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I - INTRODUCTION AND SUMMARY

1.0 PROJECT IDENTIFICATION

The original Final Safety Analysis Report (FSAR) was submitted in 1971 in support of the application of the Nebraska Public Power District (formerly known as Consumers Public Power District), for a facility operating license for the Cooper Nuclear Station (CNS), located approximately 2½ miles south of the town of Brownville, Nemaha County, Nebraska, for power levels up to 2381 MWt under Section 104 (b) of the Atomic Energy Act of 1954, as amended, and the regulations of the Atomic Energy Commission (redesignated as the Nuclear Regulatory Commission after the Energy Reorganization Act of 1974) set forth in Part 50 of Title 10 of the Code of Federal Regulations (10CFR50).

Following Refueling Outage 24, a Measurement Uncertainty Recapture (MUR) power uprate license amendment was approved in accordance with 10CFR50, Appendix K. This allowed thermal power to be increased to 2419 MWt.[2]

This Updated Safety Analysis Report (USAR) meets the content requirements of 10CFR50.71(e). The USAR is updated and reported to the NRC in accordance with the requirements of 10CFR50.71(e).

Based on the design capability of the upgraded HP Turbine, LP Turbines, MUR, and the Main Generator electrical distribution and support systems, the station at rated power is designed to provide a gross electrical output of 835.5 MWe and a net electrical output of approximately 815 MWe. Historical dates of interest relating to CNS are as follows:

Publicly Announced	August, 1966
Contract Awarded (NSSS)	April, 1967
Construction Permit Application	July, 1967
Construction Permit Issued	June, 1968
Operating License Application	February, 1971
Operating License Issued	January 18, 1974
Fuel Loading	January 22, 1974
Initial Criticality	February 21, 1974
Commercial Operation	July 1, 1974
MUR Power Uprate Approved	June 30, 2008

The Nebraska Public Power District (NPPD) owns and operates the station. The station was designed by Burns and Roe, Inc. General Electric Company (GE) furnished the Nuclear Steam Supply System (NSSS) and Westinghouse Electric Corporation furnished the Turbine Generator set.

CNS uses a single cycle, forced circulation, boiling water reactor (GE BWR-4) substantially similar at the time of the license application to the TVA Browns Ferry Nuclear Power Station (AEC Docket 50 259/260). A heat balance showing the major parameters of Nuclear Steam Supply System for the design power condition is shown in Figure I-1-1. A plot plan for the CNS is shown on Burns and Roe Drawing 4003.

1.1 Identification and Qualification of Contractors

This section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The 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.

1.1.1 Applicant

As Owner, the applicant, Nebraska Public Power District, has engaged the below noted contractors to engineer and construct the Cooper Nuclear Station. However, irrespective of the contractual responsibilities discussed below, Nebraska Public Power District is the sole applicant for the facility license, and as owner and applicant is responsible for the design, construction, and operation of the Cooper Nuclear Station.

"Effective January 1, 1970, Consumers Public Power District changed its name to Nebraska Public Power District and properties of Platte Valley Public Power and Irrigation District and Nebraska Public Power System were merged into Nebraska Public Power District. This change of name was recognized by an Order Changing Name of Company in this Docket No. 50-298 dated January 14, 1970."

Nebraska Public Power District (NPPD) is a public corporation and political subdivision of the State of Nebraska engaging in generation, transmission, distribution, and sale of electric energy.

The newly consolidated utility, NPPD, serves retail and wholesale electric customers and irrigation customers in 85 of Nebraska's 93 counties.

The Platte Valley Public Power and Irrigation District and Nebraska Public Power System were previously engaged mainly in electric transmission and distribution and lightly in hydroelectric and fossil-fueled electric generation facilities. These former utilities had no previous background in nuclear-fueled electric generation.

Prior to change of its name, Consumers Public Power District (CPPD) was a public corporation and political subdivision of the State of Nebraska, organized in 1939 to engage in generation, transmission, distribution, and sale of electric energy. CPPD had been actively engaged in the application of nuclear energy to electric generation since 1957 when CPPD agreed to operate the Hallam Nuclear Power Facility for the Atomic Energy Commission. CPPD successfully operated that facility until the decision was made to dismantle the nuclear portion of the plant. CPPD subsequently performed the work required to retire the facility for the AEC. It is principally these former CPPD personnel with nuclear experience that are involved with the construction and will operate the Cooper Nuclear Station.

With its training and experience in the electric industry in general and in nuclear electric generation in particular, Nebraska Public Power District is well qualified to design, construct and operate the Cooper Nuclear Station. In addition, NPPD retained General Electric Company to provide project management services.

1.1.2 Engineer-Constructor

The Nebraska Public Power District has retained Burns and Roe, Inc., to provide engineering and construction management services for the design and construction of the station, integrating the items furnished by General Electric Company and Westinghouse Electric Corporation with complete balance of plant items. Burns and Roe, Inc., is also responsible for all procurement specifications. Burns and Roe, Inc., has been continuously engaged in construction or engineering activities since 1935.

Burns and Roe was founded in 1932. It was incorporated in 1935 as Burns and Roe, Inc. Burns and Roe, Inc., has been active in the fields of power generation and distribution, sea water and brackish water desalination, waste water renovation, environmental quality control, chemical and industry processing, and laboratory and testing facilities. Between 1955 and 1971, when the original FSAR was filed, Burns and Roe, Inc., completed or is currently providing engineering, design and/or construction management services for over 50 thermal power generating units, representing more than 11,400,000 kilowatts of new generating capacity of which more than 4,800,000 kilowatts is nuclear.

1.1.3 Nuclear Steam Supply System Supplier

General Electric Company has been awarded the contract to design, fabricate, and deliver the nuclear steam supply system and nuclear fuel for the Cooper Nuclear Station as well as to provide technical direction for installation and startup of this equipment. The General Electric Company has been engaged in the development, design, construction, and operation of boiling water reactors since 1955. Operating boiling water reactors designed and built by General Electric include Oyster Creek Unit 1, Vallecitos Boiling Water Reactor, Nine Mile Point Unit 1, Dresden Units 1 and 2, Humboldt Bay, Big Rock Point, KRB (Germany), Tarapur (India), KAIIL (Germany), JPDR (Japan), Tsuruga (Japan), Millstone Unit 1, Monticello Unit 1, and SENN (Italy). Among the domestic reactors of General Electric design now under construction are Brown's Ferry Units 1, 2, and 3, Dresden Unit 3, Quad Cities

Units 1 and 2, Vermont Yankee Unit 1, and Peach Bottom Units 2 and 3. Thus, General Electric has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of the reactor.

1.1.4 Turbine-Generator Supplier

The District has awarded a contract to Westinghouse Electric Corporation to design, fabricate, and deliver the turbine generator for Cooper Nuclear Station as well as to provide technical assistance for installation and startup of this equipment. More recently, Westinghouse Electric Corporation was awarded a contract by the District to erect this turbine generator unit. This now gives Westinghouse an overall responsibility for the unit. Westinghouse Electric Corporation has a long history in the application of turbine generators in nuclear power stations going back to the inception of commercial electrical power production utilizing nuclear facilities. Westinghouse furnished the turbine generator unit for Shippingport No. 1. This unit was shipped in 1956. Westinghouse also furnished the turbine generator unit for Yankee Atomic Power Company Rowe No. 1. This unit was shipped in 1959. Another unit currently in operation is at the Southern California Edison Company, San Onofre No. 1. The Connecticut Yankee, Haddam Neck No. 1 unit went critical July, 1967. Between 1967 and 1972 Westinghouse has firm orders to ship 28 turbine generator units in addition to the unit for Nebraska Public Power District for application to nuclear cycles. Inlet pressures of these units vary between 750 psig and 1000 psig and temperatures vary from saturation to approximately 40° superheat. The ratings of these units range from 500,000 kW to 1,090,000 kW. Westinghouse is therefore competent to design, fabricate, deliver, and erect the turbine-generator set and to provide technical assistance for the startup of this equipment .

2.0 DEFINITIONS

The following definitions apply to the terms used in the USAR. See CNS Technical Specifications Section 1.1 for additional definitions.

Abnormal Occurrence - Abnormal occurrence refers to the occurrence of any station condition that:

(1) Exceeds an Allowable Value as established in the Technical Specifications, or

(2) Violates a Limiting Condition for Operation as established in the Technical Specifications, or

(3) Causes any abnormal operational transient, or

(4) Causes any uncontrolled or unplanned release of radioactive material from the site.

Abnormal Operational Transient - An abnormal operational transient includes the events following a single equipment malfunction or a single operator error that is reasonably expected during the course of planned operations and is one of the design basis events. Power failures, pump trips, and rod withdrawal errors are typical of the single malfunctions or errors initiating the events in this category.

Accident - An accident is a single event, not reasonably expected during the course of station operations, that has been hypothesized for analysis purposes or postulated from unlikely but possible situations, and is one of the design basis events, and that causes or threatens a rupture of a Radioactive Material Barrier. A pipe rupture qualifies as an accident; a fuel cladding defect does not.

Achieving Criticality - See Planned Operation.

Achieving Shutdown - See Planned Operation.

Activated Device - An activated device is a mechanical module in a system used to accomplish an action. An activated device is controlled by an Actuation Device. See Figure VII-1-1.

Active Component - An active component is a device characterized by an expected significant change of state or discernible mechanical motion in response to an imposed design basis load demand upon the system. Examples include switches, relays, valves, pressure switches, turbines, transistors, motors, dampers, pumps, analog meters, etc.

Active Fuel - The portion of a fuel rod that contains fuel pellets (either enriched or unenriched).

Actuation Device - An actuation device is an electrical or electromechanical module in a control and instrumentation system controlled by an electrical decision output used to produce mechanical operation of one or more Actuated Devices to accomplish the necessary action. See Figure VII-1-1.

Alteration of the Reactor Core - See Core Alteration.

Alternate Shutdown Capability - Alternate shutdown capability is the control capability provided to safely shutdown the reactor if an event, such as a fire, disables the normal control circuits to the Control Room or causes evacuation of the Control Room.

Anticipated Transient Without Scram (ATWS) - Anticipated Transient Without Scram is an Abnormal Operational Transient followed by the failure of the reactor protection system. See 10CFR50.62.

Availability - Availability is the probability that a structure, System, or component (SSC) is capable of performing its specified function when called upon.

Cold Shutdown Condition - The reactor is in MODE 4 or MODE 5 as defined in the Technical Specifications.

Component - Components are items from which the system is assembled (e.g., resistors, capacitors, wires, connectors, transistors, switches, springs, pumps, valves, piping, heat exchangers, vessels, etc.).

Cooldown - See Planned Operation.

Core Alteration - See Technical Specifications.

Core Fuel to Water Total Power - The core fuel to water total power is the sum of:

(1) The instantaneous integral, over the entire fuel clad outer surface, of the product of heat transfer area increment and position dependent heat flux, and

(2) The instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

Core Operating Limits Report (COLR) – See Technical Specifications.

Deep Dose Equivalent (DDE) - Deep dose equivalent applies to external whole-body exposure. It is the dose equivalent at the tissue depth of 1 cm (1000 mg/cm^2). See 10CFR20.

Design Basis Accident - A design basis accident is a hypothesized accident the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of Radioactive Material Barriers. The potential radiation exposures resulting from a design basis accident are greater than any similar accident postulated from the same general accident assumptions and could result in potential offsite exposure comparable to the guideline exposure of 10CFR100, or 10CFR50.67 (Fuel Handling Accident or Loss of Coolant Accident).

Design Basis Events - Conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to function to ensure

(1) the integrity of the reactor coolant pressure boundary

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition

(3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposure of 10CFR100, or 10CFR50.67 (Fuel Handling Accident or Loss of Coolant Accident).

Design Power - Design power means a steady-state power level of 2486 thermal megawatts. This is 104.4% of Rated Power (105% of rated steam flow).

Engineered Safety Feature (ESF) - An ESF is a safety-related SSC that performs a safety action necessary to maintain the consequences of postulated accidents within acceptable limits. Also known as Engineered Safeguard.

Environmental Qualification (EQ) - Environmental qualification is the generation and maintenance of evidence to assure that electrical equipment important to safety and within a harsh environment will operate upon demand to meet system performance requirements. See 10CFR50.49.

Essential - Essential functions, structures, systems and components are equivalent to safety-related functions, structures, systems and components.

Fire Safety Analysis - A Fire Safety Analysis is the document required by NFPA 805, Section 2.7.1.2, that includes fire hazards identification and nuclear safety capability assessment, on a fire area basis, for all fire areas that could affect the nuclear safety or radioactive release performance criteria defined in Chapter 1 of that Standard. The CNS Fire Safety Analysis comprises the following calculations:

NEDC 11-084 - Fire Safety Analysis for Fire Area CB-A
NEDC 11-085 - Fire Safety Analysis for Fire Area CB-A-1
NEDC 11-086 - Fire Safety Analysis for Fire Area CB-B
NEDC 11-087 - Fire Safety Analysis for Fire Area CB-C
NEDC 11-088 - Fire Safety Analysis for Fire Area CB-D
NEDC 11-089 - Fire Safety Analysis for Fire Area IS-A
NEDC 11-090 - Fire Safety Analysis for Fire Area DG-A
NEDC 11-091 - Fire Safety Analysis for Fire Area DG-B
NEDC 11-092 - Fire Safety Analysis for Fire Area RB-A
NEDC 11-093 - Fire Safety Analysis for Fire Area RB-B
NEDC 11-094 - Fire Safety Analysis for Fire Area RB-CF
NEDC 11-095 - Fire Safety Analysis for Fire Area RB-DI
NEDC 11-096 - Fire Safety Analysis for Fire Area RB-E
NEDC 11-097 - Fire Safety Analysis for Fire Area RB-FN
NEDC 11-098 - Fire Safety Analysis for Fire Area RB-J
NEDC 11-099 - Fire Safety Analysis for Fire Area RB-K
NEDC 11-100 - Fire Safety Analysis for Fire Area RB-M
NEDC 11-101 - Fire Safety Analysis for Fire Area RB-N
NEDC 11-102 - Fire Safety Analysis for Fire Area RB-P
NEDC 11-103 - Fire Safety Analysis for Fire Area RB-T
NEDC 11-104 - Fire Safety Analysis for Fire Area RB-V
NEDC 11-105 - Fire Safety Analysis for Fire Area TB-A
NEDC 11-106 - Fire Safety Analysis for Fire Area TB-C
NEDC 11-107 - Fire Safety Analysis for Fire Area YD
NEDC 13-009 - Fire Safety Analysis for Fire Area DW
NEDC 14-043 - Fire Safety Analysis for Entire Power Block
NEDC 10-080 - Fundamental Fire Protection Program and Design Elements (B-1 Table)
NEDC 11-019 - Nuclear Safety Capability Assessment (NSCA)
NEDC 10-062 - NFPA 805 Radioactive Release Review

Fuel Damage - Fuel damage is perforation of the fuel cladding which would permit the release of fission products into the reactor coolant.

Fuel Zone Zero - Fuel Zone Zero corresponds to the highest Top of Active Fuel of all fuel types that may be used at CNS, up to a maximum active fuel length of 150". (See also Top of Active Fuel.)

Heatup - See Planned Operation.

Incident - An incident is any event, abnormal operational transient, or accident not considered as part of planned operation.

Incident Detection Circuitry - Incident Detection Circuitry includes those trip systems which are used to sense the occurrence of an incident. Such circuitry is described and evaluated separately where the incident detection circuitry is common to several systems.

Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit a signal related to the plant parameter monitored by that instrument channel. A channel terminates and loses its identity where individual channel outputs are combined in logic. See Figure VII-1-1.

Locked Open/Closed – Locked Open/Closed means to maintain a component in an established position by administrative controls using a mechanical lock to physically restrain the component to its proper position.

Logic - Logic is that array of components in a control and instrumentation channel which combines individual bistable output signals to produce decision outputs. See Figure VII-1-1.

Maximum Anticipated Thermal Output - The maximum anticipated thermal output is the thermal energy output at Design Power.

Module - Any assembly of interconnected components in a control and instrumentation circuit which constitutes an identifiable device, instrument, or piece of equipment.

Notch - A notch is a control rod blade position corresponding to the grooves cut in the CRDM index tube. There are six inches between notches.

Nuclear Safety Operational Analysis - A nuclear safety operational analysis is a systematic identification of the requirements for and the limitations on station operation necessary to satisfy nuclear safety operational criteria.

Nuclear Safety Operational Criteria - A nuclear safety operational criteria is a set of standards which were used to provide input for the proposed CNS Technical Specifications contained in Appendix B of the FSAR.

Nuclear Safety System - A nuclear safety system is a safety system the action of which is required for a safety action that is necessary in response to an abnormal operational transient. See Figure I-2-3.

Nuclear Steam Supply System (NSSS) - The NSSS generally includes those systems most closely associated with the reactor vessel which are designed to contain or be in communication with the water and steam coming from or going to the reactor core. The NSSS includes the following:

Reactor vessel
Vessel internals
Reactor core
Main Steam lines from reactor vessel to the isolation valves outside the Primary Containment
Neutron Monitoring system
Reactor Recirculation system
Control Rod Drive system
Residual Heat Removal system
Reactor Core Isolation Cooling system
Emergency Core Cooling systems
Reactor Water Cleanup system
Reactor Feedwater system piping between the reactor vessel and the first valve outside the Primary Containment
Safety and Relief Valve system

Operating - Operating means a system, subsystem, train, component, or device is performing its intended function in its required manner.

Operational - The adjective operational, along with its noun and verb forms, is used in reference to the working or functioning of the station, in contrast to the design of the station.

Passive Component - A device characterized by an expected negligible change of state or negligible mechanical motion in response to an imposed design basis load demand upon the system. Examples include cable, piping, valves in stationary positions, resistors, capacitors, fluid filters, indicator lamps, cabinets, cases, etc.

Planned Operation - Planned operation is the normal station operation under planned conditions in absence of significant abnormalities. Operations subsequent to an incident (transient, accident, or special event) are not considered planned operations until the actions taken in the station are identical to those which would be used had the incident not occurred. The established planned operations can be considered as a chronological sequence:

refueling outage - achieving criticality - heatup - power operation - achieving shutdown - cooldown - refueling outage.

The following planned operations are identified:

a. Refueling Outage – A planned operation, consisting of the period of time between the shutdown of the unit prior to a refueling and the startup of the plant after that refueling, which includes all the actions associated with a normal refueling outage:

(1) Planned, physical movement of core components such as the fuel and control rods.

(2) Refueling surveillance and testing operations.

(3) Planned maintenance.

For the purpose of designating the frequency of testing and surveillances, a refueling outage shall mean a regularly scheduled refueling outage.

b. Achieving Criticality – Achieving criticality is a planned operation which includes all the actions which are normally accomplished in bringing the station from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

c. Heatup – Heatup is the planned operation which begins where achieving criticality ends and includes all actions which are normally accomplished in approaching nuclear steam supply system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the turbine generator and bypass valve operation.

d. Power Operation – Power operation is the planned operation which begins where heatup ends and includes continued operation of the station at power levels in excess of heatup power.

e. Achieving Shutdown – Achieving shutdown is the planned operation which begins where power operation ends and includes all actions normally accomplished in achieving nuclear shutdown (reactivity equivalent to more than one rod subcritical) following power operation.

f. Cooldown – Cooldown is the planned operation which begins where achieving shutdown ends and includes all actions normally accomplished in the continued removal of decay heat and the reduction of nuclear steam supply system temperature and pressure.

Power Generation - The phrase power generation, when used to modify words such as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to the mission of the station, which is to generate electric power, as opposed to concerns considered to be of primary safety importance. Thus, the phrase power generation is used to identify aspects of the station which are not considered to be of primary importance to safety.

Power Generation Action - A power generation action is an action in the station which is required for the avoidance of specified conditions considered to be of primary significance to the station mission, the generation of electrical power. The specified conditions are those that are most directly related to the following:

(1) The ability to carry out the station mission, the generation of electrical power, through planned operation.

(2) The avoidance of conditions which would limit the ability of the station to generate electrical power.

(3) The avoidance of conditions which would prevent or hinder the return to conditions permitting the use of the station to generate electrical power following an abnormal transient, accident, or special event.

There are power generation actions associated with planned operation, abnormal operational transients, accidents, and special events. See Figure I-2-3.

Power Generation Design Basis - The power generation design basis for a power generation system states in functional terms the unique design requirements which establish the limits within which the power generation objective shall be met. A safety system may have a power generation design basis which states in functional terms the unique design requirements which establish the limits within which the power generation objective for the system shall be met.

Power Generation Evaluation - A power generation evaluation is an evaluation which shows how the system satisfies some or all of the Power Generation Design Bases. Because power generation evaluations are not directly pertinent to public safety, they are generally not included in the USAR. However, where a system or component has both safety and power generation objectives, a power generation evaluation can be used to clarify the safety versus power generation capabilities.

Power Generation Objective - A power generation objective describes in functional terms the purpose of a system or component as it relates to the mission of the station. This includes objectives which are specifically established so the station can fulfill its Power Generation Actions. A system or piece of equipment has a power generation objective if it is a Power Generation System. A safety system can have a power generation objective, in addition to a safety objective, if parts of the system are intended to function for power generation purposes. See Figure I-2-3.

Power Generation System - A power generation system is any system the action of which is not required for a safety action, but which is required for a power generation action. See Figure I-2-3.

Power Operation - See Planned Operation.

Primary Containment - The primary containment is a Radioactive Material Barrier consisting of the drywell in which the reactor vessel is located, the pressure suppression chamber, and process lines out to the first isolation valve outside the containment wall. Portions of the reactor coolant pressure boundary may become part of the primary containment, depending upon the location of a postulated failure. For example, a closed main steam line isolation valve is part of the primary containment barrier when the postulated failure of the main steam line is inside the primary containment.

Primary Containment Integrity - Primary Containment Integrity means that the Technical Specification Primary Containment Limiting Conditions for Operation are satisfied.

Process Control Program - The CNS Process Control Program (PCP) establishes the processing conditions for assuring the solidification, dewatering or stabilization of CNS radioactive waste streams produced from the CNS liquid radioactive waste treatment system and from activities producing radioactive waste requiring solidification, dewatering or stabilization such as decontamination system resins, irradiated components and highly contaminated equipment. The PCP ensures that processing of radioactive waste containing liquid, which is subject to the requirements of 10CFR61, is consistent with the requirements specified in the Cooper Nuclear Station Offsite Dose Assessment Manual (ODAM).

Process Safety System - A process safety system is a safety system which performs a safety action that is necessary during planned operation. See Figure I-2-2.

Protection System - Protection system is a generic term which may be applied to nuclear safety systems and engineered safety features. See Figure I-2-3.

Protective Action - A protective action is an ultimate action at the system level which contributes to and is required for the accomplishment of a safety action. System level actions which are required to accomplish reactor scram, reactor vessel isolation, containment isolation, pressure relief, automatic depressurization, and emergency core cooling are some of the protective actions. See Figures I-2-1, I-2-2, and I-2-3.

Protective Function - A protective function is a function which encompasses the monitoring of one or more station variables or conditions and the associated initiation of intra-system actions which eventually result in protective action. See Figure I-2-2.

Radioactive Material Barrier - A radioactive material barrier includes the systems, structures, or equipment that together physically prevent the uncontrolled release of radioactive materials. These barriers include the Reactor Fuel Barrier, Reactor Coolant Pressure Boundary, Primary Containment, and Secondary Containment.

Radioactive Material Barrier Damage - Radioactive material barrier damage is defined as an unplanned, undesirable breach in a Radioactive Material Barrier. Operation of a relief or safety valve does not constitute barrier damage.

Rated Thermal Power (RTP) - See Technical Specification 1.1 definition. This is also termed 100% power in the USAR and Maximum Power Level per Operating License Condition 2.C.(1). Rated steam flow, rated coolant flow, rated neutron flux, and rated nuclear steam supply system pressure refer to the values of these parameters when the reactor is at Rated Thermal Power.

Reactor Coolant Pressure Boundary - The portion of the Nuclear System consisting of the reactor vessel and attached piping out to and including the second isolation valve in each attached pipe. Also known as Nuclear System Process Barrier.

Reactor Fuel Barrier - The reactor fuel barrier is the Radioactive Material Barrier consisting of the uranium dioxide fuel sealed in metal cladding.

Refuel Mode - The refueling mode is defined in Technical Specification Table 1.1-1 as MODE 5.

Refueling Outage - See Planned Operation.

Reliability - Reliability is the probability that an item will perform its specified function without failure for a specified time period in a specified environment.

Risk - Risk is the product of the probability of an event and the adverse consequences of the event.

Rod Density - Rod density is the fraction or percent of control rods fully inserted into the core as determined by the total number of notches inserted into the core divided by the total number of notches inserted when all control rods are inserted.

Run Mode - Power operation is defined in Technical Specification Table 1.1-1 as MODE 1.

Safe Shutdown Systems and Components - Those components or portions of systems used to achieve and maintain a cold shutdown condition.

Safety - The word safety, when used to modify such words as objective, design basis, action, and system, indicates that the objective, design basis, action, or system is related to concerns considered to be of primary safety significance, as opposed to the plant mission of generating electrical power. Thus, the word safety is used to identify aspects of the station which are considered to be of primary importance with respect to safety.

Safety Action - A safety action is an ultimate action in the station which is required for the avoidance of specified conditions considered to be of primary safety significance. The specified conditions are those that are most directly related to the ultimate limits on the integrity of the radioactive material barriers or the unplanned or uncontrolled release of radioactive material. There are safety actions associated with planned operation, abnormal operational transients, accidents, and special events. Safety actions include such actions as the indication to the operator of the values of certain process variables, reactor scram, emergency core cooling, and reactor shutdown from outside of the control room. See Figures I-2-1 and I-2-2.

Safety Design Basis - The safety design basis for a safety system states in functional terms the unique design requirements which establish the limits within which the safety objective shall be met. A Power Generation System may have a safety design basis which states in functional terms the unique design requirements that ensure that neither planned operation nor operational failure by the system results in conditions for which station safety actions would be inadequate.

Safety Evaluation - A safety evaluation is an evaluation which shows how the system satisfies the safety design basis. In the USAR, safety evaluations are provided for those systems having a safety design basis. Safety evaluations may form the bases for the Technical Specifications and establish why specific limitations are imposed. For station modifications and design changes, safety evaluations are performed on safety and non-safety systems in accordance with CNS procedures.

Safety Limit - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Nuclear Regulatory Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

Safety Objective - A safety objective describes in functional terms the purpose of a system or component as it relates to conditions considered to be of primary significance to the protection of the public. This relationship is stated in terms of radioactive material barriers or radioactive material release. The only systems which have safety objectives are safety systems. See Figure I-2-3.

Safety-Related - Safety-related functions, structures, systems and components are those that are necessary to ensure:

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposure of 10CFR100, or 10CFR50.67 (Fuel Handling Accident or Loss of Coolant Accident).

Safety System - A safety system is any system, group of systems, component, or group of components that accomplish a safety action. See Figure I-2-3.

Scram - Scram refers to the automatic rapid insertion of control rods in response to the detection of undesirable conditions.

Sealed-Open/Closed – Sealed-open/closed refers to maintaining a component in an established position by administrative controls using a wire seal such that deliberate action is necessary to defeat the mechanism.

Secondary Containment - The secondary containment is the Radioactive Material Barrier consisting of the reactor building, which completely encloses the primary containment, the standby gas treatment system, and the Elevated Release Point (ERP).

Secondary Containment Integrity - Secondary Containment Integrity means that the Technical Specification Limiting Condition for Operation for Secondary Containment is satisfied.

Sensor - A sensor is that part of an instrument channel used to detect variations in the measured station variable or parameter. See Figure VII-1-1.

Setpoint - A setpoint is that value of the monitored plant variable or parameter which causes an instrument channel trip or a relief or safety valve to relieve pressure.

Shutdown - The reactor is shutdown when the effective neutron multiplication factor (K_{eff}) is sufficiently less than 1.0 that the full withdrawal of any one control rod could not produce criticality.

Shutdown Mode - The hot and cold shutdown modes are defined in Technical Specifications Table 1.1-1 as MODES 3 and 4 respectively.

Single Failure - A single failure is a failure that can be ascribed to a single causal event. Single failures of active components are considered in the design of certain systems and are presumed in the evaluations of incidents to investigate the ability of the station to respond in the required manner under degraded conditions. See 10CFR50, Appendix A.

Special Event - A special event is an event which neither qualifies as an abnormal operational transient nor an accident and is not a design basis event but which is postulated to demonstrate some special capability of the system or systems.

Special Safety System - A special safety system is a safety system that performs an action that is necessary in response to a special event. See Figure I-2-3.

Startup/Hot Standby Mode - The startup mode is defined in Technical Specification Table 1.1-1 as MODE 2.

Station Blackout (SBO) – The complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). See 10CFR50.63.

Test Duration - The test duration is the elapsed time between test initiation and test termination.

Top of Active Fuel (TAF) - The highest elevation to which fuel pellets extend in the core (either enriched or unenriched). See also Fuel Zone Zero. Note - TAF is equivalent to the term "Top of Active Irradiated Fuel" used in the Technical Specifications and BASES.

Total Effective Dose Equivalent (TEDE) - The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). See 10CFR20.

Trip - A trip is the change of state of a bistable device in a control and instrumentation circuit which represents the change from a normal condition. A trip signal, which results from a trip, is generated in the channels of a Trip System and produces subsequent trips and trip signals throughout the system as directed by the logic.

Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems. A trip system terminates and loses its identity where outputs are combined in logic. See Figure VII-1-1.

Ultimate Heat Sink – The ultimate heat sink is the heat dissipation means for the Unit to the environment, including the necessary retaining structure, and any connecting canals or conduits.

Unavailability - Unavailability is the probability that an SSC is incapable of performing its specified function when called upon. (The sum of availability and unavailability equals unity.)

Unsafe Failure - An unsafe failure is a failure that negates system operability and which, due to its nature, is revealed only when the instrument or control channel is functionally tested or attempts to respond to a real signal.

3.0 METHODS OF TECHNICAL PRESENTATION

3.1 Purpose

The purpose of the original Final Safety Analysis Report (FSAR) was to provide the technical information required by 10CFR50.34(b) to establish a basis for evaluation of the station with respect to the issuance of a facility operating license. The purpose of the USAR is to update descriptions and analyses to reflect current plant operations as required by 10CFR50.71(e).

3.2 Radioactive Material Barrier Concept

Because the safety aspects of this report pertain to the relationship between station behavior under a variety of circumstances and the radiological effects on persons off-site, the report is oriented to the radioactive material barriers. This orientation facilitates evaluation of the radiological effects of the station on the environment. Thus, the presentation of technical information is considerably different from that which would be expected in an operational manual, maintenance manual, or nuclear engineer's handbook.

The overriding consideration that determines the depth of detailed technical information presented about a system or component is the relationship of the system or component to the radioactive material barriers. Systems that must operate to preserve or limit the damage to the radioactive material barriers are described in the greatest detail. Systems that have little relationship to the radioactive material barriers are described only with as much detail as necessary to establish their functional role in the station.

3.3 Organization of Contents

The USAR is organized into 14 chapters each of which consists of a number of sections. The principal architectural and engineering criteria, which define the broad frame of reference within which the station is designed, are set forth in Section I-4.0. The categories used for classifying the CNS SSCs with respect to safety are given in Section I-5.0.

Chapters II through XIII present detailed information about the design and operation of the station. The nuclear safety systems and engineered safety features are integrated into these chapters according to system function (emergency core cooling, control), system type (electrical, mechanical), or according to their relationship to a particular radioactive material barrier. Chapter III describes station components and presents design details that are most pertinent to the fuel barrier. Chapter IV describes station components and systems that are most pertinent to the nuclear system process barrier. Chapter V describes the primary and secondary containments. Thus, Chapters III, IV, and V are arranged according to the four radioactive material barriers.

The remainder of the chapters group system information according to station function (radioactive waste control, emergency core cooling, power conversion, control) or system type (electrical, structures). Chapter XIV provides an overall safety evaluation of the station which demonstrates both the adequacy of equipment to protect the radioactive material barriers and the ability of the engineered safety features to mitigate the consequences of situations in which one or more radioactive material barriers are assumed damaged.

3.4 Format Organization of Chapters

The format and content provided is generally based on "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy

Commission in June, 1966. Each chapter is designated by a Roman numeral, e.g., I, II, etc. Each chapter is subdivided further and given an Arabic numeral, e.g., 1.0, 2.0, etc., and is individually paginated, i.e., page number I-4-5 is the fifth page of Chapter I, section 4.0. Sections are further subdivided by numbers following decimal points.

Tabulations of data appearing throughout the text are designated as "Tables" and are identified by Roman numerals corresponding to the chapter in which it appears and an Arabic number indicating its section and its sequence of tables appearing in that section, e.g., Table I-4-2, is the second table appearing in section 4.0 of Chapter I.

Applicable sketches, pictures, and plots are placed at the end of a chapter, and are identified as "Figures" by the chapter and section numbers and the sequential order of the drawing or diagram, e.g., Figure V-2-1. Elsewhere, reference to appropriate NPPD controlled drawings is made within the text, in order to illustrate or clarify the information presented. An equipment symbol chart for the station process and instrumentation drawings is shown on General Electric Drawing 197R567 for General Electric equipment scope and Burns and Roe Drawing 2001, Sheets 1 and 2, for Burns and Roe equipment scope.

The general organization of a chapter describing a system or component is as follows:

Objective
Design Basis
Description
Evaluation
Inspection and Testing

To clearly distinguish the safety versus power generation aspects of a system, the objective, design basis, and evaluation titles are modified by the word "safety" or "power generation," according to the definitions given in Section I-2.0. Systems that have safety objectives are safety systems. A safety evaluation is included only when the system has a safety design basis; the evaluation shows how the system satisfies the safety design basis. A power generation evaluation is included only when needed to clarify the safety versus power generation aspects of a system that has both safety and power generation functions.

A nuclear safety operational analysis of the station has been performed to systematically identify the operational limitations or restrictions which must be observed with regard to certain process variables and certain station systems to satisfy specified nuclear safety operational criteria. The method used for this analysis is described in Appendix G. The resulting operational limits or restrictions formed the bases for proposed technical specifications which were submitted as Appendix B to the original FSAR. The present Technical Specifications may have various bases.

Chapters presenting information on topics other than systems or components are arranged individually according to the subject matter so that the relationship between the subject and public safety is emphasized.

Within each section of the text, applicable supporting technical material is referenced. These general references are cited either in the text or in a list of references at the end of a chapter. Documents that have been incorporated into the USAR by reference are underscored in the main text, as applicable, and are listed in Table I-3-1.

Certain text has been designated as historical information by being italicized in a distinctive font. Historical information was accurate at the time of its initial submittal to the AEC/NRC (or as subsequently revised) and may have been relied on for licensing decisions made at the time. While historical information remains an important reference when evaluating proposed changes, tests, or experiments pursuant to 10CFR50.59, updating of this information is not generally required for compliance with 10CFR50.71(e). An exception to this is when contemporary information affects previous safety analyses conclusions relative to public health, and new safety analyses have been prepared with the results placed on the CNS docket as a result of NRC requirements. Historical information relates to: a) text that describes a completed physical milestone that is inherently dated, b) topics that are outside the responsibility of NPPD to control or influence as 10CFR50.59 changes, tests, or experiments, or c) information that has been retained to provide historical perspective, but that has been superseded with equivalent up-to-date information.

TABLE I-3-1

DOCUMENTS INCORPORATED INTO THE
CNS USAR BY REFERENCE

The following documents have been submitted on the CNS docket and are considered to be incorporated by reference into the CNS USAR in accordance with the requirements of 10CFR50.32. The revision numbers for these documents were effective a maximum of 6 months prior to the most recent USAR Update filing. For documents subject to a programmatic change process, revisions and submittal of changes to the incorporated document are made in accordance with the respective regulatory requirement governing the document.

Documents Subject to Update

1. NPPD Cooper Nuclear Station Quality Assurance Program For Operation Policy Document, Revision 24. Applicable regulation - 10CFR50.54(a).

2. Cooper Nuclear Station Fifth Ten-Year Interval Inservice Inspection Program and Third Ten-Year Interval Containment Inservice Inspection Program, Revision 2. Applicable regulation - 10CFR50.55a(g).

3. NPPD Emergency Plan for Cooper Nuclear Station, Revision 72. Applicable - 10CFR50.54(q).

4. NPPD Cooper Nuclear Station controlled drawings listed in Table I-3-1. Applicable regulation - 10CFR50.59 (as applied to the CNS Drawing Control Program).

5. Cooper Nuclear Station Offsite Dose Assessment Manual for Gaseous and Liquid Effluents (ODAM), October, 2018. Applicable regulatory requirement - Technical Specifications Section 5.5.1.c. 10CFR50.59 and 10CFR50.71(e) apply to major changes to radioactive waste treatment systems (liquid, gaseous, and solid) as specified in ODAM Section D 5.5.

6. Cooper Nuclear Station Technical Requirements Manual Limiting Conditions for Operation (TLCOs). Applicable regulation - 10CFR50.59. License Condition 2.C.(4) applies to TRM Section 3.11.

7. Cooper Nuclear Station Plant Unique Analysis Report, February, 2007. Applicable regulation - 10CFR50.59.

TABLE I-3-1
CONTROLLED DRAWINGS INCORPORATED BY REFERENCE

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
DPPPG6002043800 B008247-980500	5	SIEMENS ENERGY	COOPER #1, BB96FA- 2XBB81-13.9m2, UPGRADED HP, HEAT FLOW DIAGRAM	01
117C3346		GENERAL ELECTRIC	INSTALLATION ELECTRICAL PENETRATION SEAL, MEDIUM VOLTAGE POWER	01
117C3349		GENERAL ELECTRIC	INSTALLATION PENETRATION SEAL, INDICATION & CONTROL	00
161F282BC	1	GENERAL ELECTRIC	PROCESS DIAGRAM, CORE SPRAY SYSTEM	06
197R567	1	GENERAL ELECTRIC	PIPING & INSTRUMENT SYMBOLS	00
197R576	1	GENERAL ELECTRIC	ASSEMBLY REACTOR	04
2001	1	BURNS & ROE	FLOW DIAGRAM, SYMBOLS & ABBREVIATIONS	20
2001	2	BURNS & ROE	FLOW DIAGRAM SYMBOLS & ABBREVIATIONS	08
2002	1	BURNS & ROE	FLOW DIAGRAM, MAIN, EXHAUST & AUXILIARY STEAM SYSTEMS	48
2002	2	BURNS & ROE	FLOW DIAGRAM, MAIN, EXHAUST & AUXILIARY STEAM SYSTEMS	42
2004	1	BURNS & ROE	FLOW DIAGRAM CONDENSATE & FEEDWATER SYSTEMS	36
2004	2	BURNS & ROE	FLOW DIAGRAM, CONDENSATE & FEEDWATER SYSTEMS	51
2004	3	BURNS & ROE	FLOW DIAGRAM, CONDENSATE & FEEDWATER SYSTEMS	63
2006	1	BURNS & ROE	FLOW DIAGRAM, CIRCULATING & SCREEN WASH & SERVICE WATER SYSTEMS	90
2006	2	BURNS & ROE	FLOW DIAGRAM, CIRCULATING & SCREEN WASH & SERVICE WATER SYSTEMS	49
2006	3	BURNS & ROE	FLOW DIAGRAM, CIRCULATING & SCREEN WASH & SERVICE WATER SYSTEMS	56
2006	4	BURNS & ROE, NPPD	FLOW DIAGRAM, CONTROL BUILDING, SERVICE WATER SYSTEM	60
2007		BURNS & ROE	FLOW DIAGRAM, TURBINE BUILDING CLOSED COOLING WATER SYSTEM	84

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
2009		BURNS & ROE	FLOW DIAGRAM, AIR REMOVAL SYSTEM	32
2010	1	BURNS & ROE, NPPD	FLOW DIAGRAM, INSTRUMENT AIR, CONTROL & TURBINE BUILDING	B2
2010	1A	BURNS & ROE, NPPD	FLOW DIAGRAM, INSTRUMENT AIR, CONTROL & TURBINE BUILDINGS	15
2010	2	BURNS & ROE, NPPD	FLOW DIAGRAM, INSTRUMENT AIR, REACTOR BUILDING	98
2010	3	BURNS & ROE, NPPD	FLOW DIAGRAM, SERVICE AIR	48
2010	4	BURNS & ROE, NPPD	FLOW DIAGRAM, INSTRUMENT AIR, RADWASTE & AUGMENTED RADWASTE BUILDINGS	28
2010	5	BURNS & ROE, NPPD	INSTRUMENT & SERVICE AIR, MISCELLANEOUS DETAILS	25
2012	3	BURNS & ROE	FLOW DIAGRAM, ELECTRIC HEATING BOILER SYSTEM	23
2016	1	BURNS & ROE	FLOW DIAGRAM, FIRE PROTECTION TURBINE GENERATOR BUILDING	70
2016	1A	BURNS & ROE, NPPD	FLOW DIAGRAM, FIRE PROTECTION, SERVICE BUILDINGS & YARD	09
2016	1B	BURNS & ROE, NPPD	FLOW DIAGRAM, FIRE PROTECTION, CONTROL, RADWASTE AND AUGMENTED RADWASTE BUILDINGS	05
2016	1C	BURNS & ROE, NPPD	FLOW DIAGRAM, FIRE PROTECTION, REACTOR BUILDING	04
2017		BURNS & ROE	FLOW DIAGRAM, TURBINE & RADWASTE BUILDING, CONTAMINATED FLOOR DRAIN & RADWASTE ROOF DRAIN SYSTEM	16
2018		BURNS & ROE	FLOW DIAGRAM, TURBINE GENERATOR BUILDING & CONTROL BUILDING, HEATING & VENTILATING	41
2019	1	BURNS & ROE	FLOW DIAGRAM, MAIN CONTROL ROOM & CABLE ROOM & COMPUTER ROOM, HEATING & VENTILATING & AIR CONDITIONING	50
2019	2	BURNS & ROE, NPPD	FLOW DIAGRAM CHILLED WATER SYSTEM, ELECTRICAL SHOP, NONCRITICAL SWITCHGEAR, CONTROL BUILDING & OFFICE BUILDING	11
2020		BURNS & ROE, NPPD	FLOW DIAGRAM, REACTOR BUILDING, HEATING & VENTILATING	63

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
2021		BURNS & ROE	FLOW DIAGRAM, RADWASTE BUILDING, HEATING & VENTILATING	18
2026	1	BURNS & ROE, NPPD	REACTOR VESSEL INSTRUMENTATION, FLOW DIAGRAM	68
2027	1	BURNS & ROE, NPPD	FLOW DIAGRAM, LOOP A, REACTOR RECIRCULATION & SUPPRESSION CHAMBER VENT SYSTEMS & CONNECTIONS	75
2027	2	BURNS & ROE, NPPD	FLOW DIAGRAM, LOOP B, REACTOR RECIRCULATION & SUPPRESSION CHAMBER VENT SYSTEMS & CONNECTIONS	15
2028		BURNS & ROE	FLOW DIAGRAM, REACTOR BUILDING & DRYWELL EQUIPMENT DRAIN SYSTEM	53
2030	1	BURNS & ROE	FLOW DIAGRAM, FUEL POOL COOLING & CLEAN UP SYSTEM	33
2030	2	BURNS & ROE	FLOW DIAGRAM, FUEL POOL COOLING & CLEAN UP SYSTEM	16
2031	1	BURNS & ROE	FLOW DIAGRAM, REACTOR BUILDING, CLOSED COOLING WATER SYSTEM	24
2031	2	BURNS & ROE, NPPD	FLOW DIAGRAM, REACTOR BUILDING, CLOSED COOLING WATER SYSTEM	65
2031	3	BURNS & ROE	FLOW DIAGRAM, REACTOR BUILDING, CLOSED COOLING WATER SYSTEM	34
2032	1	BURNS & ROE	FLOW DIAGRAM, HIGH CONDUCTIVITY PROCESS, FLOOR DRAINS & CHEMICAL & LAUNDRY WASTE	32
2032	2	BURNS & ROE	FLOW DIAGRAM, HIGH CONDUCTIVITY PROCESS, FLOOR DRAINS & CHEMICAL & LAUNDRY WASTE	22
2032	3	BURNS & ROE	FLOW DIAGRAM, HIGH CONDUCTIVITY PROCESS, FLOOR DRAINS & CHEMICAL & LAUNDRY WASTE	15
2032	4	BURNS & ROE	FLOW DIAGRAM, HIGH CONDUCTIVITY PROCESS, FLOOR DRAINS & CHEMICAL & LAUNDRY WASTE	26
2032	5	BURNS & ROE	FLOW DIAGRAM, HIGH CONDUCTIVITY PROCESS, FLOOR DRAINS & CHEMICAL & LAUNDRY WASTE	09
2033	1	BURNS & ROE	FLOW DIAGRAM, LOW CONDUCTIVITY PROCESSING, EQUIPMENT DRAINS	21
2033	2	BURNS & ROE	FLOW DIAGRAM, LOW CONDUCTIVITY PROCESSING, EQUIPMENT DRAINS	25

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
2033	3	BURNS & ROE	FLOW DIAGRAM, LOW CONDUCTIVITY PROCESSING, EQUIPMENT DRAINS	16
2033	4	BURNS & ROE	FLOW DIAGRAM, LOW CONDUCTIVITY PROCESSING, EQUIPMENT DRAINS	17
2034		BURNS & ROE	FLOW DIAGRAM, PLANT MAKEUP WATER TREATMENT SYSTEM	70
2035	1	BURNS & ROE	FLOW DIAGRAM, CONDENSATE FILTER DEMINERALIZER SYSTEM	17
2035	2	BURNS & ROE	FLOW DIAGRAM, CONDENSATE FILTER DEMINERALIZER SYSTEM	13
2035	3	BURNS & ROE	FLOW DIAGRAM, CONDENSATE FILTER DEMINERALIZER SYSTEM	13
2035	4	BURNS & ROE	FLOW DIAGRAM, CONDENSATE FILTER DEMINERALIZER SYSTEM	16
2036	1	BURNS & ROE, NPPD	FLOW DIAGRAM, REACTOR BUILDING, SERVICE WATER SYSTEM	A5
2037		BURNS & ROE	FLOW DIAGRAM, HEATING & VENTILATION STANDBY GAS TEATMENT & OFF GAS FILTERS	71
2038	1	BURNS & ROE	FLOW DIAGRAM, REACTOR BUILDING, FLOOR & ROOF DRAIN SYSTEMS	55
2039		BURNS & ROE, NPPD	FLOW DIAGRAM, CONTROL ROD DRIVE, HYDRAULIC SYSTEM	62
2040	1	BURNS & ROE, NPPD	FLOW DIAGRAM, RESIDUAL HEAT REMOVAL SYSTEM	82
2040	2	BURNS & ROE, NPPD	FLOW DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM, LOOP B	19
2042	1	BURNS & ROE	FLOW DIAGRAM, REACTOR WATER CLEANUP SYSTEM	37
2042	2	BURNS & ROE	FLOW DIAGRAM, REACTOR WATER CLEANUP SYSTEM	15
2042	3	BURNS & ROE	FLOW DIAGRAM, REACTOR WATER CLEANUP SYSTEM	22
2043		BURNS & ROE, NPPD	FLOW DIAGRAM, REACTOR CORE ISOLATION COOLANT & REACTOR FEED SYSTEMS	57
2044		BURNS & ROE, NPPD	FLOW DIAGRAM, HIGH PRESSURE COOLANT INJECTION & REACTOR FEED SYSTEMS	76
2045	1	BURNS & ROE, NPPD	FLOW DIAGRAM, CORE SPRAY SYSTEM	58
2045	2	BURNS & ROE, NPPD	FLOW DIAGRAM, STANDBY LIQUID CONTROL SYSTEM	21
2049	3	BURNS & ROE	FLOW DIAGRAM, CONDENSATE SUPPLY SYSTEM	20

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
2050		BURNS & ROE	GENERAL ARRANGEMENT, TURBINE BUILDING, BASEMENT FLOOR PLAN	17
2051		BURNS & ROE	GENERAL ARRANGEMENT, TURBINE BUILDING, MEZZANINE FLOOR PLAN	29
2052		BURNS & ROE	GENERAL ARRANGEMENT, TURBINE BUILDING, OPERATING FLOOR PLAN	40
2053		BURNS & ROE	GENERAL ARRANGEMENT, TURBINE BUILDING SECTION AA	02
2054		BURNS & ROE	GENERAL ARRANGEMENT, TURBINE BUILDING SECTION BB	03
2056		BURNS & ROE	GENERAL ARRANGEMENT, INTAKE STRUCTURE PLANS & SECTIONS	16
2059		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING, PLAN BELOW GRADE	05
2060		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING PLAN AT ELEVATION 903-6	15
2061		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING PLAN AT ELEVATION 931-6	12
2062		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING PLAN AT ELEVATION 958-3	07
2063		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING PLAN AT ELEVATION 976-0	07
2064		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING PLAN AT ELEVATION 1001-0	10
2065		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING SECTION AA	05
2066		BURNS & ROE	GENERAL ARRANGEMENT, REACTOR BUILDING SECTION BB	05
2067		BURNS & ROE	GENERAL ARRANGEMENT, RADWASTE BUILDING PLANS AT ELEVATION 877-6 & 903-6	13
2068		BURNS & ROE	GENERAL ARRANGEMENT, RADWASTE BUILDING, PLANS AT ELEVATION 918-0 & 934-0	13
2069		BURNS & ROE	GENERAL ARRANGEMENT, RADWASTE BUILDING SECTIONS	07
2072		BURNS & ROE	GENERAL ARRANGEMENT, AUGMENTED RADWASTE BUILDING PLAN	05

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
2073		BURNS & ROE	GENERAL ARRANGEMENT, AUGMENTED RADWASTE BUILDING SECTIONS	00
2079	1	BURNS & ROE	FLOW DIAGRAM, AUGMENTED LIQUID RADWASTE SYSTEM	14
2079	2	BURNS & ROE	FLOW DIAGRAM, AUGMENTED LIQUID RADWASTE SYSTEM	31
2080		BURNS & ROE	FLOW DIAGRAM, AUGMENTED LIQUID RADWASTE BUILDING EQUIPMENT, CHEMICAL AND FLOOR DRAINS	15
2084		BURNS & ROE, NPPD	FLOW DIAGRAM, STANDBY NITROGEN INJECTION SYSTEM	30
2298		BURNS & ROE	RADWASTE BUILDING, CONVEYOR OPERATION AREAS PLANS	00
3001		BURNS & ROE	MAIN ONE LINE DIAGRAM	30
3002	1	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM MCC Z, SWGR BUS 1A & 1B & 1E & CRITICAL SWGR BUS 1F & 1G	56
3003	2	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, MCC A & B & F & G	55
3004	3	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, MOTOR CONTROL CENTERS C& D & H & J & DG1 & DG2	25
3005	4	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, MOTOR CONTROL CENTERS M & N & P & U & V & W	78
3006	5	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, STARTER RACKS LZ & TZ, MOTOR CONTROL CENTERS K & L & LX & RA & RX & S & T & TX & X	90
3007	6	BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, MOTOR CONTROL CENTER E & Q & R & RB & Y	84
3009	1	BURNS & ROE, NPPD	ONE LINE SWITCHING DIAGRAM, 12.5 KV RING BUS SYSTEM	64
3010	1	BURNS & ROE, NPPD	VITAL ONE LINE DIAGRAM	85
3058		BURNS & ROE	DC ONE LINE DIAGRAM	68
3070		BURNS & ROE	ELECTRICAL SYMBOL LIST	13
3401		BURNS & ROE, NPPD	AUXILIARY ONE LINE DIAGRAM, MOTOR CONTROL CENTER CA & CB & MR & OG1 & OG2	33
4003, NF 13293		BURNS & ROE	CIVIL, OVERALL SITE & VICINITY PLAN	45

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
4215		BURNS & ROE	STRUCTURAL REACTOR BUILDING, SHIELDING BLOCKS, HPCI HATCH	03
4259	1	BURNS & ROE, NPPD	CONTAINMENT VESSEL, PENETRATION LOCATION	10
4259	1A	BURNS & ROE, NPPD	CONTAINMENT VESSEL PENETRATION SCHEDULE	03
4260	2A	BURNS & ROE, NPPD	CONTAINMENT SUPPRESSION CHAMBER PENETRATION LOCATION	00
4260	2B	BURNS & ROE, NPPD	CONTAINMENT SUPPRESSION CHAMBER PENETRATION SCHEDULE	05
4286		BURNS & ROE	ENERGY ABSORBING PANEL LINER LOCATIONS	04
6000302,72-17	1	COSMODYNE	PIPING & INSTRUMENTATION DIAGRAM, AUGMENTED OFF GAS SYSTEM	51
6000302,72-17	2	COSMODYNE	P & I DIAGRAM, AUGMENTED OFFGAS SYSTEM	24
719E415BB	1	GENERAL ELECTRIC	PIPING & INSTRUMENTATION DIAGRAM, NUCLEAR BOILER	14
719E479BB	1	GENERAL ELECTRIC, NPPD	PROCESS RADIATION MONITORING SYSTEM	13
719E580BB	1	GENERAL ELECTRIC	CONTROL ROD DRIVE HYDRAULIC SYSTEM	03
729E174BB	1	GENERAL ELECTRIC	RECIRCULATION FLOW CONTROL SYSTEM	01
729E211BB		GENERAL ELECTRIC	RESIDUAL HEAT REMOVAL SYSTEM, PROCESS DIAGRAM	12
729E222BB	1	GENERAL ELECTRIC	REACTOR PROECTION SYSTEM MPL 05	01
729E222BB	2	GENERAL ELECTRIC, NPPD	REACTOR PROTECTION SYSTEM, FUNCTIONAL CONTROL DIAGRAM	04
729E222BB	3	GENERAL ELECTRIC, NPPD	REACTOR PROTECTION SYSTEM	04
729E223BB	1	GENERAL ELECTRIC	NEUTRON MONITORING SYSTEM	00
729E223BB	2	GENERAL ELECTRIC	NEUTRON MONITORING SYSTEM	01
729E402BB	1	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, CORE SPRAY SYSTEM	06
729E471BB	1	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, CONTROL ROD DRIVE HYDRAULIC SYSTEM	02

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
729E471BB	2	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	03
729E471BB	3	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	02
729E471BB	4	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
729E471BB	5	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	02
729E471BB	6	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
729E471BB	7	GENERAL ELECTRIC, NPPD	CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM	03
729E517BC	1	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, REACTOR CORE ISOLATION COOLING SYSTEM	04
729E517BC	2	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, REACTOR CORE ISOLATION COOLING SYSTEM	06
729E517BC	3	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, REACTOR CORE ISOLATION COOLING SYSTEM, RCIC SYSTEM	06
729E589BB	1	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, HIGH PRESSURE COOLANT INJECTION SYSTEM	06
729E589BB	2	GENERAL ELECTRIC, NPPD	HIGH PRESSURE COOLANT INJECTION SYSTEM, FUNCTIONAL CONTROL DIAGRAM	06
729E589BB	3	GENERAL ELECTRIC, NPPD	HIGH PRESSURE COOLANT INJECTION SYSTEM, FUNCTIONAL CONTROL DIAGRAM	06
729E719BC	1	GENERAL ELECTRIC	PROCESS DIAGRAM, REACTOR CORE ISOLATION COOLANT SYSTEM (RCIC) SYSTEM	03
729E720BB		GENERAL ELECTRIC	PROCESS DIAGRAM, HIGH PRESSURE COOLANT INJECTION SYSTEM, HPCI	04
729E727BB	1	GENERAL ELECTRIC	RECIRCULATION FLOW CONTROL SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
729E727BB	2	GENERAL ELECTRIC	RECIRCULATION FLOW CONTROL SYSTEM	00

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
729E727BB	3	GENERAL ELECTRIC	RECIRCULATION FLOW CONTROL SYSTEM, FUNCTIONAL CONTROL DIAGRAM	02
729E989	1	GENERAL ELECTRIC	POWER RANGE MONITORING UNIT MPL 07	01
730E100	1	GENERAL ELECTRIC	NEUTRON MONITORING SYSTEM ARRANGEMENT	02
730E100	2	GENERAL ELECTRIC	NEUTRON MONITORING SYSTEM ARRANGEMENT	01
730E140BB	1	GENERAL ELECTRIC	RESIDUAL HEAT REMOVAL SYSTEM, FUNCTIONAL CONTROL DIAGRAM	10
730E140BB	2	GENERAL ELECTRIC	RESIDUAL HEAT REMOVAL SYSTEM	05
730E140BB	3	GENERAL ELECTRIC	RESIDUAL HEAT REMOVAL SYSTEM, FUNCTIONAL CONTROL DIAGRAM	10
730E148BB	1	GENERAL ELECTRIC	PROCESS DIAGRAM REACTOR WATER CLEANUP SYSTEM MPL 12	00
730E149BB	1	GENERAL ELECTRIC	NUCLEAR BOILER, FUNCTIONAL CONTROL DIAGRAM	05
730E149BB	2	GENERAL ELECTRIC	FUNCTIONAL CONTROL DIAGRAM, NUCLEAR BOILER, MISCELLANEOUS SYSTEM	02
730E197BB	6B	GENERAL ELECTRIC	REACTOR RECIRCULATION CONTROLLER CONFIGURATION	07
730E805BA	1	GENERAL ELECTRIC, NPPD	NEUTRON MONITORING SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
730E805BA	6	GENERAL ELECTRIC, NPPD	NEUTRON MONITORING SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
730E805BA	7	GENERAL ELECTRIC, NPPD	NEUTRON MONITORING SYSTEM, FUNCTIONAL CONTROL DIAGRAM	01
730E923		GENERAL ELECTRIC	INCORE STARTUP CHAMBER RETRACT MECHANISM, FIELD INSTALLATION	03
731E611	1	GENERAL ELECTRIC	PRIMARY STEAM PIPING	04
791E257	3	GENERAL ELECTRIC	ELEMENTARY DIAGRAM, FEEDWATER CONTROL SYSTEM	16
791E257	4	GENERAL ELECTRIC	ELEMENTARY DIAGRAM, FEEDWATER CONTROL SYSTEM	31
80E1143		AUTOMATION INDUSTRIES	FUEL STORAGE RACK	01
80E1148		AUTOMATION INDUSTRIES	FUEL STORAGE RACK 13 & SEISMIC BRACING, INSTALLATION IN POOL	03
919D690BC	3	GENERAL ELECTRIC	REACTOR VESSEL	01

DRAWING NUMBER	SHEET	VENDOR	TITLE	REVISION
919D690BC	5	GENERAL ELECTRIC	REACTOR VESSEL	01
920D225BB	1	GENERAL ELECTRIC, NPPD	STANDBY LIQUID CONTROL SYSTEM, FUNCTIONAL CONTROL DIAGRAM	03
921D796	1	GENERAL ELECTRIC, NPPD	FUNCTIONAL CONTROL DIAGRAM, AREA RADIATION MONITORING SYSTEM	02
CNS-MS-43		NPPD	LEAKAGE PATHS FROM OUTBOARD MSIV'S	05
CNS-NBI-10		NPPD	REACTOR WATER LEVEL INDICATION CORRELATION	06
CP001	1	REACTOR CONTROLS	CONTROL ROD DRIVE HYDRAULIC SYSTEM DIAGRAM	13
ILE70-3, ID105	107B, 3A	BURNS & ROE	AREA TEMPERATURE MONITOR SYSTEM, FOR NUCLEAR BOILER SYSTEM, LEAK DETECTION	00
ILE70-3, ID105	107C, 4	BURNS & ROE	INSTALLATION DETAILS, TEMPERATURE DETECTORS FOR NUCLEAR BOILER SYSTEMS LEAK DETECTION	04
ILE70-3, ID105	107E, 6	BURNS & ROE	INSTALLATION DETAILS, TEMPERATURE DETECTORS FOR NUCLEAR BOILER SYSTEMS LEAK DETECTION	00
ILE70-3, ID105	107F, 7	BURNS & ROE	INSTALLATION DETAILS, TEMPERATURE DETECTORS FOR NUCLEAR BOILER SYSTEMS LEAK DETECTION	00
NC29546		NPPD	TRANSMISSION LINE ROUTES	02
NC44587		NPPD	ROUTING OF 12.5 KV UNDERGROUND SYSTEM	18
SAA085	4	RALPH A. HILLER COMPANY	20 INCH BY 5 INCH MSIV ACTUATOR	01
SK101670R	1	BURNS & ROE	CONTAINMENT PENETRATION ASSEMBLY, EXTERNAL PENETRATION CONTROLS	02
SKM200, M200	1	BURNS & ROE	CONTAINMENT PENETRATION X7AD, X9A, X9B, ASSEMBLY & EXTERIOR PENETRATION CONTROLS	00

4.0 PRINCIPAL DESIGN CRITERIA

In meeting the FSAR content requirement of 10CFR50.34(b) for describing the station design bases it is necessary to distinguish those station structures, systems, and components (SSCs) which are required to meet specified measures of safety from those that are not. To be valid, such a determination must be performed in a consistent, systematic manner. The principal design criteria, for general and nuclear safety, has been determined by the necessary functions of a system to respond to planned operation, abnormal operational transients, accidents, and special events. The actual design of SSCs reflects the criteria that pertain to it.

The design of Cooper Nuclear Station preceded issuance of 10CFR50 Appendix A "General Design Criteria for Nuclear Power Plants" and the established regulatory concepts of safety-related and non-safety-related. However, CNS conformance to the proposed 10CFR50 Appendix A General Design Criteria is provided in Appendix F.

A systematic method has been developed to evaluate functionally the nuclear safety aspects of the BWR. The first step in this method is to specify in measurable terms the major judgements to be considered as the primary safety requirements. These top level judgements are offered as unacceptable safety results. Table I-4-1 describes the set of unacceptable safety results associated with each major category of station events. These unacceptable safety results are those used in evaluating the Cooper Nuclear Station.

Using the unacceptable safety results, the criteria for selecting the events of each category, and a consistent set of ground rules for evaluating the station events, it is possible to identify all the station actions required to avoid the unacceptable safety results. Such actions are called safety actions. Similarly, it is possible to identify the SSCs and functions required to avoid the unacceptable safety results. The station Nuclear Safety Operational Analysis (Appendix G) details this analysis method and presents the results. A major distinction is made between those BWR aspects which are required for the avoidance of the unacceptable safety results and those which are most pertinent to the station mission - the generation of electrical power.

Two methods are provided for describing the principal design criteria. The first method is by functional classification (either safety or power generation). The second is on a system-by-system (or system group) basis.

Safety analysis requires the information gained in the functional classification approach to criteria, but system description is more easily understood through the system-by-system method. In this section both approaches to criteria are given; both are useful.

4.1 Principal Design Criteria - Functional Classification

In the functional classification approach the criteria must be stated in sufficient detail to allow placement of each criterion into one classification category. Thus, there may be closely related criteria pertaining to any given system in each category. This is a natural outgrowth of the functional (unacceptable result) approach to classification. The actual design of a system must reflect all of the criteria that pertain to it.

The principal architectural and engineering criteria for the design and construction of the station are summarized below. Some of the more general criteria are so broad that they are applicable, at least in part, to more than one classification. In these very general cases all of the affected classifications are indicated. Specific design bases and design features are detailed in other sections of the USAR.

TABLE I-4-1

UNACCEPTABLE SAFETY RESULTS FOR STATION
EVENT CATEGORIES

	Event Category		Unacceptable Safety Results
1.	Planned Operation	1-1	The release of radioactive material to the environs to such an extent that the limits of 10CFR20 are exceeded.
		1-2	Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded.
		1-3	Nuclear system stress in excess of that allowed for planned operation by applicable industry code.
		1-4	The existence of a station condition not considered by station safety analyses.
2.	Abnormal Operational Transients	2-1	The release of radioactive material to the environs to such an extent that the limits of 10CFR20 are exceeded.
		2-2	Any fuel failure calculated as a direct result of the transient analyses.
		2-3	Nuclear system stress in excess of that allowed for transients by applicable industry codes.
3.	Accidents	3-1	Radioactive material release to such an extent that the guideline values of 10CFR100 (or 10CFR50.67 for Fuel Handling or Loss of Coolant Accident) would be exceeded.
		3-2	Catastrophic failure of the fuel barrier as a result of exceeding mechanical or thermal limits.
		3-3	Nuclear system stresses in excess of that allowed for accidents by applicable industry codes.
		3-4	Containment stresses in excess of that allowed by industry standard or code when containment is required.
		3-5	Overexposure to radiation of station operation personnel in the control room.
4.A	Special Event – Ability to Shutdown Reactor from Outside Control Room	4-1	The inability to bring the reactor to the shutdown condition by manipulation of the local controls and equipment which are available outside the control room.
		4-2	The inability to bring the reactor to the cold shutdown condition from outside the control room.
B	Special Event – Ability to Shutdown Reactor Without Control Rods	4-3	The inability to shutdown the reactor independent of control rods.
C	Special Event – Ability to Mitigate the Consequences of an ATWS	4-4	Exceeding limits based on 10CFR50.62.
D	Special Event – Station Blackout	4-5	Inability to cope with a station blackout for a specified duration in accordance with 10CFR50.63.

4.1.1 General Criteria

1. The station shall be designed so that it can produce electric power in a safe and reliable manner. The station design shall be in accordance with applicable codes and regulations.

2. The station shall be designed in such a way that the release of radioactive materials to the environment is limited, so that the limits and guideline values of applicable regulations pertaining to the release of radioactive materials are not exceeded.

3. The reactor core and reactivity control system shall be designed so that control rod action shall be capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

4. Adequate strength and stiffness with appropriate safety factors shall be provided so that a hazardous release of radioactive material shall not occur.

4.1.2 Power Generation Design Criteria, (Planned Operation)

1. The nuclear system shall employ a boiling water reactor to produce steam for direct use in a turbine-generator.

2. The fuel cladding shall be designed to retain integrity as a radioactive material barrier for the design power range.

3. The fuel cladding shall be designed to accommodate without loss of integrity the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

4. Heat removal systems shall be provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from plant shutdown to design power. The capacity of such systems shall be adequate to prevent fuel clad damage.

5. Control equipment shall be provided for recirculation flow control to allow the reactor recirculation flow to be manually adjusted.

6. It shall be possible to manually control the reactor power level.

7. Control of the nuclear system shall be possible from a single location.

8. Nuclear system process controls shall be arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

9. Fuel handling and storage facilities shall be designed to maintain adequate cooling for spent fuel.

10. Interlocks or other automatic equipment shall be provided as a backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineered safety features.

4.1.3 Power Generation Design Criteria, (Abnormal Operational Transients)

1. The fuel cladding, in conjunction with other station systems, shall be designed to retain integrity throughout any abnormal operational transient.

2. Heat removal systems shall be provided in sufficient capacity and operational adequacy to remove heat generated for any abnormal operational transient.

3. Standby electrical power sources shall be provided to allow removal of decay heat under circumstances where normal auxiliary power is not available.

4.1.4 Nuclear Safety Design Criteria, (Planned Operation)

1. The station shall be designed so that fuel failure during planned operation is limited to such an extent that if the freed fission products were released to the environs, via the normal discharge paths, the activity levels would be lower than permitted under 10CFR20.

2. The reactor core shall be designed so that its nuclear characteristics do not contribute to a divergent power transient.

3. The nuclear system shall be designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate station systems.

4. Gaseous, liquid, and solid waste disposal facilities shall be designed so that the discharge and off-site shipment of radioactive effluents can be made in accordance with applicable regulations.

5. The design shall provide means by which station operations personnel can be informed whenever limits on the release of radioactive material are exceeded.

6. Sufficient indications shall be provided to allow determination that the reactor is operating within the envelope of condition considered by station safety analysis.

7. Radiation shielding shall be provided and access control patterns shall be established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal station operation (see Section XII-3.1.1).

8. Fuel handling and storage facilities shall be designed to prevent inadvertent criticality.

9. Fuel handling and storage facilities shall be designed to maintain adequate shielding for spent fuel.

4.1.5 Nuclear Safety Design Criteria, (Abnormal Operational Transients)

1. The station shall be designed so that no calculated fuel failure occurs as a result of any abnormal operational transient.

2. Those portions of the nuclear system which form part of the reactor coolant pressure boundary shall be designed to retain integrity as a radioactive material barrier following abnormal operational transients.

3. Nuclear safety systems shall act to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients.

4. Where positive, precise action is immediately required in response to abnormal operational transients, such action shall be automatic and shall require no decision or manipulation of controls by operations personnel.

5. Required safety actions shall be carried out by equipment of sufficient redundancy and independence that no single failure of active components can prevent the required actions.

6. The design of nuclear safety systems shall include allowances for environmental phenomena at the site.

7. Provision shall be made for control of active components of nuclear safety systems from the Main Control Room.

8. Nuclear safety systems shall be designed to permit demonstration of their functional performance requirements.

9. Standby electrical power sources shall be provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

10. Standby electrical power sources shall have sufficient capacity to power all nuclear safety systems requiring electrical power.

4.1.6 Nuclear Safety Design Criteria, (Accidents)

1. Those portions of the nuclear system which form part of the reactor coolant pressure boundary shall be designed to retain integrity as a radioactive material barrier following accidents.

2. Engineered safety features shall act to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by accident.

3. Where positive, precise action is immediately required in response to accidents, such action shall be automatic and shall require no decision or manipulation of controls by operations personnel.

4. Required safety actions shall be carried out by equipment of sufficient redundancy and independence that no single failure of active components can prevent the required actions. For systems or components to which IEEE 279 is applicable, single failures of passive electrical components will be considered as well as single failure of active components in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components. Note: The one exception to this criterion at CNS is the Control Room Emergency Filter System (CREFS). The CREFS is designed to provide the safety function of maintaining Main Control Room habitability in the event of a design basis accident (see Item 23 below). However, the CREFS at CNS as designed is a single train system which does not meet the redundancy requirements described above, and therefore, is not completely single failure proof. For a more detailed description of the safety function of the CREFS, refer to Section X-10.4.

5. Features of the station which are required for the mitigation of accident consequences shall be designed so that they can be fabricated and erected to quality standards which reflect the importance of the safety action to be performed. Note: An exception to this criterion at CNS is the MSIV Leakage Pathway, and the Main Turbine Condenser complex. These Class IIS SSCs are credited for mitigation of DBA LOCA and CRDA dose consequences, but have been analyzed as capable of withstanding the seismic loadings of a Safe Shutdown Earthquake. SLC is also credited for the mitigation of a DBA LOCA even though it is not a safety-related system. For a more detailed discussion of the safety function of these SSCs, refer to Sections III-9, IV-11 and XI-3.

6. The design of engineered safety features shall include allowances for environmental phenomena at the site.

7. Provision shall be made for control of active components of engineered safety features from the Main Control Room.

8. Engineered safety features shall be designed to permit demonstration of their functional performance requirements.

9. A primary containment shall be provided that completely encloses the reactor vessel.

10. The primary containment shall be designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume.

11. It shall be possible to test primary containment integrity and leak tightness at periodic intervals.

12. A secondary containment shall be provided that completely encloses both the primary containment and fuel storage areas.

13. The secondary containment shall be designed to act as a radioactive material barrier under the same conditions that require the primary containment to act as a radioactive material barrier.

14. The secondary containment shall be designed to act as a radioactive material barrier, if required, whenever the primary containment is open for expected operational purposes.

15. The primary and secondary containments, in conjunction with other engineered safety features, shall act to prevent the radiological effects of accidents resulting in the release of radioactive material to the containment volumes from exceeding the guideline values of applicable regulations.

16. Provisions shall be made for the removal of energy from within the primary containment as necessary to maintain the integrity of the containment system following accidents that release energy to the primary containment.

17. Piping that both penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material to the environs shall be automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to prevent radiological effects from exceeding the guideline values of applicable regulations.

18. ECCS shall be provided to prevent the fuel clad temperature from exceeding 2200°F as a result of a loss-of-coolant accident and shall meet all acceptable criteria of 10CFR50.46.

19. The ECCS shall provide for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.

20. The ECCS shall be diverse, reliable, and redundant.

21. Operation of the ECCS shall be initiated automatically when required.

22. Standby electrical power sources shall have sufficient capacity to power all engineered safety features requiring electrical power.

23. The Main Control Room shall be shielded against radiation so that continued occupancy under accident conditions is possible.

4.1.7 Nuclear Safety Design Criteria, (Special Event)

1. Backup reactor shutdown capability shall be provided independent of normal reactivity control provisions (control rods). This backup system shall have the capability to shut down the reactor from any operating condition, and subsequently to maintain the shutdown condition.

2. In the event that the Main Control Room becomes inaccessible, it shall be possible to bring the reactor from power range operation to a hot shutdown condition by manipulation of the local controls and equipment which are available outside of the Main Control Room. Furthermore, station design shall not preclude the ability, in this event, to bring the reactor to a cold shutdown condition from the hot shutdown condition.

3. Under conditions indicative of an anticipated transient without scram (ATWS), the reactor coolant recirculating pumps will be automatically tripped.

4. In the event the reactor protection system (RPS) fails to insert the control rods, an alternate rod insertion system that is independent from the RPS from sensor output to the final actuation device will initiate control rod insertion.

5. During an ATWS, the backup reactor shutdown system will produce reactor shutdown (independent of control rods) quickly enough to maintain acceptable core and primary containment conditions.

6. The station will be able to cope with a loss of all AC power (station blackout) for the credited duration in accordance with 10CFR50.63.

4.2 Principal Design Criteria, System-By-System

The principal architectural and engineering criteria for design are summarized below on a system-by-system or system group basis. The system-by-system presentation facilitates the understanding of the actual design of any one system.

In the system-by-system presentation of criteria, only the most restrictive of any related criteria are stated for a system.

4.2.1 General Criteria

1. The station shall be designed so that it can be fabricated, erected, and operated to produce electric power in a safe and reliable manner. The design shall be in accordance with applicable codes and regulations.

2. The station shall be designed in such a way that the release of radioactive materials to the environment will be as low as practicable, which will be lower than permitted by the applicable regulations pertaining to the release of radioactive materials.

4.2.2 Nuclear System Criteria

1. The nuclear system shall employ a General Electric boiling water reactor to produce steam for direct use in a turbine-generator.

2. The fuel cladding shall be designed to retain integrity as a radioactive material barrier for the design power range and for any abnormal operational transient.

3. Those portions of the nuclear system which form part of the reactor coolant pressure boundary shall be designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents.

4. The fuel cladding shall be designed to accommodate without loss of integrity the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

5. Heat removal systems shall be provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions from shutdown to design power, and for any abnormal operational transient. The capacity of such systems shall be adequate to prevent fuel clad damage.

6. Heat removal systems shall be provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems shall be adequate to prevent fuel clad damage.

7. The reactor core and reactivity control system shall be designed so that control rod action shall be capable of bringing the core subcritical and maintaining it so, even with the rod at highest reactivity worth fully withdrawn and unavailable for insertion.

8. The nuclear steam supply system shall be designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate station systems.

9. The reactor core shall be designed so that its nuclear characteristics do not contribute to a divergent power transient.

4.2.3 Power Conversion Systems Criteria

Components of the power conversion systems shall be designed to perform two basic objectives:

1. Produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gases and particulate impurities removed.

2. Assure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

4.2.4 Electrical Power System Criteria

The station electrical power systems shall be designed to efficiently deliver the electrical power generated to the 345 kV transmission system.

Sufficient normal and standby auxiliary sources of electrical power shall be provided to attain prompt shutdown and continued maintenance of the station in a safe condition. The capacity of the power sources shall be adequate to accomplish all required engineered safety feature functions under postulated design basis accident conditions.

4.2.5 Radioactive Waste Disposal Criteria

Gaseous, liquid, and solid waste disposal facilities shall be designed so that the discharge and off-site shipment of radioactive effluents will be in accordance with 10CFR20.

Process and discharge streams shall be appropriately monitored and such features incorporated as may be necessary to maintain releases below the permissible limits of 10CFR20.

4.2.6 Nuclear Safety Systems and Engineered Safety Features Criteria

4.2.6.1 General

1. Nuclear safety systems shall act in response to abnormal operational transients so that there will be no calculated fuel failure.

2. Nuclear safety systems and engineered safety features shall act to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients or accidents.

3. Where positive, precise action is immediately required in response to accidents, such action shall be automatic and shall require no decision or manipulation of controls by operations personnel.

4. Required safety actions shall be carried out by equipment of sufficient redundancy and independence so that no single failure of active components can prevent the required actions. For systems or components to which IEEE 279 is applicable, single failures of passive electrical components will be considered as well as single failure of active components in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

5. Features of the station which are required for the mitigation of accident consequences shall be designed so that they can be fabricated and erected to quality standards which reflect the importance of the safety function to be performed.

6. The design of nuclear safety systems and engineered safety features shall include allowances for environmental phenomena at the site.

7. Provision shall be made for control of active components of nuclear safety systems and engineered safety features from the Main Control Room.

8. Nuclear safety systems and engineered safety features shall be designed to permit demonstration of their functional performance requirements.

4.2.6.2 Containment and Isolation Criteria

1. A primary containment shall be provided to completely enclose the reactor vessel. It shall be designed to act as a radioactive material barrier during and following accidents that release radioactive material into the primary containment. It shall be possible to test the primary containment integrity and leak tightness at periodic intervals.

2. A secondary containment that completely encloses both primary containment and fuel storage areas shall be provided and shall be designed to act as a radioactive material barrier.

3. The primary and secondary containments, in conjunction with other engineered safety features, shall act to prevent the release of radioactive material from within the containment volumes from exceeding the limiting values of applicable regulations.

4. Provisions shall be made for the removal of energy from within the primary containment as necessary to maintain the integrity of the containment system following accidents that release energy to the primary containment.

5. Piping that both penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material to the environs shall be automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to prevent radiological effects from exceeding the limiting values of applicable regulations.

4.2.6.3 Emergency Core Cooling Systems (ECCS) Criteria

1. ECCS shall be provided to prevent the fuel clad temperature from exceeding 2200°F as a result of a loss-of-coolant accident.

2. The ECCS shall provide for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.

3. The ECCS shall be diverse, reliable, and redundant.

4. Operation of the ECCS shall be initiated automatically when required regardless of the availability of offsite power supplies and the normal generating system of the plant.

4.2.6.4 Standby Power Criteria

Standby electrical power sources shall be provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available. They shall also provide sufficient power to all engineered safety features requiring electrical power.

4.2.7 Reactivity Control Criteria

1. Backup reactor shutdown capability shall be provided independent of normal reactivity control provisions. This backup system shall have the capability to shut down the reactor from any operating condition, and subsequently to maintain the shutdown condition. It shall also be able to produce a reactor shutdown (independent of control rods) quickly enough during postulated ATWS conditions to maintain acceptable core and primary containment conditions.

2. In the event that the Main Control Room is inaccessible, it shall be possible to bring the reactor from power range operation to a cold shutdown condition by manipulation of the local controls and equipment which are available outside of the Main Control Room.

4.2.8 Process Control Systems Criteria

4.2.8.1 Nuclear Steam Supply System Process Control Criteria

1. It shall be possible to manually control the reactor power level.

2. Control of the NSSS shall be possible from a single location.

3. NSSS process controls shall be arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

4. Interlocks or other automatic equipment shall be provided as a backup to procedural controls to avoid conditions requiring the actuation of nuclear safety systems or engineered safety features.

4.2.8.2 Power Conversion Systems Process Control Criteria

1. Controls shall be provided to maintain temperature and pressure to below design limitations. This system will result in a stable operation and response to all allowable variations.

2. Controls shall be designed to provide indication of system trouble.

3. Control of the power conversion system shall be possible from a single location.

4. Sensors shall be provided to detect a loss of the main condenser.

5. Controls shall be provided to ensure adequate cooling of power conversion system equipment.

6. Controls shall be provided to ensure adequate condensate purity.

7. Controls shall be provided to regulate the supply of water so that adequate reactor vessel water level is maintained.

4.2.8.3 Electrical Power System Process Control Criteria

1. Controls shall be provided to ensure that sufficient electrical power is provided for startup and normal operation and to attain prompt shutdown and continued maintenance of the station in a safe condition.

2. Control of the electrical power system shall be possible from a single location.

4.2.9 Auxiliary Systems Criteria

1. Multiple independent station auxiliary systems shall be provided for the purpose of cooling and servicing the station, the reactor and the station containment systems under various normal and abnormal conditions.

2. Fuel handling and storage facilities shall be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel.

4.2.10 Shielding and Access Control Criteria

1. Radiation shielding shall be provided and access control patterns shall be established to allow the operating staff to control radiation doses within the limits of applicable regulations in any mode of normal station operation (see Section XII-3.1.1). The design and establishment of the above shall include conditions which deal with fission product release from failed fuel elements and contamination of station areas from system leakage.

2. The Main Control Room shall be shielded against radiation and have suitable environmental control so that occupancy under design basis accident conditions is possible.

4.2.11 Structural Loading Criteria

The station structures shall be designed to withstand all applicable loading conditions, including environmental loads, so that a hazardous release of radioactive material shall not occur.

5.0 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Certain structures, systems, and components (SSCs) perform active or passive functions that support the Safety Design Criteria described in Section I-4. The ways in which SSCs work together to avoid the Unacceptable Safety Results associated with Normal Plant Operations, Anticipated Operational Transients, Accidents, and Special Events is explained in the Nuclear Safety Operational Analysis (NSOA) of Appendix G. As explained in Appendix G, the NSOA provides a high level basis for classifying SSCs as being of primary importance to safety or power generation. Based on this and other analyses, classification groups have been established.

5.1 SSC Quality Group Classifications

Appendix A describes the classification methodologies used by the NSSS supplier and the architect-engineer for fluid systems under their contract scope. The current methodology for classifying piping equipment pressure parts, including ASME classification, is also described in Appendix A. The NPPD Quality Assurance Program for Operation Policy Document specifies the criteria for application of the CNS Quality Assurance program to SSCs.

In response to Generic Letter 83-28, component level classifications have been established which use the terms “Essential” and “Non-Essential.” These classifications are similar to the more common industry terms “safety-related” and “non-safety-related.” An additional classification category is used for electrical equipment important to safety (both safety-related and non-safety-related) that is required to be environmentally qualified per 10CFR50.49. Figure I-5-1 shows the relationship of “safety-related” to “safety” and other terminologies used in the USAR.

5.2 Loading Classification

Structures and equipment are designed to resist structural and mechanical damage due to loads produced by environmental and thermal forces. For the purpose of categorizing mechanical strength designs for these loads, the following definitions are established:

a. Class I - Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a safe shutdown of the plant and removal of decay and sensible heat.

b. Class II - Structures and equipment which may be essential to the operation of the station, but which are not essential to a safe shutdown.

Section XII-2 describes these categories in detail and Appendix C discusses the structural loading criteria of Class I SSCs.

6.0 COMPARISON OF PRINCIPAL DESIGN CHARACTERISTICS

This USAR section contains historical information as indicated by the italicized text. USAR Section I-3.4 provides a more detailed discussion of historical information. The factual information being presented in this section on the comparable plant data between CNS and other boiling water reactors has been preserved as it was originally submitted to the Atomic Energy Commission in the CNS FSAR. While no attempt has been made to update the information on the other BWRs, salient contemporary CNS information may be found in USAR Sections I-2, III-2, III-3, III-4, III-5, III-6, III-7, IV-3, IV-7, IV-8, V-2, V-3, VI-3, VI-4, VII-5, VII-9, VIII-2, VIII-3, VIII-5, VIII-6, X-5, X-8, XI-2, XI-3, XI-5, XI-6, XI-8, and Appendix C.

This section highlights the principal design features of the station and provides a comparison of the major features with other boiling water reactor facilities.

The design of this facility is based upon proven technology attained during the development, design, construction, and operation of boiling water reactors of similar or identical types. The data, performance characteristics, and other information presented herein represent a current firm design.

6.1 Nuclear System Design Characteristics

Table I-6-1 summarizes the design and operating characteristics for the Cooper Nuclear Station. The same characteristics are presented for the nuclear systems of the Vermont Yankee Nuclear Power Station, Monticello Nuclear Generating Plant, Duane Arnold Energy Center, and the Browns Ferry Nuclear Plant, all of which have been reviewed by the Atomic Energy Commission and for which construction permits have been issued.

6.2 Power Conversion Systems Design Characteristics

Table I-6-2 summarizes the design and operating characteristics for the Cooper Nuclear Station. These same characteristics are, again, presented for the Vermont Yankee Nuclear Power Station, Monticello Nuclear Generating Plant, Duane Arnold Energy Center, and the Browns Ferry Nuclear Plant.

6.3 Electrical Power Systems Design Characteristics

The electrical power system design characteristics are presented in Table I-6-3 together with those of the Vermont Yankee Nuclear Power Station, Monticello Nuclear Generating Plant, Duane Arnold Energy Center, and the Browns Ferry Nuclear Plant.

6.4 Containment Design Characteristics

Table I-6-4 summarizes the design characteristics for the primary and secondary containments of the Cooper Nuclear Station. Design characteristics are also presented for the primary and secondary containment systems employed for Vermont Yankee Nuclear Power Station, Monticello Nuclear Generating Plant, Duane Arnold Energy Center, and the Browns Ferry Nuclear Plant. In addition, data is given for the type, construction, and height of elevated release point for the above plants.

6.5 Structural Design Characteristics

The seismic and wind loading design of the Cooper Nuclear Station are given in Table I-6-5. The seismic and wind loading design of the Vermont Yankee Nuclear Power Station, Monticello Nuclear Generating Plant, Duane Arnold Energy Center, and the Browns Ferry Nuclear Plant are presented for comparison.

TABLE I-6-1

*COMPARISON OF NUCLEAR SYSTEM DESIGN CHARACTERISTICS**

(PARAMETERS ARE RELATED TO REFERENCE DESIGN THERMAL OUTPUT FOR A SINGLE UNIT UNLESS OTHERWISE NOTED)

THERMAL AND HYDRAULIC DESIGN	VERMONT YANKEE	MONTICELLO**	DUANE ARNOLD ENERGY CENTER	COOPER STATION	BROWNS FERRY UNITS 1 & 2
Reference Design Thermal Output, Mwt	1593	1670	1593	2381	3293
Maximum Anticipated Thermal Output, Mwt	1665	1670	1670	2500	3440
Steam Flow Rate, lb/hr	6.43×10^6	6.77×10^6	6.843×10^6	9.81×10^6	13.38×10^6
Core Coolant Flow Rate, lb/hr	48.0×10^6	57.6×10^6	48.5×10^6	74.5×10^6	102.5×10^6
Feedwater Flow Rate, lb/hr	6.43×10^6	6.75×10^6	6.822×10^6	9.81×10^6	13.38×10^6
Feedwater Temperature, °F	372	376.3	420	367.1	376.1
System Pressure, Nominal in Steam Dome, psia	1020	1020	1020	1020	1015
Average Power Density, kw/liter	50.8	40.6	50.9	51.2	50.18
Maximum Design Limit Output, kw/ft	18.37	17.5	18.5	18.5	18.4
Average Thermal Output, kw/ft	7.1	5.7	7.079	7.079	7.049
Maximum Heat Flux, Btu/hr-ft^2	426,210	405,000	428,400	427,820	425,000
Average Heat Flux, Btu/hr-ft^2	163,900	131,346	163,933	164,500	163,200
Maximum UO_2 Temperature, °F	4380	4450	4380	4380	4380
Average Volumetric Fuel Temperature, °F	1100	900	1100	1100	1100
Average Fuel Rod Surface Temperature, °F	558	558	558	558	558
Minimum Critical Heat Flux Ratio (MCHFR)	≥ 1.9	≥ 1.9	≥ 1.9	≥ 1.9	≥ 1.9
Coolant Subcooling at Core Inlet, Btu/lb	27.1	27.0	24.4	29.9	28.7
Core Maximum Exit Voids Within Assemblies, %	79	76	79	79	79
Core Average Exit Quality, % Steam	13.6	12.0	14.3	13.2	13.2

*Items noted NA not available at the time of printing this table.

**Data for this plant is for the "as-built" conditions reported in the final safety analysis.

TABLE I-6-1 (CONTINUED)

DESIGN POWER PEAKING FACTOR	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Transverse Peaking Factor	*1.4*	*1.58*	*1.405*	*1.4*	*1.4*
Local Peaking Factor	*1.24*	*1.24*	*1.24*	*1.24*	*1.24*
Axial Peaking Factor	*1.5*	*1.57*	*1.5*	*1.5*	*1.5*
Total Peaking Factor	*2.6*	*3.08*	*2.6*	*2.6*	*2.6*
Nuclear Design (First Core)					
Water/UO_2 Volume Ratio (Cold)	*2.41*	*2.42 (Undished)*	*2.41*	*2.41*	*2.41*
Reactivity with Strongest Control Rod Out, k_{eff}	*< 0.99*	*< 0.99*	*< 0.99*	*< 0.99*	*< 0.99*
Moderator Temperature Coefficient					
At 68 °F, Δ K/K - °F Water	-5.0×10^{-5}	-8.0×10^{-5}	-3.5×10^{-5}	-3.5×10^{-5}	-3.5×10^{-5}
Hot, No Voids, Δ K/K - °F Water	-17.0×10^{-5}	-17.0×10^{-5}	-11.6×10^{-5}	-11.6×10^{-5}	-11.6×10^{-5}
Moderator Void Coefficient					
Hot, No Voids, Δ K/K - % Void	-1.0×10^{-3}	-1.0×10^{-3}	-8.7×10^{-4}	-8.7×10^{-4}	-8.7×10^{-4}
At Rated Output, Δ K/K - % Void	-1.6×10^{-3}	-1.4×10^{-3}	-1.05×10^{-3}	-1.05×10^{-3}	-1.05×10^{-3}
Fuel Temperature Doppler Coefficient					
At 68 °F, Δ K/K - °F Fuel	-1.3×10^{-5}	-1.2×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Hot, No Voids, Δ K/K - °F Fuel	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×110^{-5}	-1.2×10^{-5}
At Rated Output, Δ K/K - °F Fuel	$\leq 1.3 \times 10^{-5}$	$\leq 1.2 \times 10^{-5}$	$\leq 1.3 \times 10^{-5}$	$\leq 1.3 \times 10^{-5}$	$\leq 1.3 \times 10^{-5}$
Initial Average U-235 Enrichment, W/O	*2.29%*	*2.25%*	*2.25%*	*2.17%*	*2.19%*
Fuel Average Discharge Exposure, Mwd/Ton	*19,000*	*19,000*	*18,350*	*19,000*	*19,000*
Fuel Assembly					
Number of Fuel Assemblies	*368*	*484*	*368*	*548*	*764*
Fuel Rod Array	*7 x 7*	*7 x 7*	*7 x 7*	*7 x 7*	*7 x 7*

TABLE I-6-1 (CONTINUED)

CORE MECHANICAL DESIGN (CONTINUED)	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Fuel Assembly (Continued)					
Overall Dimensions, Inches	*175.88*	*174.88*	*175.88*	*175.88*	*175.88*
Weight of UO_2 per Assembly, Pounds	*Undished- 487.4*	*Undished- 492.5 Dished- 481.7*	*Undished- 490.35 Dished- 483.42*	*487.4*	*487.4*
Weight of Fuel Assembly, Pounds	*Undished- 682*	*Undished- 678.9 Dished- 668*	*Undished- 681.48 Dished- 674.55*	*682*	*682*
Fuel Rods					
Number per Fuel Assembly	*49*	*49*	*49*	*49*	*49*
Outside Diameter, Inch	*0.562*	*0.563*	*0.562*	*0.562*	*0.562*
Clad Thickness, Inch	*0.032*	*0.032*	*0.032*	*0.032*	*0.032*
Gap - Pellet to Clad, Inch	*0.005*	*0.005*	*0.005*	*0.005*	*0.005*
Length of Gas Plenum, Inches	*16*	*11.24*	*16*	*16*	*16*
Clad Material	*Zircaloy-2*	*Zircaloy-2*	*Zircaloy-2*	*Zircaloy-2*	*Zircaloy-2*
Cladding Process	*Free standing Loaded tubes*	*Free standing Loaded tubes*	*Free standing Loaded tubes*	*Free standing Loaded tubes*	*Free standing Loaded tubes*
Fuel Pellets					
Material	*Uranium Dioxide*	*Uranium Dioxide*	*Uranium Dioxide*	*Uranium Dioxide*	*Uranium Dioxide*
Density, % of Theoretical	*93%*	*93%*	*93%*	*93%*	*93%*

TABLE I-6-1 (CONTINUED)

CORE MECHANICAL DESIGN (CONTINUED)	VERMONT YANKEE	MONTICELLO	DUANE ARNOLD ENERGY CENTER	COOPER STATION	BROWNS FERRY UNITS 1 & 2
Fuel Pellets (Continued)					
Diameter, Inch	0.488	0.488	0.488	0.488	0.488
Length, Inch	0.5	0.5	0.5	0.5	0.5
Fuel Channel					
Overall Dimension, Inches (Length)	166.875	166.875	166.875	166.875	166.875
Thickness, Inch	0.085	0.080	0.085	0.085	0.085
Cross Section Dimensions, Inches	5.448 x 5.448	5.438 x 5.438	5.438 x 5.438	5.448 x 5.448	5.448 x 5.448
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly					
Fuel Weight as UO_2, Pounds	179,370	238,370	179,298	267,095	372,373
Zirconium Weight, Pounds (Z-2 + Z-4 + Spacers)	~63,300	~80,993	~63,300	~94,305	~131,476
Core Diameter (Equivalent), Inches	129.9	149	129.9	158.5	178.1
Core Height (Active Fuel), Inches	144	144	144	144	144
Reactor Control System					
Method of Variation of Reactor Power	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping	Movable Control Rods & Variable Coolant Pumping

TABLE I-6-1 (CONTINUED)

CORE MECHANICAL DESIGN (CONTINUED)	VERMONT YANKEE	MONTICELLO	DUANE ARNOLD ENERGY CENTER	COOPER STATION	BROWNS FERRY UNITS 1 & 2
Reactor Control System (Continued)					
Number of Movable Control Rods	89	121	89	137	185
Shape of Movable Control Rods	Cruciform	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of Movable Control Rods, Inches	12.0	12.0	12.0	12.0	12.0
Control Material in Movable Rods	B4C Granules Compacted in SS Tubes	B4C Granules Compacted in SS Tubes	B4C Granules Compacted in SS Tubes	B4C Granules Compacted in SS Tubes	B4C Granules Compacted in SS Tubes
Type of Control Rod Drives	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston	Bottom entry, Locking Piston
Number of Temporary Control Curtains	156	216	None	None	None
Curtain Material	Flat, boron -- stainless steel	Flat, boron -- stainless steel			
In-Core Neutron Instrumentation					
Number of In-Core Neutron Detectors	80	96	80	136	172
Number of In-Core Detector Strings	20	24	20	31	43
Number of Detectors per String	4	4	4	4	4
Number of Flux Mapping Neutron Detectors	3	3	3	4	5
Range (and Number) of Detectors					
Source Range Monitor	Source to .01% power (4)	Source to 10% power (4)	Source to .001% power (4)	Source to 10% power (4)	Source to 10% power (4)
Intermediate Range Monitor	.0001% to 10% power (6)	1% to 10% power (8)	.0001% to 10% power (6)	1% to 10% power (8)	1% to 10% power (8)
Local Power Range Monitor	5% to 125% power (80)	5% to 125% power (96)	5% to 125% power (80)	5% to 125% power (124)	5% to 125% power (172)
Average Power Range Monitor	5% to 125% power (6)	5% to 125% power (6)	2.5% to 125% power (6)	5% to 125% power (6)	5% to 125% power (6)
Number and Type of In-Core Neutron Sources	8 Sb-Be	5 Sb-Be	4 Sb-Be	5 Sb-Be	7 Sb-Be

TABLE I-6-1 (CONTINUED)

REACTOR VESSEL DESIGN	VERMONT YANKEE	MONTICELLO	DUANE ARNOLD ENERGY CENTER	COOPER STATION	BROWNS FERRY UNITS 1 & 2
Material	Carbon Steel/Clad Stainless Steel (ASME SA-336 & SA-302B)				
Design Pressure, Psia	1265	1265	1265	1265	1265
Design Temperature, °F	575	575	575	575	575
Inside Diameter, Ft-In.	17 - 1	17 - 2	15 - 3	18 - 2	20 - 11
Inside Height, Ft-In.	63 - 1.5	63 - 2	66 - 4	69 - 4	72 - 0
Side Thickness (Including Clad)	5.187	5.187	4.593 *3	5.531	6.313
Minimum Clad Thickness, Inches	1/8	1/8	1/8	1/8	1/8
Reactor Coolant Recirculation Design					
Number of Recirculation Loops	2	2	2	2	2
Design Pressure					
Inlet Leg, Psig	1175	1148	1148	1148	1148
Outlet Leg, Psig	1274	1248	1268	1274	1326
Design Temperature, °F	562	562	562	562	562
Pipe Diameter, Inches	28	28	22	28	28
Pipe Material	304/316	304	304/316	304/316	304/316
Recirculation Pump Flow Rate	32,500 gpm	32,500 gpm	27,100 gpm	45,200 gpm	45,000 gpm
Number of Jet Pumps in Reactor	20	20	16	20	20
Main Steam Lines					
Number of Steam Lines	4	4	4	4	4
Design Pressure, Psig	1146	1146	1146	1146	1146
Design Temperature, °F	563	563	563	563	563
Pipe Diameter, Inches	20	18	20	24	26
Pipe Material	Carbon Steel (ASTM A155 KC70 or ASTM A106 Grade B)				

TABLE I-6-1 (CONTINUED)

CORE STANDBY COOLING SYSTEMS	VERMONT YANKEE	MONTICELLO	DUANE ARNOLD ENERGY CENTER	COOPER STATION	BROWNS FERRY UNITS 1 & 2
(These systems are sized on maximum anticipated thermal output)					
Core Spray System					
Number of Loops	2	2	2	2	2
Flow Rate (Gpm)	3000 at 136 psid	3020 at 307 psid	3020 at 127 psid	4500 at 115 psid	6250 at 122 psid
High Pressure Coolant Injection System (No.)	1	1	1	1	1
Number of Loops	1	1	1	1	1
Flow Rate (Gpm)	4250	3000	2980	4220	5000
Automatic Depressurization System (No.)	1	1	1	1	1
Low Pressure Coolant Injection (No.)	1	1	1	1	1
Number of Pumps	4	4	4	4	4
Flow Rate (Gpm/Pump)	7000 at 20 psid	4000 at 20 psid	4800 at 20 psid	7700 at 20 psid	10,000 at 20 psid
Residual Heat Removal System					
Reactor Shutdown Cooling (Number of Pumps)	4	4	4	4	4
Flow Rate (Gpm/Pump)(1)	7000	3600	4800	7700	10,000
Capacity (Btu/Hr/Heat Exchanger)(2)	57.5×10^6	57.5×10^6	35×10^6	70×10^6	70×10^6
Number of Heat Exchangers	2	2	2	2	4
Primary Containment Cooling Flow Rate (Gpm)	28,000	16,000	19,200	30,800	40,000

(1) Capacity during reactor flooding mode with 3 of 4 pumps operating.

(2) Capacity during post-accident cooling mode with 165 °F shell side inlet temperature, maximum service water temperature, and one RHR pump and two service water pumps in operation.

TABLE I-6-1 (CONTINUED)

AUXILIARY SYSTEMS	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Service Water System					
Flow Rate (Gpm/Pump)	*2700*	*3500*	*2500*	*4000*	*4500*
No. of Pumps	*4*	*4*	*4*	*4*	*4*
Reactor Core Isolation Cooling System					
Flow Rate (Gpm)	*400*	*400*	*416*	*416*	*616*
Fuel Pool Cooling and Cleanup System					
Capacity (Btu/Hr.)	*NA*	2.87×10^6	2.37×10^6	3.4×10^6	8.8×10^6

TABLE I-6-2

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

TURBINE-GENERATOR	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Design Power, Mwt	*1665*	*1670*	*1670*	*2487*	*3440*
Design Power, Mwe	*564*	*545*	*597*	*836*	*1152*
Generator Speed, RPM	*1800*	*1800*	*1800*	*1800*	*1800*
Design Steam Flow, lb/hr	6.423×10^6	6.77×10^6	6.894×10^6	10.049×10^6	14.049×10^6
Turbine Inlet Pressure, Psig	*950*	*950*	*950*	*970*	*965*
TURBINE BYPASS SYSTEM					
Capacity, percent of turbine design steam flow	*100*	*15*	*25*	*25*	*25*
MAIN CONDENSER					
Heat removal capacity, Btu/hr	3500×10^6	3760×10^6	3681×10^6	5367.6×10^6	7770×10^6
CIRCULATING WATER SYSTEM					
Number of Pumps	*3*	*2*	*2 or more*	*4*	*3*
Flow Rate gpm/pump	*117,000*	*140,000*	*130,000 or less*	*162,500*	*200,000*
CONDENSATE AND FEEDWATER SYSTEMS					
Design Flow Rate, lb/hr	6.4×10^6	6.77×10^6	7.143×10^6	*9,773,000*	*13,999,000*
Number Condensate Pumps	*2*	*2*	*2*	*3*	*3*
Number Feedwater Pumps	*2*	*2*	*2*	*2*	*3*
Condensate Pump Drive	*a-c power*	*a-c power*	*a-c power*	*a-c power*	*a-c power*
Feedwater Pump Drive	*a-c power*	*a-c power*	*a-c power*	*turbine*	*turbine*

TABLE I-6-3

COMPARISON OF ELECTRICAL POWER SYSTEMS DESIGN CHARACTERISTICS

TRANSMISSION SYSTEM	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Outgoing lines (number-rating)	*2-345 kV*	*2-345 kV*	*2-345 kV*	*4-345 kV*	*4-500 kV*
NORMAL AUXILIARY A-C POWER					
Incoming lines (number-rating)	*2-345 kV* *1-230 kV* *1-115 kV* *1-4160 kV*	*2-345 kV* *2-230 kV* *1-115 kV*	*2-345 kV* *3-161 kV*	*1-161 kV* *1-69 kV*	*2-161 kV*
Auxiliary transformers	*1*	*1*	*2*	*1*	*2*
Start-up transformers	*1*	*1*	*1*	*1*	*2*
Shutdown transformers	-	-	-	*1*	-
STANDBY A-C POWER SUPPLY					
Number diesel generators	*2*	*2*	*2*	*2*	*3 of 4*
Number of 4160 V standby buses	*2*	*2*	*2*	*2*	*4*
Number of 480 V standby buses	*3*	*3*	*3*	*3*	*5*
D-C POWER SUPPLY					
Number of 125 V or 250 V batteries	*2*	*3*	*2*	*2*	*2*
Number of 125 V or 250 V buses	*4*	*3*	*2*	*4*	*4*

TABLE I-6-4

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

*PRIMARY CONTAINMENT**	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Type	*Pressure Suppression*	*Pressure Suppression*	*Pressure Suppression*	*Pressure Suppression*	*Pressure Suppression*
Construction					
Drywell	*Light bulb shape; steel vessel*	*Light bulb shape; steel vessel*	*Light bulb shape; steel vessel*	*Light bulb shape; steel vessel*	*Light bulb shape; steel vessel*
Pressure Suppression Chamber	*Torus; steel vessel*	*Torus; steel vessel*	*Torus; steel vessel*	*Torus; steel vessel*	*Torus; steel vessel*
Pressure Suppression Chamber - Internal Design Pressure (psig)	*+56*	*+56*	*+56*	*+56*	*+56*
Pressure Suppression Chamber - External Design Pressure (psi)	*+2*	*+2*	*+2*	*+2*	*+1*
Drywell - Internal Design Pressure (psig)	*+56*	*+56*	*+56*	*+56*	*+56*
Drywell Free Volume (ft^3)	*134,00*	*134,200*	*130,930*	*145,430*	*159,000*
Pressure Suppression Chamber Free Volume (ft^3)	*99,000*	*98,280*	*94,630*	*109,810*	*119,000*
Pressure Suppression Pool Water Volume (ft^3)	*78,000*	*68,000*	*61,500*	*87,660*	*85,000*
Submergence of Vent Pipe Below Pressure Pool Surface (ft)	*+4*	*+4*	*+4*	*+4*	*+4*
Design Temperature of Drywell (°F)	*281*	*281*	*281*	*281*	*281*
Design Temperature of Pressure Suppression Chamber (°F)	*281*	*281*	*281*	*281*	*281*
Downcomer Vent Pressure Loss Factor	*6.21*	*6.21*	*6.21*	*6.21*	*6.21*

**Where applicable, containment parameters are based on maximum anticipated power output.*

TABLE I-6-4 (CONTINUED)

PRIMARY CONTAINMENT (CONTINUED)	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Break Area/Total Vent Area	*0.019*	*0.019*	*0.019*	*0.019*	*0.019*
Calculated Maximum Pressure After Blowdown Drywell (psig)	*35*	*41*	*45*	*46*	*40*
Pressure Suppression Chamber (psig)	*22*	*26*	*29*	*28*	*25*
Initial Pressure Suppression Chamber Temperature Rise (°F)	*35*	*50*	*50*	*50*	*32*
Leakage Rate (% Free Volume/Day at 56 psig and 281 °F)	*0.5*	*0.5*	*0.5*	*0.5*	*0.5*
SECONDARY CONTAINMENT					
Type	*Controlled Leakage Elevated Release*	*Controlled Leakage Elevated Release*	*Controlled Leakage Elevated Release*	*Controlled Leakage Elevated Release*	*Controlled Leakage Elevated Release*
Construction					
Lower Levels	*Reinforced Concrete*	*Reinforced Concrete*	*Reinforced Concrete*	*Reinforced Concrete*	*Reinforced Concrete*
Upper Levels	*Steel Super-structure & siding*	*Steel Super-structure & siding*	*Steel Super-structure & siding*	*Steel Super-structure & siding*	*Steel Super-structure & siding*
Roof	*Steel Sheeting*	*Steel Sheeting*	*Steel Sheeting*	*Steel Sheeting*	*Steel Sheeting*
Internal Design Pressure (psig)	*0.25*	*0.25*	*0.25*	*0.25*	*0.25*
Design Inleakage Rate (% free Volume/Day at 0.25 inches H_2O)	*100*	*100*	*100*	*100*	*100*
ELEVATED RELEASE POINT					
Type	*Stack*	*Stack*	*Stack*	*Stack*	*Stack*
Construction	*Reinforced concrete*	*Reinforced concrete*	*To be determined*	*Steel*	*Steel*
Height (above ground)	*318 feet*	*89 meters*	*100 meters*	*100 meters*	*200 meters*

TABLE I-6-5

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

SEISMIC DESIGN	*VERMONT YANKEE*	*MONTICELLO*	*DUANE ARNOLD ENERGY CENTER*	*COOPER STATION*	*BROWNS FERRY UNITS 1 & 2*
Maximum Design (horizontal g)	*0.07*	*0.06*	*0.06*[(1)]	*0.10*	*0.10*
Maximum Hypothetical Earthquake	*0.14*	*0.12*	*0.12*[(1)]	*0.20*	*0.20*
WIND DESIGN					
Maximum Sustained (mph)	*80*	*100*	*105*	*100*	*100*
Tornados (mph)	*300*	*300*	*300*	*300*	*300*

[(1)] *On rock.*

7.0 STATION RESEARCH DEVELOPMENT AND FURTHER INFORMATION; REQUIREMENTS AND RESOLUTIONS SUMMARY

The design of the boiling water reactor for this station was based upon proven technological concepts developed during the development, design, and operation of numerous similar reactors. The AEC, in reviewing the Browns Ferry Nuclear Plant and Cooper Nuclear Station dockets at the construction permit stage, identified several areas where further research and development efforts were required to more definitely assure safe operation of this station. Also, both the AEC Staff and the Advisory Committee on Reactor Safeguards (ACRS) had in their reviews of this and other recent reactor projects, identified several additional technical areas for which further detailed support information should be obtained. All of these development efforts were of three general types: (a) Those which pertained to the broad category of water-cooled reactors, (b) those which pertained specifically to boiling water reactors, and (c) those which have been noted particularly for a facility during the construction permit licensing activities by the AEC Staff and ACRS reviews.

Appendix H of this SAR provides a comprehensive examination and discussion of each of these concern areas, indicating resolution. A summary conclusion of the analysis is provided in this section in Tables I-7-1 through I-7-4. The tables cover the following areas of concern:

a. Areas specified in the Cooper Nuclear Station ACRS construction permit letter. (Refer to Table I-7-2).

b. Areas specified in the Cooper Nuclear Station AEC Staff construction permit safety evaluation. (Refer to Table I-7-3).

c. Areas specified in other related AEC-ACRS construction and operating permit letters. (Refer to Table I-7-4).

The scope of many of the areas of technology for items in a, b, and c above is discussed in Appendix H in detail as part of an official response[1] by the General Electric Company to the various concerns expressed by the ACRS.

The General Electric Company had submitted many topical reports to the AEC in support of this application and those of other GE-BWR facilities. (Refer to Table I-7-1).

TABLE I-7-1

COOPER NUCLEAR STATION TOPICAL REPORTS SUBMITTED TO THE AEC IN SUPPORT OF DOCKET

The information contained in this table is designated as historical as indicated by the italicized text. It represents a listing of Topical Reports which were submitted in support of the original Licensing process. No attempts are made to update this table to reflect Topical Reports submitted since receipt of the CNS Operating License. USAR Section I-3.4 provides a more detailed discussion of the purpose of highlighting certain text with Italics.

	GE Report No.		Title
1.	*APED 5286*		*Design Basis for Critical Heat Flux Condition in Boiling Water Reactors (September, 1966)*
2.	*APED 5446*	-	*Control Rod Velocity Limiter (March, 1967)*
3.	*APED 5449*	-	*Control Rod Worth Minimizer (March, 1967)*
4.	*APED 5450*	-	*Design Provisions for In-Service Inspection (April, 1967)*
5.	*APED 5453*	-	*Vibration Analysis and Testing of Reactor Internals (April, 1967)*
6.	*APED 5555*	-	*Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November, 1967)*
7.	*TR67SL211*	-	*An Analysis of Turbine Missiles Resulting from Last Stage Wheel Failure (October, 1967)*
8.	*APED 5608*	-	*General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns (April, 1968)*
9.	*APED 5455*	-	*The Mechanical Effects of Reactivity Transients (January, 1968)*
10.	*APED 5528*	-	*Nuclear Excursion Technology (August, 1967)*
11.	*APED 5448*	-	*Analysis Methods of Hypothetical Super-Prompt Critical Reactivity Transients in Large Power Reactors (April, 1968)*
12.	*APED 5458*	-	*Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March, 1968)*
13.	*APED 5640*	-	*Xenon Considerations in Design of Large Boiling Water Reactors (June, 1968)*
14.	*APED 5454*	-	*Metal Water Reactions - Effects on Core Cooling and Containment (March, 1968)*
15.	*APED 5460*	-	*Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September, 1968)*
16.	*APED 5654*	-	*Considerations Pertaining to Containment Inerting (August, 1968)*

TABLE I-7-1 (CONTINUED)

	GE Report No.		*Title*
17.	*APED 5696*		*Tornado Protection for the Spent Fuel Storage Pool (November, 1968)*
18.	*APED 5706*	-	*In-Core Neutron Monitoring System for General Electric Boiling Water Reactors, Rev. 1 (April, 1969)*
19.	*APED 5703*	-	*Design and Analysis of Control Rod Drive Reactor Vessel Penetrations (November, 1968)*
20.	*APED 5698*	-	*Summary of Results Obtained From a Typical Startup and Power Test Program for a General Electric Boiling Water Reactor (February, 1969)*
21.	*APED 5750*	-	*Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March, 1969)*
22.	*APED 5756*	-	*Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor (March, 1969)*
23.	*APED 5652*		*Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April, 1969)*
24.	*APED 5736*	-	*Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April, 1969)*
25.	*APED 5447*	-	*Depressurization Performance of the General Electric Boiling Water Reactor High Pressure Coolant Injection System (June, 1969)*
26.	*NEDO 10017*	-	*Field Testing Requirements for Fuel, Curtains, and Control Rods (June, 1969)*
27.	*NEDO 10029*	-	*An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident (July, 1969)*
28.	*NEDO 10045*		*Consequences of a Steam Line Break in a General Electric Boiling Water Reactor (July, 1969)*
29.	*NEDO 10139*		*Compliance of Protection Systems to Industry Criteria; GE BWR NSSS (June, 1970)*
30.	*NEDO 10173*		*Current State of Knowledge High Performance BWR Zircaloy-clad UO_2 Fuel (May, 1970)*
31.	*NEDO 10179*	-	*Effects of Cladding Temperature and Material on ECCS Performance (June, 1970)*
32.	*NEDO 10174*	-	*Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor (May, 1970)*

TABLE I-7-1 (CONTINUED)

	GE Report No.		Title
33.	NEDO 10189	-	An Analysis of Functional Common-Mode Failures in GE BWR Protection and Control Instrumentation (July, 1970)
34.	NEDO 10208	-	Effects of Fuel Rod Failure on ECCS Performance (August, 1970)

TABLE I-7-2

COOPER NUCLEAR STATION ACRS CONCERNS - RESOLUTION

Appendix H Section No.	ACRS Concern	Cooper Resolution
H-2.2	Station Foundation Support	USAR (Incorporated in Design – Chapter II)
H-2.3	Emergency On-Site Power System	USAR (Incorporated in Design – Chapter VIII)
H-2.4	AEC General Design Criteria No. 35 Intent Design	USAR (Incorporated in Design – Chapter IV, Appendix A and Appendix F)
H-2.5.1	Effects of Fuel Failure on ECCS Performance	Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10208)
H-2.5.2	Effects of Cladding Temperature and Materials on ECCS Performance	Topical Report (GE-APED-5608) Topical Report (GE-APED-5458) Topical Report (GE-NEDO-10179)
H-2.5.3	Control Systems for Emergency Power	USAR (Incorporated in Design – Chapters VI, VII, and VIII)
H-2.5.4	Diversification of ECCS Initiation Signals	USAR (Incorporated in Design – Chapters VI and VII)
H-2.5.5	Main Steam Line Isolation Valve Testing Under Simulated Accident Conditions	USAR (Incorporated in Design – Chapters IV) Topical Report (GE-APED-5750) Topical Report (GE-NEDO-10045) Topical Report (GE-APED-5608)
H-2.5.6	Misorientation of Fuel Assemblies	USAR (Incorporated in Design – Chapter III)
H-2.5.7	Effects of Fuel Bundle Flow Blockage	Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10174)
H-2.5.8	Verification of Fuel Damage Limit Criteria	Topical Report (GE-APED-5608) Dresden 2/3 – Amendment 14/15 Topical Report (GE-NEDO-10173)
H-2.5.9	Control Rod Block Monitor Design	USAR (Incorporated in Design – Chapter VII) Dresden 2/3 – Amendments 17/18 and 19/20, Brunswick 1/2 – Supplement 5

TABLE I-7-2 (CONTINUED)

Appendix H Section No.	ACRS Concern	Cooper Resolution
H-2.5.10	Quality Assurance and Inspection of the Reactor Primary System	USAR (Incorporated in Design – Chapter IV, Appendix D and Appendix J)
H-2.5.11	Formulation of an In-Service Inspection Program	USAR (Incorporated in Design – Appendix J)
H-2.5.12	Station Start Up Program	Topical Report (GE-APED-5698) USAR (Incorporated in Design – Chapter XIII)

TABLE I-7-3

COOPER NUCLEAR STATION AEC STAFF CONCERNS - RESOLUTIONS

Appendix H Section No.	AEC Concern	Cooper Resolution
H-3.3.1	Linear Heat Generation Rate Fuel Damage Limit	Topical Report (GE-APED-5608) Dresden 2/3 - Amendment 14/15 Topical Report - (GE-NEDO-10173)
H-3.3.2	Local Fuel Melting Resulting from Inlet Coolant Orifice Blockage	Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10174)
H-3.3.3	Effect of Fuel Clad Failure on Emergency Core Cooling	Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10208)
H-3.4.1	Core Spray Effectiveness	Topical Report (GE-APED-5608) Topical Report (GE-APED-5458) Topical Report (GE-NEDO-10179)
H-3.4.2	Steam Line Isolation Valve Testing	Topical Report (GE-APED-5608) Topical Report (GE-APED-5750) Topical Report (GE-NEDO-10045)
H-3.4.3	Adequacy of HPCI System as a Depressurizer	USAR (Incorporated in Design – Chapter VI) Topical Report (GE-APED-5608) Topical Report (GE-APED-5447)
H-3.4.4	Electrical Equipment Inside Containment	USAR (Incorporated in Design – Chapters V and VII)
H-3.4.5	Control Rod Worth Minimizer	Topical Report (GE-APED-5449) USAR (Incorporated in Design – Chapter VII)
H-3.4.6	Jet Pump Development	Topical Report (GE-APED-5460)
H-3.4.7	Rod Velocity Limiter	Topical Report (GE-APED-5446) USAR (Incorporated in Design – Chapter III)
H-3.4.8	In-Core Neutron Monitor System	Topical Report (GE-APED-5456) Topical Report (GE-APED-5706) USAR (Incorporated in Design – Chapter VII)
H-3.5.1	Failure of Passive Components of ECCS	USAR (Incorporated in Design – Chapters IV and XII)

TABLE I-7-3 (CONTINUED)

Appendix H Section No.	AEC Concern	Cooper Resolution
H-3.5.2	Thermal Shock	Topical Report (GE-NEDO-10029) USAR (Incorporated in Design – Chapters III and IV)
H-3.5.3	Interchannel Flow Stability	USAR (Incorporated in Design – Chapters III and VII) Topical Report (GE-APED-5652) Topical Report (GE-APED-5640) Peach Bottom 2/3 – Amendment 2
H-3.5.4	In-Service Inspection	USAR (Incorporated in Design – Appendix J)
H-3.5.5	Primary System Leak Detection	USAR (Incorporated in Design – Chapter IV)

TABLE I-7-4

AEC-ACRS CONCERNS ON OTHER RELATED DOCKETS
COOPER CAPABILITY FOR RESOLUTION

Appendix H Section No.	AEC-ACRS Concerns	Cooper Resolution
H-4.2	LPCIS-Logic Control System Design	The LPCI Loop Select Logic was removed by DC 76-2. (See USAR Chapter VII)
H-4.3	Re-Evaluation of Main Steam Line Break Accident	Topical Report (GE-APED-5608) Topical Report (NEDO-10045) USAR (Incorporated in Design – Chapter XIV)
H-4.4	Design of Piping Systems to withstand Earthquake Forces	USAR (Incorporated in Design – Chapter XII and Appendices A and C) Dresden 2/3 – Amendment 13/14
H-4.5	Fuel Clad Disintegration Limitations	USAR (Incorporated in Design – Chapter VI) Topical Report (GE-APED-5608) Topical Report (GE-NEDO-10179) Pilgrim Amendment 14
H-4.6	Automatic Pressure Relief System - Initiation Interlock	USAR (Incorporated in Design – Chapters VI and VII)
H-4.7	Effects of Blowdown Forces on Reactor Primary System Components	USAR (Incorporated in Design – Chapters III and IV, and Appendix C)
H-4.8	Separation of Control and Protection System Functions	USAR (Incorporated in Design – Chapters VI and VII and Appendix F)
H-4.9	Instrumentation for Prompt Detection of Gross Fuel Failures	USAR (Incorporated in Design – Chapters VII and XIV) Brunswick 1/2 – Supplements 3 and 4 (The requirement for scram and containment isolation functions of the MSIV Radiation Monitors was removed by Amendment 158)
H-4.10	Scram Reliability Study	USAR (Incorporated in Design – Chapters III and VII) Topical Report (GE-NEDO-10189) Brunswick 1/2 – Supplement 6
H-4.11	Design Basis of Engineered Safety Features	Topical Report (GE-APED-5756) USAR (Examined Capability of Design – Chapter XIV)

TABLE I-7-4 (CONTINUED)

Appendix H Section No.	AEC-ACRS Concerns	Cooper Resolution
H-4.12	Hydrogen Generation Study	USAR (Incorporated in Design – Chapter V)
H-4.13	Seismic Design and Analysis Models	USAR (Re-Confirmation of Design – Chapter XII and Appendix C) Dresden 2 – Re-Confirmation Information (Submitted October, 1969)
H-4.14	Automatic Pressure Relief System - Single Component Failure Capability - Manual Operation	USAR (Incorporated in Design – Chapters VI, VII and VIII) Topical Report (GE-NEDO-10139)
H-4.15	Flow Reference Scram	USAR (Incorporated in Design – Chapter VII)
H-4.16	Matters of Current Regulatory Staff - Applicant Discussion	
	a) Standby Gas Treatment System Electrical and Physical Separation	USAR (Incorporated in Design – Chapters IV, VII, and VIII)
	b) Official, Issued Technical Specifications - License Appendix A	Technical Specifications
H-4.17	Future Items of Consideration for Incorporation....	
	a) Radiolytic Decomposition of Cooling Water	Topical Report (GE-APED-5454) Topical Report (GE-APED-5654) Brunswick 1/2, Supplement 4 Dresden 3, Amendment 23
	b) Development of Instrumentation - Vibration and Loose Parts Detection	USAR (Justified Design – Chapters III, IV, and Appendix C)
	c) Consequences of Water Contamination - Structural Material - LOCA	USAR (Incorporated in Design – Chapter XIV)

TABLE I-7-4 (CONTINUED)

Appendix H Section No.	AEC-ACRS Concerns	Cooper Resolution
H-4.18	Development of Instrumentation - Primary Containment Leakage Detection Increased Sensitivity Studies	USAR (Justified Design – Chapter IV) Technical Specifications - (Chapters III and IV)
H-4.19	Development of Instrumentation - Vibration and Loose Parts Detection Studies	USAR (Justified Design Chapter III, IV, and Appendix C)
H-4.20	ECCS - Leakage Detection, Protection and Isolation Capability	USAR (Justified in Design – Chapters IV, X, and Appendix F) Brunswick 1/2 – Supplement 4, C/R 6.4
H-4.21	Main Steam Lines - Standards for Fabrication, Q/C, and Inspection	USAR (Chapter IV, Appendices D and H)
H-4.22	Primary Containment Inerting	USAR (Incorporated in Design – Chapters V and VI)

8.0 REFERENCES FOR CHAPTER I

1. Bray, A. P., et al., APED-5608, "The General Electric Company Analytical and Experimental Program for Resolution of ACRS Safety Concerns", April, 1968.

2. Cooper Nuclear Station License Amendment 231, dated June 30, 2008, Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate.