterion 47 -- Testing of Emergency Core Cooling Systems F-2-18
2.7.2.5 Criterion 48 -- Testing of Operational Sequence of Emergency Core Cooling Systems F-2-18
2.7.3 Containment (Criteria 49-57) F-2-18
2.7.3.1 Criterion 49 -- Containment Design Basis F-2-18
2.7.3.2 Criterion 50 -- NDT Requirement for Containment Material F-2-19
2.7.3.3 Criterion 51 -- Reactor Coolant Pressure Boundary Outside Containment F-2-19
2.7.3.4 Criterion 52 -- Containment Heat Removal Systems F-2-20
2.7.3.5 Criterion 53 -- Containment Isolation Valves F-2-20
2.7.3.6 Criterion 54 -- Containment Leakage Rate Testing F-2-20
2.7.3.7 Criterion 55 -- Containment Periodic Leakage Rate Testing F-2-20
2.7.3.8 Criterion 56 -- Provisions for Testing of Penetrations F-2-21
2.7.3.9 Criterion 57 -- Provisions for Testing of Isolation Valves F-2-21
2.7.4 Containment Pressure-Reducing Systems (Criteria 58-61) F-2-21
2.7.4.1 Criterion 58 -- Inspection of Containment Pressure-Reducing Systems F-2-21
2.7.4.2 Criterion 59 -- Testing of Containment Pressure-Reducing Systems Components F-2-21
2.7.4.3 Criterion 60 -- Testing of Containment Spray Systems F-2-22

2.7.4.4 Criterion 61 -- Testing of Operational Sequence of Containment Pressure-Reducing Systems F-2-22
2.7.5 Air Cleanup Systems (Criteria 62-65) F-2-22
2.7.5.1 Criterion 62 -- Inspection of Air Cleanup Systems F-2-22
2.7.5.2 Criterion 63 -- Testing of Air Cleanup Systems Components F-2-22
2.7.5.3 Criterion 64 -- Testing of Air Cleanup Systems F-2-23
2.7.5.4 Criterion 65 -- Testing of Operational Sequence of Air Cleanup Systems F-2-23
2.8 Group VIII -- Fuel and Waste Storage Systems (Criteria 66-69) F-2-23
2.8.1 Criterion 66 -- Prevention of Fuel Storage Criticality F-2-23
2.8.2 Criterion 67 -- Fuel and Waste Storage Decay Heat F-2-24
2.8.3 Criterion 68 -- Fuel and Waste Storage Radiation Shielding F-2-24
2.8.4 Criterion 69 -- Protection Against Radioactivity Release from Spent Fuel and Waste Storage F-2-24
2.9 Group IX -- Plant Effluents (Criterion 70) F-2-25
2.9.1 Criterion 70 -- Control of Releases of Radioactivity to the Environment F-2-25

3.0 REFERENCES FOR APPENDIX F F-3-1

APPENDIX F

CONFORMANCE TO AEC PROPOSED GENERAL DESIGN CRITERIA

1.0 SUMMARY DESCRIPTION

The proposed 70 General Design Criteria for Nuclear Power Plant Construction Permits were issued in July of 1967 to serve as a guide in the establishment of design criteria and bases for the design and construction of a nuclear power station. It is the purpose of this appendix to show that the design and construction of the Cooper Nuclear Station has been performed in accordance with these general design criteria.

It should be recognized that these criteria, which appeared in the July 11, 1967 issue of the Federal Register, were issued in order to secure comments from the nuclear industry, and at that time had not yet been adopted as regulatory requirements. It was anticipated that revisions and clarifications would take place prior to such adoption. The Atomic Energy Commission (AEC) accepted Cooper Nuclear Station's conformance with the proposed general design criteria during the proceedings for a Construction Permit with the exception of Criterion 35 which was later resolved with the Advisory Committee on Reactor Safeguards. However, the AEC indicated that their reviews for the Operating License stage would be made in light of the degree of conformance with the latest version of the criteria to be formulated[1]. The comparison which follows is presented to show that the concerns expressed by the proposed general design criteria, as interpreted by NPPD, have been fully considered in the design of the station. After performing the technical evaluation of the CNS application for an Operating License, the AEC concluded that the intent of the criteria contained in the 1971 10CFR50 Appendix A Final Rule had also been met[2]. Notwithstanding that evaluation, the 1967 Proposed General Design Criteria (as interpreted by NPPD and to the degree of conformance described in this Appendix) is the licensing basis of the plant except where specific commitments have been made to 1971 GDCs and have been described in the USAR.

The method of presentation is to consider the criteria in nine groups. The grouping of the criteria is that given in the above referenced draft. For each group, a statement of NPPD's understanding of the intent of the criteria in the group is given along with discussion of conformance which is applicable to all of the criteria within the group. Each criterion in the group is then discussed as necessary to enlarge upon the general statements and a list of references where the subject material of the individual criterion is found in the CNS USAR is presented. The statements of the criteria are presented as listed in the July 11, 1967 Federal Register.

2.0 CRITERION CONFORMANCE

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion on historical information. The factual information being presented in this section as historical has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR.

2.1 Group -- Overall Plant Requirements (Criteria 1-5)

The purpose of these criteria is to insure that those systems and components of the station which have a vital role in the prevention or mitigation of consequences of accidents affecting public health and safety are designed and constructed to high quality standards which include consideration of natural phenomena and fire. Also, there must be sufficient surveillance and record keeping during fabrication and construction to ensure that these high quality standards have been met. As the station consists of a single nuclear plant, Criterion 4, Sharing of Systems, is not applicable. It will be seen that the concerns of these criteria have been properly considered throughout the design of the station.

2.1.1 Criterion 1 -- Quality Standards

"Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of 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, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required."

A thorough quality assurance program has been undertaken during design and construction of the station to ensure that highest quality standards were used. Applicable codes were used where they were sufficient and more stringent requirements were placed on the design, where available codes were not sufficient. The Construction Phase Quality Assurance Program is presented in Appendix D(1). The description of the various systems and components includes the codes and standards that are met in the design and their adequacy.

References: Sections I-5, I-10, III-2 through III-8, IV-1 through IV-8, VII-2 through VII-9, VII-12, VII-17, VIII-4 through VIII-6, VIII-8, X-3, X-5, X-6, X-8, X-10, Chapter V, VI, XII and Appendices A and D(1).

2.1.2 Criterion 2 -- Performance Standards

"Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of 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 the 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 to the structural loading criteria presented in Appendix C insures that those structures, systems and components (SSCs) affected by this criterion are designed and built to withstand the forces that might be imposed by the occurrence of the various natural phenomena mentioned in the criterion, and this presents no risk to the health and safety of the public. The phenomena considered and margins of safety are also given.

References: Sections I-5, II-3, II-4, II-5, XII-2 and Appendix C.

2.1.3 Criterion 3 -- Fire Protection

"The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features."

As described in Section X-9, the materials and layout used in the station design have been chosen to minimize the possibility and to mitigate the effects of fire. Sufficient fire protection equipment is provided in the unlikely event of a fire, and in no case will the ability of the station to be shutdown be compromised by fire.

In accordance with 10CFR50.48(a), CNS is required to have a fire protection plan that is in compliance with 1971 GDC 3. Conformance with Draft GDC 3 is therefore superseded in its entirety by 1971 GDC 3.

References: Section X-9, X-18, and Chapter XII.

2.1.4 Criterion 5 -- Records Requirement

"Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor."

Complete records of the as-built design of the station, changes during operation and quality assurance records will be maintained throughout the life of the station, or as required by ANSI N45.2.9-1974.

References: Section XIII-8, and Appendices D and D(1).

2.2 Group II -- Protection by Multiple Fission Product Barriers (Criteria 6-10)

Conformance to these criteria assures, through proper design, that the station has been provided with multiple barriers against the release of, or means for, the mitigation of the consequences of the release of fission products to the environs and that these barriers remain intact during abnormal operational transients. These criteria also provide for proper containment and barrier against the release of fission products in the event of design basis accidents.

To provide the required protection, the reactor design provides six means of containing, preventing, or mitigating the release of fission products. These are: the fuel barrier consisting of highly compacted UO_2 fuel

sealed in high integrity Zircaloy cladding, the reactor coolant pressure boundary, the primary containment, the reactor building (secondary containment), the reactor building standby gas treatment system, and the plant stack (ERP).

2.2.1 Criterion 6 -- Reactor Core Design

"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 all offsite power."

The basis of the reactor core design, in combination with the station equipment characteristics and nuclear safety systems, is to provide sufficient margins to ensure that fuel damage does not occur during normal operation or as a result of abnormal operational transients. The core design is described in Chapter III and analysis of abnormal operational transients is given in Chapter XIV. The residual heat removal system and the reactor core isolation cooling system which remove decay heat during normal shutdowns and when the core is isolated from the condenser, are discussed in Chapter IV.

References: Sections I-5, III-2, III-6, III-7, IV-3, IV-7, IV-8, VII-2, XIV-2, XIV-4, XIV-5, and Appendix G.

2.2.2 Criterion 7 -- Suppression of Power Oscillations

"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."

The core design alone and the design of the nuclear system including the core have been analyzed to determine if power oscillations could occur. This analysis, which is presented in Section III-10, demonstrates that the nuclear system can be operated safely without danger of compromising any radioactive material barriers or fuel safety limits because of instability.

In 1998, CNS adopted a long-term stability solution (described in Section III-10) meeting the requirements of GDC-12[7].

References: Sections I-5, III-4, III-6, III-7, III-10, IV-4, VII-2, VII-5, and VII-7.

2.2.3 Criterion 8 -- Overall Power Coefficient

"The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive."

As indicated in Chapter III, the core is designed to be self-limiting; i.e., an arbitrary increase in core power over the power operating range results in a negative feedback. Thus, the overall power coefficient is negative.

References: Sections I-5, III-6, III-7, and III-10.

2.2.4 Criterion 9 -- Reactor Coolant Pressure Boundary

"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."

The reactor coolant pressure boundary consists of the vessels, pipes, pumps, tubes and similar process components that contain steam, water, gases, and radioactive materials coming from, going to, or in communication with the reactor core, as defined in 10CFR50.2. These are described primarily in Chapter IV. The reactor coolant system is designed to carry its dead weight and specified live loads separately or concurrently; these include pressure and temperature stresses, vibrations, and seismic loads prescribed for the station. Provisions are made to control or shutdown the reactor coolant system in the event of malfunction of operating equipment or leakage of coolant from the system. The reactor vessel and support structures 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 of any vessel nozzle to the containment atmosphere with the reactor vessel at design pressure concurrent with the Safe Shutdown Earthquake loads.

References: Sections I-5, IV-2, IV-3, IV-4, IV-10, VII-8, XII-2, XIV-5, XIV-6, Appendix A and Appendix C.

2.2.5 Criterion 10 -- Containment

"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 break, 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."

Two containment systems are provided; the drywell suppression chamber primary containment and the reactor building (secondary containment). These are described in Chapter V.

The primary containment system is designed, fabricated, and erected to accommodate without failure the pressures and temperatures resulting from or subsequent to the 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 for primary containment when the primary containment is open, as required. The two containment systems and such other associated engineered safety features 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. (or 10CFR50.67 for a Loss of Coolant or Fuel Handling Accident.

References: Sections V-2, V-3, XIV-4, XIV-6, and Appendix G.

2.3 Group III -- Nuclear and Radiation Controls (Criteria 11-18)

These criteria identify and define the station instrumentation and control systems necessary for maintaining the station 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 nuclear safety systems and engineered safety features.

To satisfy the intent of these criteria the station is provided with a comprehensive control and instrumentation system, most of which is described

in Chapter VII. Control of the station is from a central control room. Shielding and radiation protection are discussed in Section XII-3.

2.3.1 Criterion 11 -- Control Room

"The facility shall be provided with a control room from which actions 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 shutdown and maintain safe control of 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 to the control room is lost due to fire or other cause."

CNS is committed to the radiation protection provisions of 1971 GDC 19, which supersedes the Draft GDC 11 radiation exposure criteria[3]. The station is provided with a centralized control room having adequate shielding to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. For Loss of Coolant or Fuel Handling Accident conditions, the radiation exposure is maintained less than 5 rem TEDE for the duration of the accident. This allows the station to be shut down when necessary and allows safe control of the station to be maintained following shutdown. The station design does not contemplate the necessity for evacuation of the control room. However, if it is necessary to evacuate the control room, the station can be brought to a safe, cold shutdown from outside the control room.

References: Sections I-5, VII-2 through VII-5, VII-7 through VII-10, VII-12, VII-16 through VII-18, X-10, XII-2, XII-3, and Chapter XIV.

2.3.2 Criterion 12 -- Instrumentation and Control Systems

"Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges."

The necessary station controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These instruments and systems allow complete monitoring control of the facility throughout the normal operating range and through startup and shutdown. Sufficient instrumentation is provided to allow monitoring of variables necessary for effective station control.

References: Sections I-5, III-5, IV-10, VII-2 through VII-5, VII-7 through VII-14, VII-16, and VII-17.

2.3.3 Criterion 13 -- Fission Process Monitors and Control

"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 variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons."

Continuous monitoring of the performance of the reactor and the reactor power level are provided by the nuclear instrumentation system as described in Section VII-5. Control of core reactivity is through the use of control rods, the positions of which are continuously available on the control board, and through recirculation flow control.

References: Sections I-5, III-4, III-5, III-8, III-10, VII-2, VII-5, and VII-7 through VII-9.

2.3.4 Criterion 14 -- Core Protection Systems

"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."

The reactor protection system, described in Section VII-2 in association with other safety systems, automatically senses and limits conditions which could lead to unacceptable fuel damage. This system acts independently of, and overrides, other controls over control rod movement to initiate the necessary protective action. Evaluation of the protective action is given in the safety analysis.

References: Sections I-5, III-4, III-5, IV-4 through IV-8, VI-1 through VI-7, VII-2 through VII-5, VII-7, VII-12, XIV-1 through XIV-6, and Appendix G.

2.3.5 Criterion 15 -- Engineered Safety Features Protection Systems

"Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features."

The reactor emergency core cooling system control and instrumentation system description in Section VII-4 details the instrumentation provided to monitor the necessary variables and to automatically initiate the proper safety action in the event of an accident. This system also acts independently of the station process control systems and overrides other controls necessary to initiate the safety actions.

CNS is committed to conformance with 1971 GDC 20 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Sections I-5, VI, VII-2 through VII-5, VII-12, and VII-17.

2.3.6 Criterion 16 -- Monitoring Reactor Coolant Pressure Boundary

"Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage."

The methods of detecting leakage through the reactor coolant pressure boundary, and the limits imposed on this leakage, are discussed in Section IV-10.

References: Sections IV-10, V-2, VII-8, and X-14.

2.3.7 Criterion 17 -- Monitoring Radioactivity Releases

"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."

The station process and area radiation monitoring system and station sampling procedures are provided for monitoring significant parameters from specific station process systems and specific areas including the station

effluents to the site environs and to provide alarms and signals for appropriate corrective actions. These are described in Sections VII-12 and VII-13.

References: Sections I-4, VII-12, VII-13, and VII-15.

2.3.8 Criterion 18 -- Monitoring Fuel and Waste Storage

"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."

The new and spent fuel storage areas have been analyzed to determine their safety, and instrumentation is provided for monitoring where needed. Control and monitoring of waste storage is provided as described in Chapter IX, and Sections VII-12 and X-5.

CNS is committed to monitor fuel storage and radioactive waste systems in accordance with 1971 GDC 63[4]. Conformance with Draft GDC 18 is superseded in its entirety.

References: Sections I-5, VII-12, VII-13, X-2, X-3, and X-5.

2.4 Group IV -- Reliability and Testability of Protection Systems (Criteria 19-26)

The purpose of these criteria is to ensure that the systems used to prevent breach of the clad barrier will: (1) function when needed in spite of the failure of a component within the system, (2) be designated such that a condition requiring a protection system will not prevent the proper functioning of that system, and (3) be designed so that each channel of a protection system is independent of other channels within that system and the control systems. Protection system testability and detection of failures within the protection systems are necessary to ensure the reliability of these systems. As seen in the design bases and descriptions of these systems, sufficient attention has been paid to component reliability, system testability and alarms, independence and power supply, to ensure that the protection systems are adequate with respect to these criteria. The description of these systems appears largely in Chapter VII.

2.4.1 Criterion 19 -- Protection Systems Reliability

"Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed."

The components of the protection systems are designed to a high standard of reliability. Each system is designed with provisions for testing which approximate, to the degree practicable, the functioning of the system as credited in the accident analysis.

CNS is committed to conformance with 1971 GDC 21 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Sections I-5, III-4, VI, VII-2 through VII-5, VII-12, and Chapter XIV.

2.4.2 Criterion 20 -- Protection Systems Redundancy and Independence

"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 independence of redundant instrumentation components."

Protection system design includes the capability of providing the required protection, even with a component or channel inoperative due to failure or removal. Each of the protection function actions are initiated by a variety of sensed station conditions and by at least two instrument channels. This protection action is not dependent on a single channel.

CNS is committed to conformance with 1971 GDC 21 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Sections I-5, VII-2 through VII-5, VII-12, and Chapter XIV.

2.4.3 Criterion 21 -- Single Failure Definition

"Multiple failures resulting from a single event shall be treated as a single failure."

This definition is used in the design throughout the USAR for safety systems.

References: Section I-2, and XIV-4.

2.4.4 Criterion 22 -- Separation of Protection and Control Instrumentation Systems

"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 all requirements for the protection channels."

The systems which initiate the scram, isolation, and emergency core cooling actions are designed to automatically override normal operation controls whenever station conditions monitored by these systems exceed pre-established limits. Removal from service of a control instrumentation system cannot compromise any reactor protection function. Thus, protection action is independent of the state in normal operational process control actions.

CNS is committed to conformance with 1971 GDC 24 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Section I-4, III-4, VII-2 through VII-5, VII-12, and Chapter VI.

2.4.5 Criterion 23 -- Protection Against Multiple Disability for Protection Systems

"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."

These systems are designed to provide the required protection as long as necessary and in the presence of the most severe conditions which would be encountered. This includes conditions resulting from transients and accidents for which the protective action is required. Protection System electrical equipment that is located in a harsh environment has been environmentally qualified as required by 10CFR50.49.

CNS is committed to conformance with 1971 GDC 22 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Section I-5, VII-1 through VII-5, VII-12, and Chapter XIV.

2.4.6 Criterion 24 -- Emergency Power for Protection Systems

"In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems."

In the event of a loss of offsite power, the station auxiliary power system, the standby diesel generators, and the 125 volt battery system provide adequate power and redundancy to permit the required functioning of the protection systems. In addition, the 100% capacity redundant halves of each system are adequately separated to prevent the loss of power to the protection system resulting from any single active or passive failure.

NPPD is committed to 1971 GDC 17 which supersedes Draft GDC 24 in its entirety.

References: Chapters VI and VII, Sections I-5, and VIII-4 through VIII-6.

2.4.7 Criterion 25 -- Demonstration of Functional Operability of Protection Systems

"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."

The protection systems contain sufficient test signals, bypasses and indicators to allow testing of the system under simulated conditions that approximate, to the degree practicable, the actual condition for which the protective action is required. There are sufficient protection system channels to allow testing of a single channel without loss of the protective action.

CNS is committed to conformance with 1971 GDC 21 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Sections I-5, VI-7, VII-2 through VII-5, and VII-12.

2.4.8 Criterion 26 -- Protection Systems Fail-Safe Design

"The protection systems shall be designed to fail into a 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."

Systems essential to the protection functions are designed to fail-safe in their most probable failure modes. Thus, a systematic or environmentally caused failure will be detectable and will not compromise the protective function of the system.

CNS is committed to conformance with 1971 GDC 23 for the safety-related actuation instrumentation of the Reactor Building Ventilation Radiation Monitoring System[5].

References: Sections I-5, VI-1 through VI-6, VII-1 through VII-5, VIII-4 and VIII-5.

2.5 Group V -- Reactivity Control (Criteria 27-32)

Conformance to these six criteria provides assurance that the reactor core can be made and held subcritical from normal operation or from normal anticipated operational transients, by at least two reactivity control systems and that malfunction of a reactivity control system will not result in unacceptable damage to the fuel, rupture of the reactor coolant pressure boundary, or disrupt the core to the point of preventing emergency core cooling if needed. Two systems, an operational control system, consisting of moveable control rods, and control by recirculation flow control; and a standby liquid control system are provided to meet the intent of these criteria. The moveable control rod system design is given in Section III-5 and control of the moveable rod system is described in Section VII-7; the nuclear design, including the control rod reactivity worths, is given in Section III-6; reactor coolant recirculation system flow control is described in Section VII-9; and the standby liquid control system is described in Section III-9.

2.5.1 Criterion 27 -- Redundancy of Reactivity Control

"At least two independent reactivity control systems, preferably of different principles, shall be provided."

The two reactivity control systems provided are completely independent and of different principal. The operational control system accommodates fuel burnup, load changes and long-term reactivity changes. The standby liquid control system provides independent shutdown capability if it is needed.

References: Section I-5, III-4, III-5, III-9, VII-7, and VII-9.

2.5.2 Criterion 28 -- Reactivity Hot Shutdown Capability

"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 conditions, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits."

Both the control rod drive system and the standby liquid control system are capable of making and holding the core subcritical from any hot standby or hot operating condition up through full power. Consistent with current practice, this criterion is not interpreted to require a fast scram capability of both systems but only the stated shutdown capability.

References: Section I-5, III-4 through III-6, III-9, VII-7, and Chapter XIV.

2.5.3 Criterion 29 -- Reactivity Shutdown Capability

"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 exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided."

Reactor shutdown by the control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for normal operation and abnormal operational transients for which control rod motion is available, even with the most reactive control rod fully withdrawn. The nuclear design assures that sufficient reactivity compensation is always available to make the reactor subcritical from its most reactive condition including compensation for positive and negative reactivity changes resulting from nuclear coefficients, fuel depletion and fission product transients and buildup.

References: Sections I-5, III-4, III-6, VII-2, and Chapter XIV.

2.5.4 Criterion 30 -- Reactivity Holddown Capability

"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."

As indicated in the previous criterion response, the control rod drive system is designed to make and hold the reactor subcritical from its most reactive condition under normal credible operating conditions.

References: Sections I-5, III-4, III-6.

2.5.5 Criterion 31 -- Reactivity Control Systems Malfunction

"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."

Reactivity control systems designs (in conjunction with the reactor protection systems and rod block monitor) ensure that acceptable fuel damage limits will not be exceeded for any credible reactivity transient resulting from a single equipment malfunction or a single operator error.

References: Sections I-5, III-4, III-6, III-7, VII-2, VII-7, and Chapter XIV.

2.5.6 Criterion 32 -- Maximum Reactivity Worth of Control Rods

"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."

The system design is such that control rod worths and the rate at which reactivity can be added are sufficiently limited to assure that the design basis reactivity accident is not capable of damaging the reactor coolant system or disrupting the reactor core, its support structures, or other vessel internals

sufficiently to impair emergency core cooling system (ECCS) effectiveness, if ECCS is needed.

References: Sections I-5, III-4, III-6, III-7, VII-7, VII-16 and Chapters VI and XIV.

2.6 Group VI -- Reactor Coolant Pressure Boundary (Criteria 33-36)

The intent of this group of proposed criteria is to establish the reactor coolant pressure boundary design requirements and to identify the means used to satisfy these design requirements. The reactor coolant system design, described in Chapter IV and Section III-3, together with the construction and operational phase quality assurance program (Appendices D and D(1)), show that these criteria have been properly considered. In-service inspection of components and parts inside this boundary is discussed in Appendix J.

2.6.1 Criterion 33 -- Reactor Coolant Pressure Boundary Capability

"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."

As shown in Chapter XIV, the consequences of the design basis control rod drop accident (which is bounding for reactivity insertion events) cannot result in damage (either by motion or rupture) to the reactor coolant pressure boundary. This is due to the inherent safety features of the reactor core design combined with the control rod velocity limiter.

References: Sections I-5, III-3 through III-6, IV-2, IV-5, IV-6, and XIV-4 through XIV-6.

2.6.2 Criterion 34 -- Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

"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 of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (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."

The ASME and USAS Codes are used as the established and acceptable criteria for design, fabrication, and operation of components of the reactor coolant pressure boundary. The reactor coolant pressure boundary is designed and fabricated to meet the following, as a minimum:

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--USAS B.31.1, Code for Pressure Power Piping. Piping and valve replacements meet the requirements of the original design and construction codes, later editions of these codes, or ASME Section III.

The brittle fracture failure mode of the reactor coolant pressure boundary components is prevented by control of the notch toughness properties of the ferritic steel. This control is exercised in the selection of materials and fabrication of equipment and components. In the design, appropriate consideration is given to the different notch toughness requirements of each of the various ferritic steel product forms, including weld and heat-affected zones. In this way, assurance is provided that brittle fracture is prevented under the potential service loading temperatures.

References: Sections III-3, IV-2, IV-3, VII-8, and Appendices A and D(1).

2.6.3 Criterion 35 -- Reactor Coolant Pressure Boundary Brittle Fracture Prevention

"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 nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range."

NPPD's selected approach to brittle fracture prevention is to use a temperature based rule with modifications drawn from fracture mechanics technology. The approach, which is generally accepted by materials specialists, establishes the requirements for brittle fracture prevention. These requirements are less stringent, when measured in terms of NDTT requirement, for thin section materials than the thick section materials assumed in the first draft of this criterion.

The toughness properties of ferritic material and the service temperature of the reactor coolant pressure boundary shall assure:

1. Fully ductile behavior (e.g., in the energy absorption region of 100 percent shear fracture) whenever the boundary can be pressurized beyond the ASME Section III Code allowed safety valve setting by operational transients and postulated accidents; and

2. A ductile to brittle fracture transition temperature at least 60°F below the service temperature whenever the boundary can be pressurized beyond 20 percent of its design pressure by operational transients, hydro tests, and postulated accidents.

The response of the reactor system pressure to postulated accidents is discussed in the General Electric Company reply to Comment 3.8.1 of Amendment 1 to Bell Station Unit 1, Docket No. 50-319. There are no operational transients which can pressurize the system boundary beyond the ASME Section III Code allowed safety valve setting (1250 psig), so requirements for fully ductile behavior in pressure boundary materials are not anticipated for these events. During an ATWS Special Event, the reactor coolant pressure boundary is greater than 120° F above the nil ductility transition temperature (NDTT) and stresses

remain below the Service Level C Limits as defined in Section III of the ASME Code. Therefore, fully ductile behavior is assured.

It is believed that Criterion 35 should be applicable only to those components or systems whose failure would result in a loss of coolant in excess of the normal make-up capability of the reactor coolant system. On this ground small lines such as instrument lines have been excluded; certain other lines, such as the main steam lines, have been exempted from temperature control during hydrostatic test conditions in which failure would not affect core cooling.

Reference: Sections I-11 and IV-2, Appendix H.

2.6.4 Criterion 36 -- Reactor Coolant Pressure Boundary Surveillance

"Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight 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."

Extensive quality control assurance programs were followed during the entire fabrication of the reactor coolant system. The reactor coolant system was given a final hydrostatic test at 1563 psig in accordance with code requirements prior to initial reactor startup. Inservice inspections are performed in accordance with ASME Section XI. The system is checked for leaks, and abnormal conditions are corrected before reactor startup. The minimum vessel temperature during hydrostatic tests is at least 60°F above the calculated NDTT prior to pressurizing the vessel. Vessel material surveillance samples are used to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, heat affected zone metal and standards specimens. Also, leakage from the reactor coolant system is monitored during reactor operation, in accordance with Technical Specifications. The material surveillance program conforms to BWRVIP-86-A, October 2002.

References: Sections IV-2, IV-3, IV-10, and Appendix J.

2.7 Group VII -- Engineered Safety Features (Criteria 37-65)

The intent of this group of proposed criteria is (1) to identify the engineered safety features and their key support systems, (2) to examine them for independency, functional redundancy, capability, testability, inspectability, and reliability, (3) to determine their suitability relative to their intended duty, and (4) justify that each engineered safety feature's capability-scope encompasses the anticipated and credible phenomena associated with the abnormal operational transients or design basis accidents being considered. While the first seven criteria are applicable to all of the engineering safety features, the remaining criteria fall into four groups: emergency core cooling systems (Criteria 44-48); containment (Criteria 49-57); containment pressure reducing systems (Criteria 58-61); and air cleanup systems (Criteria 62-65). Examination of the engineered safety features will show that their design conforms to the Group VII Criteria. The Control Room Emergency Filter System was not originally classified as an engineered safety feature and so was not designed to be in conformity with Draft GDCs 37-43. These criteria have not been retroactively applied to the system as the engineered safety feature definition has broadened

to include systems that mitigate the radiological dose consequences to control room operators during accidents.

2.7.1 General Requirements for Engineered Safety Features (Criteria 37-43)

2.7.1.1 Criterion 37 -- Engineered Safety Features Basis for Design

"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 coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends."

The normal station control systems maintain station variables within operating limits. These systems are thoroughly engineered and backed up by a significant amount of experience in system design and operation. Even if an improbable maloperation or equipment failure occurs (including a reactor coolant pressure boundary break up to and including the circumferential rupture of any piping in that barrier), the nuclear safety systems and engineered safety features limit the effects to levels well below those which are of public safety concern. These engineered safety feature include those systems which are essential to the containment, isolation, and emergency core cooling functions.

References: Sections I-5, III-3, III-4, IV-2, IV-4, IV-6, V-2, V-3, VI-1 through VI-7, VII-2 through VII-4, VIII-4, through VIII-6, and XIV-1 through XIV-7.

2.7.1.2 Criterion 38 -- Reliability and Testability of Engineered Safety Features

"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 engineered 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."

The design of each of the systems essential to the engineered safety features includes the use of highly reliable components and provides for ready testability of these systems. Extensive analytical and experimental programs have shown that these systems are capable of performing their designated tasks.

References: Sections I-5, III-4, III-5, IV-6, V-2, V-3, VI-6, VII-2, VII-4, VII-5, VII-12, and VIII-4 through VIII-6.

2.7.1.3 Criterion 39 -- Emergency Power for Engineered Safety Features

"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 onsite power system and the offsite power system

shall each, independently, provide this capacity assuming a failure of a single active component in each power system."

With the redundant, full capacity diesel generators and batteries and redundant sources of offsite power, adequate power sources to accomplish all required safety functions under postulated design basis accident conditions is assured. Furthermore, each power source can be periodically tested for availability.

NPPD is committed to 1971 GDC 17 which supersedes Draft GDC 39 in its entirety.

References: Sections VII-2, VII-3, VII-4, and VIII-2 through VIII-6.

2.7.1.4 Criterion 40 -- Missile Protection

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

The systems and equipment which are required to function after design basis accidents or abnormal operational transients are designed to withstand the most severe forces and environmental effects, including missiles from station equipment failures anticipated from the accidents, without impairment of their performance capability.

References: Sections V-2, XII-2, and Appendix C.

2.7.1.5 Criterion 41 -- Engineered Safety Features Performance Capability

"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."

Those systems that comprise the engineered safety features are designed with sufficient redundancy and independence to fulfill their integrated required safety functions even with a failure of a single active component.

References: Sections VI-1 through VI-5, VII-4, and XIV-6.

2.7.1.6 Criterion 42 -- Engineered Safety Features Components Capability

"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."

The components which are required to function following a design basis loss-of-coolant accident are designed to withstand the most severe forces and environmental effects resulting from the accident.

References: Sections III-4 V-2, V-3, VI-1 through VI-5, VII-2, VII-3, VII-4, VIII-2 through VIII-6, and XIV-6.

2.7.1.7 Criterion 43 -- Accident Aggravation Prevention

"Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided."

The systems comprising the engineered safety features are designed to act in a positive manner in reducing the consequences of a loss-of-coolant accident.

References: Sections III-4, V-2, V-4, VI-1 through VI-5, VII-3, VII-4, and VIII-2 through VIII-6.

2.7.2 Emergency Core Cooling Systems (Criteria 44-48)

2.7.2.1 Criterion 44 - Emergency Core Cooling Systems Capability

"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 for 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 component 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."

The Emergency Core Cooling Systems (ECCS) are designed to limit clad temperature to below 2200°F over the entire credible spectrum of postulated design basis reactor coolant system breaks. Such capability is available concurrently with the loss of all offsite a-c power. The ECCS themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident can prevent adequate core cooling.

References: Sections VI-1 through VI-5, VII-4, and XIV-6.

2.7.2.2 Criterion 45 -- Inspection of Emergency Core Cooling Systems

"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."

The ECCS design includes provisions to enable physical and visual inspection of the ECCS components. Components are inspected prior to installation. In-service inspection is discussed in Appendix J.

References: Sections III-3, IV-2, and VI-6.

2.7.2.3 Criterion 46 -- Testing of Emergency Core Cooling Systems Components

"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 required functional performance."

To assure that the ECCS functions properly, if needed, specific design provisions have been made that allow for testing the operability and functional performance of active system components.

References: Sections I-5, VI-6, and VII-4.

2.7.2.4 Criterion 47 -- Testing of Emergency Core Cooling Systems

"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."

Specific provisions such as full-flow test lines have been provided in the ECCS design to allow periodic testing of the delivery capability of these systems under conditions that can be correlated to required performance during accident conditions.

References: Sections VI-6, and VII-4.

2.7.2.5 Criterion 48 -- Testing of Operational Sequence of Emergency Core Cooling Systems

"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."

To assure that the ECCS functions properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system. Testing of the systems is done in parts rather than testing of the entire operational sequence. This is due to the unavailability of these systems during a complete operational test as described, as well as the difficulty of performing such a test during reactor operation. The design complications which would have been required in order to permit testing in that manner would have complicated an already complex system, which might have been detrimental to safety.

References: Sections I-5, VI-4, VI-6, VII-4, VIII-5, VIII-6, and X-8.

2.7.3 Containment (Criteria 49-57)

2.7.3.1 Criterion 49 -- Containment Design Basis

"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."

The primary containment structure, including access openings and penetrations, is designed to withstand the peak accident pressure and temperatures which could occur due to the postulated design basis loss-of-coolant accident. The containment design includes considerable allowance for energy addition from metal-water or other chemical reactions beyond conditions that could exist during the accident.

References: Sections I-5, IV-6, V-2, V-3, VI-1, VI-2, VI-5, VII-3, VII-4, XIV-2 through XIV-7, and Appendix C.

2.7.3.2 Criterion 50 -- NDT Requirement for Containment Material

"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."

The design of the containment and its material are described in Section V-2. The criterion as stated is considered to be overly specific, considering the general nature of the other criteria. In keeping with the intent of these criteria to serve as a general guide, this criterion is interpreted to mean that the containment will be designed in accordance with applicable engineering codes.

References: Sections V-2 and V-3.

2.7.3.3 Criterion 51 -- Reactor Coolant Pressure Boundary Outside Containment

"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."

Analyses have been made to ensure that a rupture of a pipe which is part of the reactor coolant pressure boundary will not jeopardize the health and safety of the public according to the requirements established by 10CFR100. When needed, isolation valves are provided. The largest of these pipes are the steam lines. The analysis of a circumferential rupture of the steam line is discussed in Chapter XIV. The definition of the reactor coolant pressure boundary is given in Section I-2.

References: Sections I-5, II-2, II-3, IV-6, V-2, VII-3, and XIV-6.

2.7.3.4 Criterion 52 -- Containment Heat Removal Systems

"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."

Provisions are made for the removal of heat from within the primary containment as necessary to maintain the integrity of the containment for as long as necessary following the various postulated design basis accidents. Pressure suppression phenomena and the manually-actuated 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.

References: Sections I-5, IV-8, V-2, VI-1 through VI-5, VII-4, X-8, XIV-6 and XIV-7.

2.7.3.5 Criterion 53 -- Containment Isolation Valves

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

Lines which penetrate the primary containment and which communicate either with the reactor vessel or the primary containment free space are provided with at least two isolation valves (or equivalent) in series.

References: Sections I-5, IV-6, V-2, and VII-3.

2.7.3.6 Criterion 54 -- Containment Leakage Rate Testing

"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 the leakage rate measured over a sufficient period of time to verify its conformance with required performance."

After completion and installation of all penetrations, an integrated leakage rate test is performed at design pressure to verify that the containment design does meet the required maximum leakage rate. The test is performed over a 24 hour interval or longer as required, to show conformance to the required performance.

Reference: Section V-2.

2.7.3.7 Criterion 55 -- Containment Periodic Leakage Rate Testing

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

Primary Containment leakage rate testing is performed in accordance with 10CFR50 Appendix J at peak accident pressure. Leakage rate testing of Primary Containment penetrations is also performed at peak accident pressure, except in cases where the NRC has granted exemptions to 10CFR50 Appendix J and allowed reduced pressures.

Reference: Section V-2.

2.7.3.8 Criterion 56 -- Provisions for Testing of Penetrations

"Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time."

Provisions are made to demonstrate leak tightness of resilient seals and expansion bellows on containment penetrations on a individual basis. The test pressure and testing periodicity are in accordance with 10CFR50 Appendix J, except where relief has been granted or is otherwise specified in the Technical Specifications.

Reference: Sections V-2 and V-3.

2.7.3.9 Criterion 57 -- Provisions for Testing of Isolation Valves

"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."

Provisions are also made for demonstrating the functional performance of containment system isolation valves and determining valve leakage in accordance with the requirements of 10CFR50 Appendix J.

References: Sections IV-6, IV-10, V-2, VII-3, and VII-12.

2.7.4 Containment Pressure-Reducing Systems (Criteria 58-61)

2.7.4.1 Criterion 58 -- Inspection of Containment Pressure-Reducing Systems

"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."

The containment spray cooling system, an integral part of the residual heat removal system, is designed to allow periodic inspection of the pumps, pump motors, valves, heat exchangers, and piping of this system. The torus and torus water and the spray nozzles may also be periodically inspected.

References: Sections IV-8, V-2, V-3, VI-4, VI-6, X-6, X-8, and XII-2.

2.7.4.2 Criterion 59 -- Testing of Containment Pressure-Reducing Systems Components

"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."

The valves and pumps of these systems can be tested periodically for operability and capability to perform as required.

References: Sections IV-8, V-2, VI-4, VI-6, VII-3, VII-4, X-6, and X-8.

2.7.4.3 Criterion 60 -- Testing of Containment Spray Systems

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

The capability to test the functional performance of the containment spray cooling system is provided by inclusion in the design of appropriate test connections.

References: Sections IV-8, VI-4, VI-6, and VII-7.

2.7.4.4 Criterion 61 -- Testing of Operational Sequence of Containment Pressure-Reducing Systems

"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."

Those concerns expressed in the similar criterion for ECCS (Criterion 48) are reiterated here. These concerns are the detrimental effects on safety of the unavailability of the systems during test and the system complications required to carry out the test. Because of these concerns, this criterion is also interpreted to mean testing of such systems in parts rather than testing of the entire operational sequence. Such testing is provided in the design of the containment pressure-reducing systems, including the transfer to alternate power sources.

References: Sections V-2, V-3, VI-4, VI-6, VII-4, VIII-4, VIII-5, and VIII-7.

2.7.5 Air Cleanup Systems (Criteria 62-65)

2.7.5.1 Criterion 62 -- Inspection of Air Cleanup Systems

"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."

The standby gas treatment system (see Section V-3) which is located in the reactor building, may be physically inspected. This includes ducting, fans, filters, valves and heaters.

References: Sections V-2, V-3, and X-10.

2.7.5.2 Criterion 63 -- Testing of Air Cleanup Systems Components

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

Fans and dampers for the standby gas treatment system can be tested periodically for operability and required functional performance.

References: Sections V-2, V-3, and X-10.

2.7.5.3 Criterion 64 -- Testing of Air Cleanup Systems

"A capability shall be provided for in situ 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."

The standby gas treatment system includes provisions for periodic testing and surveillance to verify that degradation of the system has not occurred.

References: Sections V-2, V-3, and X-10.

2.7.5.4 Criterion 65 -- Testing of Operational Sequence of Air Cleanup Systems

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

The standby gas treatment system can be periodically tested for system performance under full flow conditions. Charcoal filter and HEPA filter performance is verified through the Ventilation Filter Testing Program.

References: Sections V-3, VII-12, and XIII-4; T.S. 5.5.7.

2.8 Group VIII - Fuel and Waste Storage Systems (Criteria 66-69)

The intent of this group of criteria is to ensure that fuel and waste storage systems are designed to minimize the probability of radioactivity release to station operating areas or public environs. A review of the new and spent fuel storage systems (Sections X-2 and X-3) and the radwaste systems (Chapter IX) shows that the intent of these criteria has been met. These GDCs do not apply to the CNS ISFSI, which is governed by 10 CFR 72.

2.8.1 Criterion 66 -- Prevention of Fuel Storage Criticality

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

Appropriate station fuel handling and storage facilities are provided to preclude accidental criticality for spent fuel. New fuel shipments are initially placed on the refueling floor in a controlled manner which prevents a critical array. They are eventually loaded into the Spent Fuel Pool. The CNS Technical Specifications preclude using the new fuel storage vault for new fuel storage out of concerns for inadvertent criticality if the application of foam fire retardants were to occur. The high density spent fuel racks are designed to prevent criticality.

CNS complies with the requirements of paragraph (b) of 10CFR50.68, "Criticality Accident Requirements."

References: Sections VII-6, X-2 and X-3.

2.8.2 Criterion 67 -- Fuel and Waste Storage Decay Heat

"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."

The fuel pool cooling system is designed to remove decay heat to maintain the Spent Fuel Pool (SFP) water temperature. The SFP contains sufficient water so that in the event of the failure of the fuel pool cooling system (such as might occur from an SSE), sufficient time is available prior to SFP boiling to either restore system cooling capability or provide alternate means of cooling the Spent Fuel Pool Pool.

References: Section X-5.

2.8.3 Criterion 68 -- Fuel and Waste Storage Radiation Shielding

"Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR20."

The handling and storage of spent fuel is done in the spent fuel storage pool. Water depth in the pool is maintained at a level to provide sufficient shielding for normal reactor building occupancy (10CFR20) by operating personnel. The spent fuel pool cooling and demineralizer system is designed to control water clarity (to allow safe fuel movement) and to reduce water radioactivity. Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within the limits of 10CFR20.

References: Sections IX-1 through IX-4, X-3, X-5, XII-2 and XII-3.

2.8.4 Criterion 69 -- Protection Against Radioactivity Release From Spent Fuel and Waste Storage

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

The consequences of a fuel handling accident are presented in Section XIV-6. In this analysis, it is demonstrated that undue amounts of radioactivity are not released to the public.

The spent fuel and waste storage systems are conservatively designed with ample margin, to prevent the possibility of gross mechanical failure which could release significant amounts of radioactivity. Backup systems such as floor and trench drains are provided to collect potential leakages. The fuel handling and waste disposal systems are described in Chapters X and IX, respectively. Operators are rigorously trained and administrative procedures are strictly followed to reduce the potential for human error.

The radiation monitoring system as described in Sections VII-12 and VII-13 is designed to provide station personnel with early indication of possible station malfunctions.

References: Sections V-1, V-2, V-3, IX-2 through IX-4, X-2, X-3, X-5, X-14, XII-1, XII-2, and XIV-6.

2.9 Group IX -- Plant Effluents (Criterion 70)

The intent of this criterion is to establish the station effluent release limits as defined by applicable regulations and to ensure that the station design provides means of controlling the releases within these limits. The various systems provided for radioactive effluent control are designed to meet the intent of this criterion.

2.9.1 Criterion 70 -- Control of Releases of Radioactivity to the Environment

"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 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR100 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."

The station radioactive waste control systems (which include the liquid, gaseous and solid radwaste systems) are designed to limit the off-site radiation exposure to levels below limits set forth in 10CFR20. The station engineered safeguards (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 10CFR100. The air ejector off-gas system is designed with sufficient holdup retention capacity so that during planned station operation the controlled release of radioactive materials does not exceed the established release limits at the ERP.

NPPD is committed to 1971 GDC 60 for the liquid, gaseous, and solid radwaste systems[6]. 1971 GDC 60 supersedes Draft GDC 70 in its entirety. NPPD is also committed to the 10CFR50.67 dose requirements for the Loss of Coolant and Fuel Handling Accidents.

References: Sections I-5, V-2, V-3, VII-12, VII-13, IX-2 through IX-4, XIV-2 through XIV-7, and the CNS Offsite Dose Assessment Manual (ODAM).

3.0 REFERENCES FOR APPENDIX F

1. "Safety Evaluation By the Division of Reactor Licensing U. S. Atomic Energy Commission In the Matter of Consumers Public Power District Cooper Nuclear Station Nemaha County, Nebraska Docket No. 50-298", dated April 4, 1968.

2. "Safety Evaluation Report For Cooper Nuclear Station Docket 50-298", dated February 14, 1973.

3. Letter from G. R. Horn (NPPD) to NRC Document Control Desk, dated July 26, 1994 (NLS940006) regarding Proposed Change No. 135 to Technical Specifications.

4. Deleted.

5. FSAR Amendment 13, Q/A 7.1.

6. Letter from G. R. Horn (NPPD) to NRC Document Control Desk, dated March 27, 1997 (NLS970002) requesting conversion of CNS Technical Specifications to Improved Technical Specifications.

7. GENE-A13-00395-01, 'Application of the "Regional Exclusion with Flow-Biased APRM Neutron Flux Scram" Stability Solution (Option I-D) to the Cooper Nuclear Station,' November 1996.