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CHAPTER 1.0

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CHAPTER 1.0

INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

Kansas City Power & Light Company, Kansas Gas and Electric Company and Union Electric Company joined together to design, purchase, and license a nuclear block for a generating station acceptable at any of several sites, under the acronym of SNUPPS*, Standardized Nuclear Unit Power Plant System.

1.1.1 LICENSES REQUESTED AND RECEIVED

The Final Safety Analysis Report (FSAR) was submitted to the NRC in support of the application by Union Electric for a Class 103 license to operate a nuclear power facility. The FSAR was originally submitted in two parts, the SNUPPS Standard Plant and the Callaway Site Addendum. Some of the chapters common to both reports are currently being combined into one report, the Callaway - SP. This report follows the format recommended by Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Sufficiently detailed design information is provided in this report to make a definitive evaluation that the Callaway Plant can be operated without undue risk to the health and safety of the public. Union Electric received a low power (<5 percent) license to operate the Callaway Plant on June 11, 1984. The full power license was issued on October 18, 1984 and commercial operation began on April 9, 1985.

1.1.2 PLANT UNITS

The power block was built to the SNUPPS duplicate plant design. The Callaway Plant power block, consists of these structures, including enclosed systems and components:

a. Reactor building (containment)

b. Turbine building

c. Control building

d. Auxilliary building

e. Diesel generator building

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* Table 1.1-1 provides a list of the acronyms used in this document and their definitions.

f. Fuel building

g. Radwaste building

h. Storage tanks (refueling water, condensate, demineralized water, reactor makeup water, and emergency fuel oil)

i. Transformers (main, unit auxiliary, ESF, and station service) and vaults

Due to the use of the SNUPPS standard design for these items, design envelopes were developed by use of the most restrictive site conditions imposed by any one of the four original sites or by generic design criteria which are conservative for each of the sites. With the cancellation of the Tyrone plant, however, the four-site enveloping approach was modified in the seismic design area (e.g., development of spectra) to include only the three remaining sites. Refer to Sections 2.5 and 3.7(B) for details. The design envelopes were not revised to reflect the cancellation of Sterling.

1.1.3 PLANT LOCATION

The site of the Callaway Plant is 10 miles southeast of the city of Fulton in Callaway County, Missouri, and 80 miles west of the St. Louis metropolitan area. The nearest population center is Jefferson City, Missouri, located 25 miles west-southwest of the site. The plant site, consisting of approximately 2,767 acres of rural land, is located on a high plateau approximately 300 feet above the Missouri River, which is about five miles to the south.

1.1.4 CONTAINMENT STRUCTURE

The containment, which was designed by the Bechtel Power Corporation, is a carbon steel-lined, concrete structure. The walls and dome are post-tensioned, prestressed concrete, and the base slab is reinforced concrete.

1.1.5 NUCLEAR STEAM SUPPLY SYSTEM AND TURBINE-GENERATOR

The nuclear steam supply system (NSSS) for Callaway is a pressurized water reactor (PWR) which was designed and supplied by the Westinghouse Electric Corporation.

The reactor core is designed for an output of 3,565 MWt. When the reactor coolant pump input of 14 MWt is added to the core output, the warranted nuclear steam supply system output is 3,579 MWt, which is defined as the rated power in the license application. The engineered safety features are designed for a core power of 3,565 MWt. An additional 2 percent conservatism is added for some analyses to give a maximum accident analysis power of 3,636 MWt. The turbine generator is rated for operation at the NSSS output of 3,579 MWt. The corresponding turbine generator valve wide open capability electrical output is 1,284 MWe. The turbine generator was designed and supplied by the General Electric Company. and the high and low pressure turbnes were retrofit by Alstom.

1.1.6 SCHEDULE FOR FUEL LOADING AND OPERATION

On June 11, 1984 Union Electric received a low-power (5%) license to operate the Callaway Plant with initial criticality being achieved on October 2, 1984. The full-power license was issued to Union Electric on October 18, 1984 and commercial operation began on April 9, 1985.

1.1.7 DESIGN BASES

As used within this FSAR, the design bases are a list of requirements that the system must meet in order to:

a. Perform directly a specified safety or power generation function including support of another function (e.g., provide cooling water flow for other components, maintain a given compartment temperature).

b. Comply with a regulatory or statutory requirement or guideline (e.g., a jurisdictional building code).

c. Meet a specific operator interface, startup, or specific testing requirement.

d. Meet a design classification or code requirement (e.g., be designed to withstand the safe shutdown earthquake). Items implicit in contemporary design practices (e.g., use of the English system of weights and measures or the exercise of good engineering practice) are not specified as design bases.

Safety design bases are engineering objectives which must be met by safety-related structures, systems, or components. Safety-related items are defined as those plant features necessary to ensure the following:

a. The integrity of the reactor coolant pressure boundary

b. The capability to shut down the reactor and maintain it in a safe shutdown condition

c. The capability to prevent or mitigate the consequences of accidents that could potentially result in offsite exposures approaching the guideline exposures of 10 CFR 100.

Items which are associated with safety-related equipment, but which in themselves are not absolutely essential to the safety function of the equipment, are not considered safety-related.

Power generation design bases support, either directly or indirectly, the major electrical power generation function of the station. Examples of power generation design bases are the requirements to provide adequate radiation shielding and domestic water for plant personnel.

Sections describing Westinghouse-supplied systems and components do not provide safety design bases or power generation design bases as such. These sections do give functional descriptions and are in compliance with Regulatory Guide 1.70.

TABLE 1.1-1 ACRONYMS USED IN THE FSAR

AC	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
A/E	Architect/Engineer
AFAS	Auxiliary Feedwater Actuation System
AFS	Auxiliary Feedwater System
AISC	American Institute of Steel Construction
ALARA	As Low as Reasonably Achievable
ANSI	American National Standards Institute
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
ARW	Chemical Waste
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transients Without Scram
AVT	All Volatile Treatment
AWS	American Welding Society
BOP	Balance of Plant
B&PVC	Boiler and Pressure Vessel Codes
BRS	Boron Recycle System
BTP	Branch Technical Position
CAS	Compressed Air System
CCS	Condensate Cleanup System
CCWS	Component Cooling Water System
CDS	Condensate Demineralizer System
CeCWS	Central Chilled Water System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater System
CGCS	Combustible Gas Control System
CHC	Cask Handling Crane
CHF	Critical Heat Flux
CIS	Containment Isolation Signal
ClCWS	Closed Cooling Water System
CLP	Cask Loading Pit
CM	Center of Mass
CMAA	Crane Manufacturing Association of America

TABLE 1.1-1 (Sheet 2)

CP	Construction Permit
CPR	Critical Power Ratio
CPIS	Containment Purge Isolation System/Signal
CR	Center of Rigidity
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Mechanism
CREA	Control Rod Ejection Accident
CRVIS	Control Room Ventilation Isolation System/Signal
CRW	Tritiated Waste
CSD	Cold Shutdown
CST	Condensate Storage Tank
CSTS	Condensate Storage and Transfer System
CtCS	Containment Cooling System
CVCS	Chemical and Volume Control System
CWP	Cask Washdown Pit
CWS	Circulating Water System
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DCSS	Dry Cask Storage System
DEPSG	Double Ended Pump Suction Guillotine
DG	Diesel Generator
DGB	Diesel Generator Building
DoWS	Domestic Water System
DNB	Departure From Nucleate Boiling
DNBR	Departure From Nucleate Boiling Ratio
DRW	Potentially Radioactive Nontritiated Waste
DWMS	Demineralized Water Make-up System
DWST	Demineralized Water Storage Tank
DWSTS	Demineralized Water Storage and Transfer System
DWT	Dead Weight Test
ECCS	Emergency Core Cooling System
EHC	Electrohydraulic Control
EOL	End of Life
EDECAIES	Emergency Diesel Engine Combustion Air Intake and Exhaust System
EDECWS	Emergency Diesel Engine Cooling Water System
EDEFSTS	Emergency Diesel Engine Fuel Oil Storage and Transfer System
EDELS	Emergency Diesel Engine Lubrication System

TABLE 1.1-1 (Sheet 3)

EDESS	Emergency Diesel Engine Start System
EFOST	Emergency Fuel Oil Storage Tank
ER	Environmental Report
ESFS	Engineered Safety Feature System
ESFAS	Engineered Safety Feature Actuation System
ESWS	Essential Service Water System
FBIS	Fuel Building Isolation Signal
FED	Floor and Equipment Drainage
FDDR	Field Deviation Disposition Request
FHA	Fuel Handling Accident
FHS	Fuel Handling System
FMEA	Failure Modes and Effects Analysis
FPCC	Fuel Pool Cooling and Cleanup
FPRCS	Fission Product Removal and Control System
FPS	Fire Protection System
FRS	Frequency Response Spectra
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GRWS	Gaseous Radwaste System
HCST	Hardened Condensate Storage Tank
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air (filter)
HERMIT	Holtec Earthquake Response Mitigator
HEX	Heat Exchanger
Hga	Inches of Mercury Absolute
HI-STORM	Holtec International Storage Module
HI-TRAC VW	HI-TRAC Variable Weight (VW) Transfer Cask
HSST	Heavy Section Steel Technology
HVAC	Heating, Ventilation and Air Conditioning
IAC	Interim Acceptance Criteria
IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated Leakage Rate Test
ISFSI	Independent Spent Fuel Storage Installation
ISI	Inservice Inspection
LCO	Limiting Condition of Operation
LEFM	Linear Elastic Fracture Mechanics
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor

TABLE 1.1-1 (Sheet 4)

LPZ	Low Population Zone
LRWS	Liquid Radwaste System
LRW	Potentially Radioactive Secondary Liquid Waste
LSP	Low Suction Pressure
MCARS	Main Condenser Air Removal System
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MFIV	Main Feedwater Isolation Valve
MG	Motor Generator Set
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPC	Multi-Purpose Canister
MS	Manufacturer's Standard
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MSLB	Main Steam Line Break
MSSS	Main Steam Supply System
NDT	Nondestructive Testing
NDTT	Nil-Ductility Transition Temperature
NFSF	New Fuel Storage Facility
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OL	Operating License
OPS	Offsite Power Systems
ORE	Occupational Radiation Exposures
OSGSF	Old Steam Generator Storage Facility
PA	Public Address
PAMS	Post-Accident Monitoring System
PCT	Peak Cladding Temperature
PHS	Plant Heating System
P&ID	Piping and Instrumentation Diagram
PLS	Precautions, Limitations, and Setpoints
PMF	Probable Maximum Flood
PRA	Peak Recording Accelograph
PRM	Process Radiation Monitoring
PSAR	Preliminary Safety Analysis Report

TABLE 1.1-1 (Sheet 5)

PSS	Process Sampling System
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pumps
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RMWCS	Reactor Makeup Water Control System
RMWS	Reactor Makeup Water System
RMWST	Reactor Makeup Water Storage Tank
RO	Reactor Operator
RPV	Reactor Pressure Vessel
RRS	Required Response Spectrum
RSG	Replacement Steam Generator
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SACF	Single Active Component Failure
SAR	Safety Analysis Report
SFSF	Spent Fuel Storage Facility
SGB	Steam Generator Blowdown
SGBIS	Steam Generator Blowdown Isolation System/Signal
SGBS	Steam Generator Blowdown System
SIS	Safety Injection Signal
SIT	Structural Integrity Test
SLWS	Secondary Liquid Waste System
SMA	Strong Motion Accelerometer
SNUPPS	Standard Nuclear Unit Power Plant System
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Solid Radwaste System
SRSS	Square Root of the Sum of the Squares
SRW	Detergent Waste
SSE	Safe Shutdown Earthquake
SWS	Service Water System
TBS	Turbine Bypass System
TG	Turbine Generator
TGSS	Turbine Gland Sealing System
TRS	Test Response Spectrum
UHS	Ultimate Heat Sink
USGS	U.S. Geological Survey

TABLE 1.1-1 (Sheet 6)

UT	Ultrasonic Testing
VECASP	Variable Elevation Cask Staging Pedestal
VVM	Vertical Ventilated Module
VWO	Valves Wide Open
W	Westinghouse

1.2 GENERAL PLANT DESCRIPTION

1.2.1 SITE CHARACTERISTICS

1.2.1.1 Location

The Callaway Plant site is in east-central Missouri in Callaway County. The site is 25 miles east-northeast of Jefferson City and approximately 5 miles north of the Missouri River. It is situated approximately 10 miles southeast of Fulton and 80 miles west of St. Louis. Site location is discussed in more detail in Section 2.1.1.

1.2.1.2 Site Ownership

The plant site area consists of approximately 2,767 acres. UE owns all of the area within the designated plant site boundary and will therefore exercise full ownership control of the site with full authority to determine all activities, including exclusion or removal of personnel and property from the area. Union Electric has complete ownership of the minerals on and under this area.

An additional 2,461 acres were acquired in UE's acquisition efforts prior to determination of the site boundaries to insure adequate coverage and to negotiate for part of the properties within the plant site area. These properties comprise the plant site peripheral area and will be a part of the Low Population Zone.

The plant corridor area, which extends from the plant site area south to Missouri Route 94, consists of approximately 2,002 acres. This corridor area will provide for road and water pipeline access to the site. UE may ultimately devest itself of property in this tract not needed in the development of the plant site or related activities.

1.2.1.3 Access to the Site

Access to the plant site by motor vehicle is available from three directions. From the south, access is via the east-west Missouri Route 94 and northbound county road 459. From Fulton and the west, entry is available by way of Route O to Route CC. Access to the site from the north is via east-west Interstate 70 and southbound Route D to Routes O and CC.

1.2.1.4 Site Environs

The plant site is situated in an area of low population. The land area within 5 miles of the site consists of approximately 60 percent forests, 20 percent farming, and 20 percent pastures. The plateau upon which the site is located covers approximately 8 square miles and varies in elevation from 800 feet near the perimeter to a maximum of 858 feet. (In the FSAR, all elevation references are on a mean sea level base). This upland plateau overlooks the Missouri above the Missouri River flood plain which has an elevation of 525 feet MSL adjacent to the site.

The area within 10 miles of the site is rural. The only incorporated communities within this 10 miles are Chamois, part of Fulton, and Mokane. The 1990 census populations for these communities are 546, 11,046, and 293 respectively. However, the total resident population within 10 miles of the reactor is only 8,996. The population center, or city nearest to the site with a population greater than 25,000 persons, is Jefferson City, Missouri, 25 miles west-southwest of the site. In 1990, Jefferson City recorded a population of 34,046 residents.

1.2.1.5 Geology

The Callaway Plant site is located within the Central Stable Region of North America. This vast region is characterized by a relatively gentle tectonic history since the beginning of Cambrian time. Most of the structural features in the region were formed during the Paleozoic era. The site is located along the northern flank of the Ozark Uplift at the southern edge of glaciation in North America. Unconsolidated soils blanket Paleozoic rocks.

The most influential tectonic structure within the regional study is the Mississippi Valley fault zone, over 200 miles from the site. The zone is located in the Mississippi Valley, in the Upper Mississippi Embayment, and includes the New Madrid seismotectonic region.

The site straddles the boundary between the Dissected Till Plains Physiographic Section to the North and the Ozark Plateau Physiographic Province to the south. During early Pleistocene time, the site area was part of a glacial till plain. Subsequent erosion and downcutting of the Missouri River and its tributary streams have dissected the plain, leaving a nearly isolated plateau between 6 and 8 square miles in size. Topographic relief varies more than 150 to 200 feet between valleys and ridges, and the overall drop in elevation between the crest of the plateau and the river is about 300 feet.

The gently rolling plateau surface surrounding the site is deeply dissected by the Missouri River, which lies about 5 miles to the south, and by the tributaries to the Missouri River, all of which have steep gradients. The elevation at the site is approximately 845 feet, and local relief is on the order of 320 to 330 feet.

Thirty to forty feet of glacial and post-glacial unconsolidated materials constitute the surficial deposits. These consist of 6 to 10 feet of modified loess overlying 10 to 15 feet of moderately to highly plastic gray lacustrine clay. A 5- to 15-foot thick layer of glacial till consisting of reddish-brown silty clay with some mixed sand and gravel underlies the lacustrine deposit. Approximately 2,000 feet of Paleozoic sedimentary rocks underlie the surficial deposits at the site.

Beneath the glacial till lies a hard gravelly clay -- referred to as the Graydon Chert Conglomerate -- composed of white, gray and reddish-brown chert fragments in a reddish-brown or white clay matrix with some lenses of alluvial and colluvial materials. This deposit, which averages 25 feet in thickness in the site area, is in part the product of weathering and formed in Pennsylvanian or late Mississippian time. The alluvial and colluvial materials accumulated prior to glaciation in Pleistocene time. The Burlington Formation of Middle Mississippian age, a coarse-grained, cherty, fossiliferous, light tan limestone, underlies the Graydon Chert Conglomerate and is 0 to 41 feet thick in the site area. The Bushberg Formation, Lower Mississippian in age, consists of 1 to 6 feet of fine- to medium-grained, green to yellowish brown, argillaceous sandstone and underlies the Burlington. The Upper Devonian Snyder Creek Formation underlies the Bushberg and is approximately 30 feet thick in the area. It is a highly silty, calcitic, gray shale with some beds of light gray, very fine-grained, silty, argillaceous limestone. The Middle Devonian Callaway Formation underlies the Snyder Creek Shale. It is 40 feet thick and consists of fine- to coarse-grained, brownish-gray, fossiliferous limestone.

Below the Callaway Formation lies the Lower Ordovician Jefferson City Formation. It is several hundred feet thick and consists of fine- to medium-grained, light-colored dolomite with some shale, thin sandstone lenses, and oolitic or banded chert. Detailed exploration test borings in the plant site area revealed the presence of St. Peter sandstone filling two paleokarst features within the Jefferson City Formation. The larger is 130 feet thick while the smaller is 23 feet in thickness. The St. Peter sandstone is a friable, white, relatively pure sandstone. Several borings revealed 5 to 10 feet of the Middle Ordovician Joachim Formation directly over the paleokarst features. It is a buff-colored dolomite with minor beds of shale and sandstone.

Lower Ordovician and Upper Cambrian sandstones and dolomites underlie the Jefferson City Formation; the Cambrian units rest unconformably on the Precambrian basement rocks.

The site area is located along the northern flank of the Ozark Uplift. The Illinois Basin lies to the east in Illinois while the Forest City and Cherokee basins, separated by the Bourbon Arch, are located to the west in Missouri and Kansas. The sedimentary rocks in the site area display a regional dip of 5 to 10 feet per mile to the northwest away from the Ozark Uplift.

No active faults are known to be located within 50 miles of the site. The nearest fault to the site is 12 miles distant at Kingdom City and it is inactive. Subsurface materials are dense and stable and are adequate for the support of the plant structures.

1.2.1.6 Seismology

The site is west of the Ste. Genevieve and Chester-Dupo Seismogenic Regions and northwest of the New Madrid Seismogenic Region. No earthquake epicenter has been reported closer than about 40 miles, and only three earthquakes have been reported within 50 miles of the site since the beginning of the 19th century. None of these shocks exceeded Intensity VI on the Modified Mercalli Intensity Scale. More distant earthquakes ranged up to Intensity XI.

The three earthquakes in 1811-1812 near New Madrid, Missouri, are considered to be the largest ever to have occurred in the central and eastern United States. Extensive research has determined that the intensities in the site vicinity for these three events were probably on the order of VI-VII. The intensities at St. Louis for the 1811-1812 New Madrid events are considered to be VII-VIII. At that time, however, the town of St. Louis was located on loose Mississippi River alluvium and the shocks were amplified through this material. The town of Herculaneum, 50 miles nearer to the New Madrid epicenter and founded on more substantial material, recorded intensities of only VI. This supports the studies which establish intensities of VI-VII in the site vicinity which is also located on more substantial materials than was old St. Louis. Analysis indicates that horizontal acceleration resulting from the operating basis earthquake will be 0.12g and that from the safe shutdown earthquake will be 0.20g.

1.2.1.7 Hydrology

The site is located on a plateau about 5 miles north of the Missouri River and 117 river miles upstream from the confluence of the Missouri and Mississippi rivers. The top of the plateau is nearly level with elevations ranging from 830 to 850 feet. The elevation of the flood plain of the Missouri River nearest the site is about 525 feet. Auxvasse Creek and its tributary, Cow Creek, drain the western and northern sides of the plateau in the site area. Mud Creek drains the southern portion of the plateau while Logan Creek carries the runoff from the plateau's eastern portion. All these streams have steep gradients and are deeply incised into the upland terrain.

The Probable Maximum Flood level in the Missouri River at the site associated with an estimated discharge of 2,420,000 cubic feet per second would be at elevation 559 feet. Since all safety-related structures and components are located on the plateau some 280 feet or more feet above the maximum flood level, flooding on the Missouri River would never reach the level of these structures. Further, since the elevation of the site is higher than the surrounding terrain and since well-developed natural streams drain the plateau, isolated local flooding will not occur on the site as the result of an event as severe as Probable Maximum Precipitation. The Ultimate Heat Sink Retention Pond is of small surface area and is designed to preclude the danger of plant flooding from storm wave spillover under probable maximum storm conditions.

The groundwater aquifer systems above the Potosi Formation in the site area exhibit the characteristics of a leaky artesian system. Because of varying permeabilities, these systems appear to behave as a single hydrologic unit with varying hydrogeologic properties. Groundwater elevations in wells which penetrate the deep aquifers in the area decrease towards the Missouri River.

Only minor groundwater supplies could be produced at the site from wells drilled into the systems above the Potosi Formation. Large sustained supplies of groundwater could be obtained from the Potosi Dolomite which is capable of producing 300 to 500 gallons per minute.

1.2.1.8 Meteorology

The climate of the Callaway site is temperate continental with cold snowy winters and warm, humid summers. Based on climatological data from nearby weather stations, the normal annual average temperature is 55°F at Columbia, Missouri. Extreme temperatures for the area are 116°F for Fulton, Missouri, and -26°F for Fulton, Missouri. The normal annual mean surface wind speed is 10.3 miles per hour, and the prevailing wind direction is south-southeast for Columbia. The fastest 1-minute wind speed of 63 miles per hour from the northwest occurred in Columbia in 1952.

For 35 years of record, between 1916 and 1950, only 273 tornadoes were officially observed in the state of Missouri. However, with the development of improved tornado observing techniques by the National Weather Service Severe Storms Center, 608 tornadoes were observed in the state of Missouri for the period 1952 through 1971. The probability estimate of a tornado striking any point in 1 longitude-latitude area (3,752 square miles) centered at the plant site is 7.5×10^{-4} and the recurrence interval is 1,331 years.

The annual average precipitation is 37 inches with a maximum 24-hour precipitation of 6.61 inches recorded at Columbia, 32 miles to the northwest, in 1918. Snowfall in the site region averages 22 inches annually while the maximum 24-hour snowfall was 10.3 inches in January 1958.

Heavy fog has an annual average frequency of 16 days with a peak in January. Thunderstorms occur most frequently during the summer, occurring on 1 day out of 3. The annual average frequency of thunderstorms is 55 days.

1.2.2 FACILITY ARRANGEMENT

1.2.2.1 General Arrangement

The principal structures located at the Callaway Plant site are listed below.

a. Reactor building - houses the reactor, reactor coolant system, steam generators, pressurizer, reactor coolant pumps, accumulators, and the containment air coolers.

b. Auxiliary building - houses the engineered safety features and nuclear auxiliary systems equipment.

c. Turbine building - houses the turbine generator, condensers, main feed pumps, and other power-conversion equipment.

d. Fuel building - houses the new fuel storage vault, the spent fuel storage pool, the fuel handling system, a portion of the spent fuel pool cooling and cleanup system, the HI-TRAC Variable Weight (VW) transfer cask, the

Variable Elevation Cask Staging Pedestal (VECASP), and the Holtec Earthquake Response Mitigator (HERMIT).

e. Radwaste building - houses the radioactive waste treatment facilities and boron recycle system components.

f. Control building - houses the main control room, the computer, the Class 1E switchgear, the Class 1E battery rooms, the access control area, cable spreading rooms, and the main control room habitability systems.

g. Storage tanks - include the condensate storage tank, the hardened condensate storage tank, the refueling water storage tank, the reactor makeup water storage tank, the demineralized water tank, the emergency fuel oil storage tanks, and the radwaste discharge monitor tanks.

h. Diesel generator building - houses the diesel generators and associated equipment.

i. Transformer vaults - oil retaining pits for the main transformers, startup transformer, station service transformer, unit auxiliary transformer, and ESF transformers.

j. Technical Support Center - includes working space that contain displays of plant status, meeting and discussion areas, communications facilities, and document storage for use during a site or general emergency. The Technical Support Center also houses a support area for use during a site or general emergency.

k. Emergency Operations Facility - houses the Emergency Control and Recovery Center for use during a site or general emergency.

l. Service Building - houses offices, records storage area, lunch facilities, shower and sanitary facilities.

m. Stores Building - houses storage facilities for equipment and supplies needed for plant operation.

n. Training Center - houses training facilities which include the Callaway Plant simulator.

o. Security Building - houses protected area access control checkpoint, central alarm station, guard offices and facilities.

p. Intentionally left blank.

q. Maintenance Training Annex/Operations Support Facility - houses offices and storage areas for Operations Support organizations.

r. Cooling tower for circulating water system with associated pump house.

s. Ultimate heat sink and associated pump house and cooling tower.

t. Sewage treatment plant.

u. Electrical switchyard.

v. Water treatment plant.

w. Site-related storage tanks - including the fire water storage tanks and fuel oil storage tank.

x. Plant water intake and discharge structures (on Missouri River).

y. Outage Maintenance Facility - houses offices and shop facilities used for plant maintenance.

z. RAM Storage Building - houses storage area for equipment and materials used for plant maintenance (some with contamination).

aa. Plant Support Facility - Office building within the plant Protected Area, originally built to support the steam generator replacement project

ab. Maintenance Storage Facility - Warehouse building within the plant Protected Area, originally built to temporarily house the replacement steam generators for installation preparation.

ac. Old Steam Generator Storage Facility (OSGSF) - Reinforced concrete building located outside the plant Protected Area and within Site Owner Controlled Area used to provide storage of the original steam generators and original reactor vessel closure head removed from the plant.

ad. Technical Training Facility/Callaway Learning Center - houses training facilities including instructor offices, classrooms, and laboratory areas.

ae. Hardened Storage Building - storage facility of portable equipment to support FLEX.

af. Independent Spent Fuel Storage Installation (ISFSI) - The ISFSI is a dry, in-ground spent fuel storage system consisting of any number of Vertical Ventilated Modules (VVM) each containing one canister.

ag. ISFSI Support Building - A structural steel buidling for storing dry cask storage system (DCSS) equipment between loading campaigns.

The general arrangement of these and other major structures and equipment is shown in Figures 1.2-1 through 1.2-43.

Figures 1.2-44 through 1.2-48 show structures and facilities on the plant site, layout of yard piping, plan and section views of the ESWS pumphouse and UHS cooling tower.

1.2.3 PRINCIPAL DESIGN CRITERIA

The plant was designed so that it could be constructed and operated to produce electric power in a safe and reliable manner. Plant design conforms to applicable codes, standards, and regulations identified in appropriate sections of the FSAR.

The plant was designed, fabricated, constructed, and is operated in such a way that the release of radioactive materials to the environment is limited to values less than the limits and guideline values of applicable federal regulations pertaining to the release of radioactive materials for normal operations, abnormal events, and design basis accidents.

The plant was designed in accordance with 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, as described in Section 3.1.

The Callaway Independent Spent Fuel Storage Installation (ISFSI) was designed in accordance with 10 CFR 72, Subpart F, General Design Criteria, as described in the HI-STORM UMAX FSAR.

1.2.3.1 SNUPPS Design Envelope

The Standard Plant, or power block, as defined in Section 1.1.2 was designed and evaluated to the SNUPPS design envelope which was established by:

a. A design criterion which was conservative for all of the sites, or

b. The limiting site condition existing at any site for the condition of interest.

A tabulation of the SNUPPS design envelope is presented in Table 1.2-1.

1.2.4 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system (NSSS) consists of a reactor and four closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. The NSSS also contains an electrically heated pressurizer and various other auxiliary systems.

High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives up its thermal energy to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire reactor coolant system is composed of leaktight components to ensure that all fluids are confined to the system.

The core is of the multiregion type. All fuel assemblies are mechanically compatible, although the fuel enrichment is not the same in all the assemblies.

In the typical initial core loading, three fuel enrichments were used. Fuel assemblies with the highest enrichments were placed in the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies were arranged in a selected pattern in the central region. In subsequent refuelings, some of the fuel assemblies (e.g., one-third) were discharged. The remaining fuel assemblies were, in general, moved to other positions in the core, and fresh fuel assemblies were added to fill out the core.

Rod cluster control assemblies, which consist of clusters of cylindrical absorber rods, are used for controlling core reactivity. The absorber rods move within guide tubes in certain fuel assemblies. Each absorber rod is attached to a spider connector above the core. The spider connector is attached to a drive shaft, which may be raised and lowered by a drive mechanism mounted on the reactor vessel head. The rod cluster control assemblies drop into the core under the effect of gravity when a reactor trip (SCRAM) occurs. Supplementary reactivity control is provided by boric acid dissolved in the reactor coolant water.

The reactor coolant pumps are Westinghouse Model 93A1 vertical, singlestage, centrifugal pumps of the shaft-seal type. The Callaway Plant is the first domestic operating unit to utilize Model 93A1 reactor coolant pumps. However, Westinghouse Model 93A1 pumps were previously reviewed by the NRC in conjunction with the RESAR-3S application (Docket No. STN-50-545). A Safety Evaluation Report (NUREG-0104) and Preliminary Design Approval (PDA-7) were issued on RESAR-3S in December 1976. In addition, the Model 93A1 reactor coolant pumps are similar to the Model 93A pumps used in the Comanche Peak units (the major difference is the flow capacity as indicated in Table 1.3-1).

The steam generators are Framatome Model 73/19T vertical u-tube units, which contain Alloy 690 thermally treated (TT) Inconel tubes. The 73/19T steam generator includes features discussed in Section 5.4.2 including forged shell sections to eliminate longitudinal welds, self draining channel head to enhance decontamination efforts, and a foreign object capture system. The RSG also has a high circulation ratio to minimize corrosion, sludge buildup and chemical attack. Integral moisture separation equipment reduces the moisture content of the exiting steam to 0.10 percent or less.

Essentially all of the metal surfaces in contact with the reactor water are stainless steel, except the steam generator tubes and fuel rods, which are Inconel 690TT and Zircaloy/

Zirlo, respectively. An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation and limits pressure variations during plant load transients. The pressurizer utilized at Callaway Plant is essentially identical to those utilized in the Comanche Peak units and many facilities that are currently in operation.

Auxiliary system components are provided to charge makeup water into the reactor coolant system, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, cool system components, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

1.2.5 ENGINEERED SAFETY FEATURES AND EMERGENCY SYSTEMS

1.2.5.1 Containment

1.2.5.1.1 Containment Structure

The containment structure is a prestressed, post-tensioned concrete structure with a cylindrical wall, a hemispherical dome, and a flat foundation slab. The wall and dome form a prestressed, post-tensioned system consisting of horizontal tendons in the wall and inverted U-shaped vertical tendons in the wall and dome. The foundation slab is reinforced with carbon steel. The inside surface of the structure is lined with a carbon steel liner to ensure a high degree of leaktightness. The containment structure completely encloses the reactor and reactor coolant system, i.e., the reactor pressure vessel, the steam generators, the reactor coolant loops and portions of the associated auxiliary systems, the pressurizer, accumulator tanks, and associated piping, described in Section 1.2.4. The design ensures that the containment structure is protected against postulated missiles from both equipment failures and external sources. The containment design provides means for the integrated leak rate testing of the containment structure and for local leak rate testing of individual piping, electrical, and access penetrations of the containment. For details, refer to Chapters 3.0 and 6.0.

The containment structure design is the same standard state-of-the-art design that has been applied to several other Bechtel-designed plants. The basic structure is similar to the containment structures at Farley, Palo Verde, and Turkey Point.

1.2.5.1.2 Containment Spray System

The containment spray system is one of the two independent pressure-reducing systems and is operated in conjunction with the containment fan coolers to provide adequate cooling of the containment atmosphere following a LOCA. Two of the four fan coolers and one of the two containment spray pumps operating are sufficient to cool the containment atmosphere. This reduces the pressure inside the containment, thus minimizing the release of radioactivity from the structure.

The containment spray system supplies borated water to cool the containment atmosphere. The pumps take suction from the refueling water storage tank. When the storage tank supply is depleted, suction of the pumps is aligned to pump water from the containment sump directly into the containment during the recirculation mode of operation.

1.2.5.1.3 Containment Cooling System

The containment fan cooler system is the second of the two independent pressure-reducing systems. The system consists of four fan cooling units. The operation of two of these units and one of the containment spray pumps is sufficient to meet the design requirements for containment depressurization after a postulated LOCA. Containment atmosphere is drawn through the fan cooler units to cool the air and condense steam from the containment atmosphere after a LOCA. During normal plant operation, the fan cooler units are required to remove heat generated from equipment inside the containment and maintain the containment atmosphere below 120°F.

The containment cooling system design is a state-of-the-art design used throughout the nuclear industry. Components of the containment cooling system are similar in design and function to individual components that are used in the containment cooling systems at Farley and Palo Verde.

1.2.5.1.4 Combustible Gas Control Systems

Control of combustible gases in the containment following a LOCA is provided by two 100 percent capacity electric (thermal) hydrogen recombiners located within the containment, which maintain the post-LOCA hydrogen concentration in the containment atmosphere below the lower flammability limit.

A hydrogen purge subsystem is also provided for combustible gas control as a backup system.

The hydrogen recombiners utilized at Callaway Plant are essentially identical to those utilized in the Comanche Peak units and many facilities that are currently in operation.

1.2.5.1.5 Containment Isolation System

The containment isolation system preserves the ability of the containment boundary to minimize the release of fission products to the environment while at the same time allowing the normal and emergency passage of fluids through the containment boundary. System components include isolation valves that satisfy the containment isolation criteria. The containment isolation system is similar in design and function to the standard design that has been applied to several other Bechtel-designed plants.

1.2.5.2 Emergency Core Cooling System

The emergency core cooling system (ECCS) injects borated water into the reactor coolant system following a LOCA. This limits damage to the fuel assemblies and limits metal-water reactions and fission product release. The ECCS also provides continuous long-term post-LOCA cooling of the core by recirculating borated water between the containment sumps and the reactor core. The ECCS design at Callaway Plant is functionally identical to the ECCS design for the Comanche Peak units.

1.2.5.3 Auxiliary Feedwater System

When the main feedwater system is not in operation and the reactor coolant temperature is greater than 350°F, the auxiliary feedwater system is used to supply water to the secondary side of the steam generators.

The system consists of two motor-driven pumps which are powered by the emergency diesel generators, if there is loss of offsite power, and one steam-turbine-driven pump. During normal plant cooldown, the auxiliary feedwater system can be used to supply feedwater to the steam generators for removal of decay heat and the stored thermal energy of the reactor coolant system.

The motor-driven pumps are started automatically by low-low water level in one steam generator, loss of both main feedwater pumps, the blackout sequence signal, or the safeguards sequence signal. The turbine-driven pump is started automatically by low-low level in two steam generators or loss of voltage sequence signal on Class 1E bus.

The auxiliary feedwater system design utilizes steam-driven and ac-powered motor-driven auxiliary feedwater pumps. The Callaway design utilizes two motor-driven pumps for reliability.

1.2.6 PLANT INSTRUMENTATION AND CONTROL SYSTEMS

The plant instrumentation and control systems ensure safe and orderly operation of all systems and processes over the full operating range of the plant. The control room is designed to enable operators to start up, operate, and shut down the plant. Supervision of both the nuclear and turbine generator systems is accomplished from the control room. Additional controls at appropriate locations outside the control room (in particular, an auxiliary shutdown panel in the auxiliary building) ensure the capability of reaching and maintaining a safe shutdown condition in the unlikely event that the control room becomes uninhabitable. (Note that the control room is protected from fire, breach of security, and missiles, and contains a redundant ventilation system filtered to remove iodine.)

The Callaway instrumentation and control systems summarized below and discussed in detail in Chapter 7.0 are functionally similar to those systems utilized in the Comanche

Peak units. The Distributed Control System (DCS) described below was not compared to the Comanche Peak units.

1.2.6.1 Protection System

The plant protection system monitors selected plant parameters in order to initiate reactor trip and/or engineered safety features actuation. Multiple independent channels monitor each of the selected plant parameters. The plant protection system logic is designed to initiate automatically protective action whenever the monitored parameters reach a limiting safety system setting. Redundancy is provided to assure that no single failure will prevent protective action when it is required. The plant protection system is designed in conformance to IEEE Standard 279 "Criteria for Protection Systems for Nuclear Power Generating Stations."

1.2.6.1.1 Reactor Trip System

The reactor trip system shuts down the reactor whenever the limits of safe operation are approached. The safe operating region is defined by such considerations as mechanical/hydraulic limitations on equipment and heat transfer phenomena. Therefore, the reactor trip system keeps surveillance on process or calculated variables which are directly related to those equipment limitations. Whenever a direct process or calculated variable exceeds a setpoint, the reactor will be automatically shut down.

1.2.6.1.2 Engineered Safety Features Actuation System

The engineered safety features actuation system is designed to detect symptoms of a loss-of-coolant accident, a steam-line break, a feedwater-line break, loss of offsite power, or a fuel handling accident and to actuate the appropriate engineered safety features systems as certain threshold levels of each indicator symptom are passed.

1.2.6.2 Reactor Instrumentation and Control System

The reactor is controlled (1) by taking advantage of inherent neutronic characteristics, e.g., temperature coefficients of reactivity; (2) by control rod cluster motion, which is used for load transients and for start-up and shutdown; (3) and by a soluble neutron absorber, boron, in the form of boric acid, inserted during cold shutdown, partially removed at start-up, and adjusted in concentration during core lifetime to compensate for fuel consumption and accumulation of fission products. The control system allows the plant to accept step load decreases of up to 10 percent and ramp load decreases of up to 5 percent per minute over the entire power range.

1.2.6.3 Radiation Monitoring System

The liquid and gaseous effluents from the plant are continuously monitored for radioactivity. Release rates are monitored and recorded. The process radiation monitoring system detects radioactivity in fluid systems which is indicative of fuel clad defects and/or fluid leakage between process systems.

Area monitoring stations are provided to measure gamma radiation at selected locations in the plant. Radiation levels, as detected by these monitors, are indicated in the control room, and above normal values are annunciated.

1.2.6.4 Balance-of-Plant Instrumentation and Control Systems

The turbine and generator control systems are designed to regulate generator load. The turbine-generator protection system is designed to ensure safe operation of the unit.

The plant Distributed Control System (DCS) and additional instrumentation and controls allow manual or automatic control of various temperatures, pressures, flows, and liquid levels throughout the plant. The DCS indicators, recorders, annunciators, and the plant computer inform the operating personnel at the equipment location and/or the control room of plant conditions and performance.

The DCS provides a Human System Interface (HSI) for operators to monitor, trend, and control plant conditions and performance for systems that interface with the DCS. The DCS acquires, transmits, stores, and logs plant parameters and events. The DCS performs calculations and logic functions to control plant equipment and processes. The DCS alerts operators to plant conditions requiring operator attention.

1.2.7 PLANT ELECTRIC POWER SYSTEM

1.2.7.1 Transmission and Generation Systems

The generating unit is connected to the utility transmission system. The Union Electric Co. transmission system nominal voltage is 345 kV. Union Electric has integrated transmission networks and interconnections with neighboring systems. A description of system network and interconnection is given in Chapter 8.0.

The main generator is a General Electric 1,800 rpm, three-phase, 60-cycle synchronous unit. The generator is connected directly to the turbine shaft and is equipped with an excitation system.

Power from the generator is stepped up from 25 kV by the main transformer and supplied by overhead lines to the switchyard. An auxiliary transformer is connected to the main generator through an isolated phase bus duct to supply the auxiliary loads of the plant during power generation.

1.2.7.2 Electric Power Distribution System

Electric power is supplied from the switchyard to the on-site power system for the electrical auxiliaries through two independent circuits. One circuit supplies power through a startup transformer and the other through an engineered safety features (ESF) transformer. The startup transformer feeds two 13.8-kV busses and a second ESF transformer. Power is supplied to auxiliaries at 13.8 kV, 4.16 kV, 480 V, 480/277 V and 208/120 V ac. Refer to Figure 8.3-1.

The power distribution system includes the Class 1E and non-Class 1E ac and dc power systems. The Class 1E power system supplies equipment used to shut down the reactor and limit the release of radioactive material following a design basis event.

The Class 1E ac system consists of two independent and redundant load groups and four independent 120-V vital ac instrumentation and control power supply systems. The load groups include 4.16-kV switchgear, 480-V load centers, and motor control centers.

Two diesel generators are provided as a standby power source, one for each of the two Class 1E load groups. Each generator has sufficient capacity to operate all the equipment of one load group, which is necessary to prevent undue risk to public health and safety in the event of a design basis accident.

The non-Class 1E ac system includes 13.8-kV switchgear, 4.16-kV switchgear, 480-V load centers, and motor control centers.

The vital ac instrumentation and control power supply systems include battery systems, static inverters, and distribution panels. All voltages listed are nominal values, and all electrical Class 1E equipment is designed to accept the expected range in voltage.

The Class 1E electrical systems are similar to Class 1E systems utilized on many other Bechtel-designed plants since designs that meet the requirements and standards of the nuclear industry develop in a similar manner. The Class 1E dc systems and components are similar to the dc systems and components at Palo Verde.

Direct current power for the Class 1E dc loads is supplied by four independent Class 1E 125-V dc batteries and associated battery chargers. One 250-V and two 125-V non-Class 1E batteries and associated battery chargers are also provided to supply 250-V and 125-V dc power for the non-Class 1E dc system loads.

1.2.8 POWER CONVERSION SYSTEM

Thermal energy that is generated by the NSSS is converted into electrical energy through the steam cycle process by the turbine generator.

The turbine is a tandem-compound, six-flow, four-element, 1,800-rpm unit, having one high-pressure and three low-pressure elements. Combination moisture separator-reheaters are employed to dry and reheat the steam between the high- and low-pressure turbines. The auxiliaries include deaerating surface condensers, condenser evacuation system, turbine-driven main feedwater pumps, motor-driven condensate pumps, and seven stages of feedwater heating. The steam and turbine systems are designed to receive the heat energy produced in the reactor during normal operation as well as a 50 percent load reduction of the turbine-generator. Heat dissipation under the latter condition is accomplished by steam dump to the condenser (40 percent full load). The steam dump enables the plant to accept a loss of 50 percent external load without reactor or turbine trip. The condensers are cooled by the circulating water system.

1.2.8.1 Main Steam Supply System

The main steam supply system consists of the piping and valves that are necessary to deliver saturated steam from the steam generators to the turbine generator. Four 28-inch main steam lines carry steam from the top of the steam generators, through four main steam isolation valves, one in each line, to the turbine stop valves at the inlet to the turbine generator. Each main steam line is also equipped with five code safety valves and one power-operated relief valve. The main steam supply system is similar to the main steam supply system at Palo Verde.

1.2.8.2 Main Condenser Evacuation System

The main condenser evacuation system provides a means for removing air and noncondensible gases from the main condenser. The main condenser evacuation system uses three vacuum pumps to perform this function; two for normal operation and the third that is started to help draw a vacuum during the startup mode.

This system is similar to the main condenser evacuation system that is utilized on Palo Verde with the exception that the Palo Verde design includes a fourth vacuum pump.

1.2.8.3 Turbine Gland Sealing System

The turbine gland sealing system seals the turbine shaft penetrations and the turbine valve stems to prevent the escape of steam or the introduction of air at these places in the steam areas of the turbine. This system utilizes standard industry components and is similar in design and function to the system utilized at San Onofre.

1.2.8.4 Turbine Bypass System

The turbine bypass system, more commonly known as the steam dump system, is provided to reduce the transient effects of plant start-up, hot shutdown, cooldown, and step reductions in generator loadings on the reactor coolant system. The steam dump system has the capability to bypass up to 40 percent of full steam flow from the steam generators to the main condenser. This system uses air-operated globe valves to perform its function and is similar in design and function to the steam dump system at Palo Verde.

1.2.8.5 Circulating Water System

The circulating water system supplies cooling water from the plant's cooling water source to the main condenser to condense the steam that discharges from the exhaust of the turbine or the turbine bypass system. The Callaway site utilizes a large natural draft cooling tower for its source of cooling.

1.2.8.6 Condensate Cleanup System

The condensate cleanup system, more commonly known as the condensate demineralizer system, contains demineralizers that are utilized to maintain the required purity of the feedwater which supplies the steam generators. The condensate demineralizer system includes additional components, such as a waste collection tank and sluice water pumps.

1.2.8.7 Condensate and Feedwater System

The condensate and feedwater system receives condensate from the condenser hotwells and delivers feedwater to the steam generators at a temperature that provides maximum steam plant efficiency.

The condensate and feedwater system includes seven stages of feedwater heaters, six demineralizer vessels, and various pumps, valves, and piping to perform its intended function. This system is similar in design and function to the condensate and feedwater system at Trojan.

1.2.8.8 Steam Generator Blowdown System

The steam generator blowdown system functions to maintain the secondary side water within Callaway's water chemistry specifications. This system includes flash tanks, filters, demineralizers, containment isolation valves, and various piping, all of which are common to most plant designs. This system, however, employs an improved design which provides much larger flow rates, four to five times larger, which enhances the blowdown function.

1.2.8.9 Secondary Liquid Waste System

The secondary liquid waste system processes condensate demineralizer regeneration wastes and either directs these wastes for processing and reuse or discharge. This system also processes potentially radioactive liquid wastes. Secondary liquid waste systems are usually plant specific and depend upon the design of the systems they serve. The secondary liquid waste system is similar in design and function to most other Bechtel-designed projects.

1.2.9 AUXILIARY SYSTEMS

1.2.9.1 Chemical and Volume Control System

The chemical and volume system (CVCS) performs the following functions:

a. Reactivity control

b. Regulation of reactor coolant inventory

c. Reactor coolant purification

d. Chemical additions for corrosion control

e. Seal water injection to reactor coolant pump seals

Reactor coolant system is continuously purified by removing a small fraction of the reactor coolant flow through the letdown system. Letdown water is cooled in the regenerative heat exchanger. From there, the coolant flows to a letdown heat exchanger and through a demineralizer where corrosion and fission products are removed. The coolant then passes through a filter and is sprayed into the volume control tank, from which it is returned to the reactor coolant system by the charging system.

The CVCS automatically adjusts the amount of reactor coolant to compensate for changes in specific volume due to coolant temperature changes and reactor coolant pump shaft seal leakage in order to maintain a programmed level in the pressurizer.

The CVCS design for the Callaway Plant is similar to the CVCS design for the Comanche Peak units. The major difference is that the Callaway unit includes provisions in the CVCS and residual heat removal system (see Appendix 5.4A) to improve the capability to achieve and maintain cold shutdown.

1.2.9.2 Residual Heat Removal System

The residual heat removal system (RHRS) is used to remove heat from the reactor coolant at a controlled rate when the reactor coolant pressure is less than approximately 400 psig and the temperature is from 350°F to 140°F, and to maintain the proper reactor coolant temperature during refueling.

The residual heat removal pumps are used to circulate the reactor coolant through two residual heat removal heat exchangers, returning it to the reactor coolant system through the low-pressure injection header.

The RHRS design for the Callaway Plant is similar to the RHRS design for the Comanche Peak units. The major difference is that the Callaway unit includes provisions in the CVCS and RHRS (see Appendix 5.4A) to improve the capability to achieve and maintain cold shutdown.

1.2.9.3 Fuel Handling and Storage System

The reactor is refueled using equipment designed to handle and store spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for on-site storage or shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat. This system also provides capability for receiving, handling, and storing new fuel.

The fuel handling system is divided into two areas: the reactor cavity, which is flooded for refueling, and the fuel storage pool, which is outside the reactor containment and is always accessible to plant personnel. The fuel storage pool consists of the spent fuel pool and the cask loading pool (with fuel storage racks installed). The two areas are connected by a fuel transfer system which carries the fuel through an opening in the reactor containment. The fuel pool cooling and cleanup system removes decay heat from fuel stored in the fuel storage pool and maintains the purity of the fuel pool water.

Spent fuel is removed from the reactor vessel by a refueling machine and placed in the fuel transfer system. In the fuel storage pool, the fuel is removed from the transfer system and placed into storage racks. After a suitable decay period, the fuel may be removed from storage and loaded into a storage canister for loading in the ISFSI for storage on-site or into a shipping cask for transfer.

The fuel handling system and new fuel storage racks utilized at Callaway Plant are essentially identical to those utilized in the Comanche Peak units and many facilities that are currently in operation.

1.2.9.4 Service Water Systems

The station service water system consists of the service water system and the essential service water system.

1.2.9.4.1 Service Water System

During normal plant operation, the service water system supplies cooling water to the turbine building closed cooling water heat exchangers, turbine building chiller condensers, condenser air removal seal water coolers, steam packing exhauster, generator stator coolers, generator hydrogen cooler, turbine-generator lube oil coolers, steam generator blowdown heat exchanger, letdown chiller condenser, and air compressor. The system also supplies cooling water to the essential service water system during normal operation.

1.2.9.4.2 Essential Service Water System

The essential service water system provides cooling water from the ultimate heat sink (cooling tower) for plant components which require cooling for safe shutdown of the reactor following an accident and/or loss of offsite power. These components are the component cooling water heat exchangers, containment air coolers, diesel generator coolers, safety injection pump room coolers, RHR pump room coolers, containment spray pump room coolers, ECCS centrifugal charging pump room coolers, component cooling water pump room coolers, auxiliary feedwater pump room coolers, control room airconditioning condensers, Class 1E switchgear air-conditioning condensers, and penetration room coolers.

The essential service water system provides emergency makeup to the fuel pool and component cooling water systems. It is also the backup water supply to the auxiliary feedwater system.

1.2.9.5 Component Cooling Water System

The component cooling water system is a closed loop circulating water system serving heat exchangers whose operation is required for the safe shutdown of the reactor. Heat is removed from the closed loop by the essential service water system. Radiation monitors are provided to detect any radioactive leakage into the component cooling system.

1.2.9.6 Compressed Air Systems

Three nonlubricated air compressors, with separate aftercoolers, discharge compressed air to three air receivers that supply compressed air to a common header. This header furnishes compressed air for both the plant air system and the instrument air system. Instrument air is dried and filtered downstream of the common supply header.

The plant air system provides compressed air for normal maintenance service at various stations throughout the plant. The instrument air system provides compressed air for the operation of all air-operated instruments and valves.

1.2.9.7 Fire Protection Systems

The Fire Protection System (FPS) provides means for detecting, alarming, and extinguishing fires. The FPS was designed to provide an adequate supply of water to the plant FPS and site fire protection facilities to meet all anticipated fire water requirements. The site system also provides diversified manual and automatic fire alarms and manual and automatic fire-suppression systems to protected areas at the plant.

The plant fire protection system consists of the following materials, structures, detection devices, alarms, and suppression and extinguishing facilities, selected and designed to minimize fire hazards and fire damage:

a. Automatic wet-pipe sprinklers

b. Automatic pre-action systems

c. Water spray systems

d. Halon 1301 systems

e. Standpipe and hoserack assemblies

f. Portable extinguishers

g. Fire and smoke detection and alarm systems

h. Fire walls and barriers

i. Fire resistant and noncombustible materials of construction

j. Smoke and heat vents

Portions of the fire protection system that protect or pass through areas containing equipment required for safe shutdown of the plant during and after an earthquake are seismically analyzed and supported to prevent damage to this equipment. The system is designed to preclude flooding of safety-related equipment under seismic conditions.

The fire protection system provides an adequate supply of water to hydrants, hose stations, sprinklers, and deluge systems, based on the maximum automatic sprinkler or fixed water spray system demand with the simultaneous flow for hose streams outside the power block.

Noncombustible and heat-resistant materials are used throughout the plant. Plant fire walls are provided and rated according to their particular location in the plant, and penetrations through fire barriers are fitted with fire stops having, as a minimum, the same rating as the barrier.

Instrumentation and controls are provided for the proper operation of the fire-fighting systems and for fire detection and annunciation.

The fire protection system is designed with components and systems that are common to many plants throughout the nuclear industry and, therefore, is comparable to most fire protection systems utilized at other plants. This system is most similar in design and function to the fire protection system utilized at San Onofre.

1.2.9.8 Heating, Ventilating, and Air-Conditioning Systems

The heating, ventilating, and air-conditioning (HVAC) systems are designed to provide a suitable environment for equipment and to ensure the safety of personnel.

Redundant cooling and ventilating systems serving engineered safety features equipment rooms and the main control room meet seismic Category I requirements and are each supplied from separate Class 1E electrical busses. These systems satisfy the single failure criterion.

The nonsafety-related HVAC systems are specifically tailored to suit the design of other portions of the plant but are similar in design and function to that of other Bechtel-designed projects.

1.2.9.9 Sampling Systems

The sampling systems collect representative samples of the various process fluids. The systems include a primary sampling system, secondary sampling system, radwaste sampling system, and local grab sample provisions. Samples are routed to centralized sampling stations or local sample stations, all of which are located outside the reactor containment. Both liquid and gaseous samples are taken. Automatic and "on-line" analyses are made for some samples. Chemical and radiochemical laboratory analyses are performed on other samples to determine chemical composition, boron concentration, fission and corrosion product activity levels, dissolved gas concentration, gross radioactivity, and specific isotopic analyses. The results are used to regulate boron control adjustments, monitor fuel rod integrity, evaluate demineralizer performance, control effluent releases, and maintain correct water chemistry.

The sampling system is specifically tailored to suit the design of other portions of the plant but is similar in design and function to sampling systems utilized throughout the nuclear industry. Portions of the primary sampling system samples reactor coolant and returns excess flow back to that system (RCS). This design feature enables small chemical additions to be made to the RCS. The chemical additions may be made via two methods, (1) batch additions, or (2) continuous injection.

1.2.9.10 Service Gas System

The service gas system provides for the handling and storage of commonly used service gasses. The service gas system has provisions to protect against nitrogen and hydrogen gas ruptures and is comparable in design and function to the service gas system at Palo Verde. The service gas system also provides its function for several other gasses, e.g. carbon dioxide and oxygen.

1.2.9.11 Communication System

The communications system provides components and distribution for the total communications network of the plant, including intercom systems and remote communications devices. The communication system is similar in design and function to the communications system at Arkansas Nuclear One - Unit 2.

1.2.9.12 Diesel Generator Support Systems

The emergency diesel engine fuel oil storage and transfer system provides onsite fuel oil storage and transfer of fuel oil to the diesel engines. The storage capacity of this system is somewhat larger than the storage capacity at other plants with a similar design.

The emergency diesel engine cooling water system is a closed cycle system that provides a source of cooling water to the diesel engines. The emergency diesel engine cooling water system is the intermediate system that transfers heat between the diesel engine and the essential service water system and is similar in design and function to the typical nuclear industry design.

The emergency diesel engine starting system provides startup air to the diesel engines with two independent, redundant starting air trains per engine. The emergency diesel engine lubricating system consists of two major subsystems; 1) the main oil system, and 2) the rocker oil system, Each engine has its own independent redundant lubricating system. The emergency diesel engine combustion air intake and exhaust system provides filtered combustion air to the diesel engines and discharges the exhaust via silencers in the discharge stacks.

The diesel generator support systems are tailored to the specific design of the diesel engines and are similar in design and function to the diesel generator support systems at San Onofre.

1.2.10 WASTE PROCESSING SYSTEMS

The waste processing systems provide all the equipment necessary for controlled treatment and preparation for retention or disposal of all liquid, gaseous, and solid wastes produced as a result of reactor operation. The liquid waste processing system collects, processes, and recycles reactor grade water, removes or concentrates radioactive constituents, and processes them until suitable for reuse or for processing in the solid radwaste system. The gaseous waste processing system removes fission product gases from the reactor coolant and contains these gases during normal plant operation. The solid radwaste system receives, processes, packages, and stores all radioactive wastes generated until shipment offsite.

TABLE 1.2-1 DESIGN ENVELOPE[1]

Parameter	SNUPPS Standard Plant	FSAR Reference	Remarks
Hazards	There are no hazards which have an adverse effect on the standard plant structures	Section 2.2	Explosions from accidents involving explosives or gases have been postulated at each site in accordance with Regulatory Guide 1.91. The maximum overpressure and ground shock on the plant structures from such explosions are well below the design pressures for tornado protection and the design OBE ground accelerations.
Temperatures		Sections 2.3, 3.11(B).2.5, and 9.4	These temperatures envelop the historically recorded minimum and maximum regional temperatures and are within a range of 100-year recurrence temperatures.
1. Design min and max for exposed outside structures or components	-60°F to 120°F		
2. Design for ultimate heat sink	Site-related		
3. Design winter air conditions for ventilation	-25°F and 15 mph wind		
4. Design for summer ventilation	97°F dry bulb, 79°F wet bulb		
Flood level	Flooding of the standard plant structures is precluded by the elevation of the plant (the natural topography of each site) and by the site drainage system.	Section 2.4 and Table 3.4-1	No special flood protection measures (such as external flood doors) are incorporated into standard plant design.
Maximum rainfall	7.4 in/hr, excess allowed to run off roofs	Section 2.4	Site drainage designed to preclude local flooding.

(1) The design envelope was conservatively developed using data from all SNUPPS sites. The enveloping parameters are also conservative for Tyrone Energy Park and Sterling.

TABLE 1.2-1 (Sheet 2)

Parameter	SNUPPS Standard Plant	FSAR Reference	Remarks
Ice and snow loading		Section 2.4	Basic snow load (100-year recurrence snowpack) adjusted for geometery and drifting for roof design using Section 7.2 of ANSI A58.1-1972. "Extreme environmental" extreme environmental includes the effects of PMWP.
1. Basic snow load, normal and severe environmental	91 psf		
2. Basic snow load, extreme environmental	153 psf		
Ground water elevation	Maximum hydrostatic level is at grade	Section 2.4	Conservative assumption for buoyancy calculations and computation of uplift pressure on foundation base mats.
Seismology (OBE and SSE)	OBE = 0.12g, SSE = 0.20g	Section 2.5	The standard plant OBE and SSE are used with each site's soil properties to generate seismic structural loads. These loads are enveloped for design. Floor response spectra are generated in the same manner. All items are designed either to the envelope or all of the individual floor response spectra so that these items can be interchangeable at all sites.
Foundation characteristics			
1. Settlement	Design settlements are within the following criteria: a. Total - 3 in. b. Post construction - 1 in. c. Post construction differential (between buildings and/or columns) - 1/2 in.	Section 2.5	
2. Static and dynamic lateral earth pressures	The equations for the lateral earth pressures, shown in Figure 2.5-1, are used in conjunction with the soil parameters and the enveloping earthquake parameters (i.e., enveloping SSE and OBE) to compute the lateral pressures on the foundation walls. The maximum earth pressure thus computed is taken as the envelope pressure and is used in design.	Section 2.5	Used for design of subsurface Category I walls

TABLE 1.2-1 (Sheet 3)

Parameter	SNUPPS Standard Plant	FSAR Reference	Remarks
3. Liquefaction	Subsurface materials at all sites not susceptible to liquefaction	Section 2.5	
4. Local subsidence	No evidence of any actual or potential subsidence	Section 2.5	All sites are free from major surface or subsurface subsidence or collapse resulting from tectonic depressions, cavernous conditions, solutioning, or extraction of subsurface fluids or mining resulting from man-made activities.
Windspeed	100 mph. Tornado maximum wind speed is 360 mph with 3 psi pressure drop in 1-1/2 secs	Sections 3.3.1 and 3.3.2	BC-TOP-3-A, ANSI-A58.1-1972, and Regulatory Guide 1.76
Relative humidity	97°F dry bulb 79°F wet bulb 45% (summer) -25°F dry bulb (winter)	Section 9.4 and Table 9.4-1	These are temperature conditions based on 1972 ASHRAE Handbook of Fundamentals and are used for design of the plant HVAC systems for safety-related structures.

1.3 COMPARISON TABLES

1.3.1 COMPARISON WITH SIMILAR FACILITY DESIGNS

Table 1.3-1 presents a design comparison of the major NSSS parameters or features of Callaway Plant with Comanche Peak, Units 1 and 2 (Docket Nos. 50-445 and 50-446), W. B. McGuire, Units 1 and 2 (Docket Nos. 50-369 and 50-370), and Trojan (Docket No. 50-344). For a general design comparison of the major BOP systems utilized by Callaway Plant with similar systems on other Bechtel-designed plants, refer to the general system descriptions in Section 1.2. Refer to Table 1.3-2 for a listing of major analyses that have been used on Callaway Plant but are not included in topical reports. Most of these analyses have been previously reviewed by the NRC on other dockets. Note that approved topical reports issued by Bechtel and Westinghouse are listed in Section 1.6.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

Table 1.3-3 identifies all the significant changes that have been made to the standard plant or power block since submittal of the SNUPPS PSAR. Only items not reported in the PSAR and its subsequent amendments are listed in Table 1.3-3.

1.3.3 COMPLIANCE WITH NRC REGULATIONS

Table 1.3-4 presents a list of NRC regulations and a corresponding description regarding the degree of compliance to each regulation. Compliance with 10 CFR Parts 20, 50, 51, 70, 71, 73, and 100 is considered.

TABLE 1.3-1 DESIGN COMPARISON

Parameter or Feature	Callaway FSAR Chapter/Section	Callaway	Comanche Peak	W. B. McGuire	Trojan
Reactor core heat output, MWt	4.0, 5.0, 15.0	3,565	3,411	3,411	3,411
WRB-1,2 correlation DNBR limit for design transients (a)	4.1, 4.4, 15.0	1.22 (typ.) 1.21 (thimble)	>1.30	>1.30	>1.30
Total reactor flow rate, 10^6 lb/hr	4.1, 4.4, 5.1	139.4(b)	142.1	140.3	132.7
Reactor coolant temperatures,°F (c)(d)	4.1, 4.4				
Core outlet		625.2	621.4	620.8	619.5
Vessel outlet		620.0	618.2	618.2	616.8
Core average		593.1	591.8	589.4	585.9
Vessel average		588.4(d)	588.5	588.2	584.7
Core inlet		556.8(e)	558.8	558.1	552.5
Vessel inlet		556.8(e)	558.8	558.1	552.5
Average linear power, kW/ft	4.1, 4.4	5.69(f)	5.44	5.44	5.44
Peak linear power for normal operation, kW/ft	4.1, 4.4	14.23(g) (h)	12.6	12.6	13.6
Heat flux hot channel factor, F_Q	4.1, 4.4, 15.0	2.50(g)	2.32	2.32	2.50
Fuel assembly array	4.1, 4.3	17 x 17	17 x 17	17 x 17	17 x 17
Number of fuel assemblies	4.1, 4.3	193	193	193	193
Uranium dioxide rods per assembly	4.1, 4.3	264(h)	264	264	264
Fuel weight as uranium dioxide, lb	4.1, 4.3	204,280	222,739	222,739	222,739
Number of grids per assembly	4.1, 4.3	6(Zirc-mix) 3(Zirc-IFM) 2(Inconel-non mix)	8-Type R	8-Type R	8-Type R
Rod cluster control assemblies	4.1, 4.3				
Number of full/part length assemblies		53/-	53/-	53/8	53/8
Absorber material		Ag-In-Cd	Ag-In-Cd or hafnium	Ag-In-Cd	Ag-In-Cd

TABLE 1.3-1 (Sheet 2)

Parameter or Feature	Callaway FSAR Chapter/Section	Callaway	Comanche Peak		W. B. McGuire	Trojan
Clad material		Stainless steel with chrome plating	Stainless steel		Stainless steel	Stainless steel
Clad thickness, in.		0.0185 with 0.00075 plating	0.0185		0.0185	0.0185
Equivalent core diameter, in.	4.1, 4.3	132.7	132.7		132.7	132.7
Active fuel length, in.	4.1, 4.3	143.7	143.7		143.7	143.7
Fuel enrichment (weight percent)	4.1, 4.3		Unit 1	Unit 2		
Region 1		(i)	1.60	1.40	2.10	2.10
Region 2		(i)	2.40	2.10	2.60	2.60
Region 3		(i)	3.10	2.90	3.10	3.10
Number of coolant loops	5.0	4	4		4	4
Total steam flow, 10^6 lb/hr	5.1	15.95(j)	15.14		15.14	15.07
Reactor vessel	5.3					
Inside diameter, in.		173	173		173	173
Inlet nozzle inside diameter, in.		27 1/2	27 1/2		27 1/2	27 1/2
Outlet nozzle inside diameter, in.		29	29		29	29
Number of reactor closure head studs		54(k)	54		54	54
Reactor coolant pumps	5.4.1					
Horsepower		7,000	7,000		7,000	6,000
Capacity, gpm		100,200	99,000		99,000	88,500
Steam generators	5.4.2					
Model		73/19T	D		D	51
Heat transfer area, ft^2		78,946	48,000		48,000	51,500
Number of U-tubes		5,872	4,578		4,674	3,388
Residual heat removal	5.4.7					
Initiation pressure, psig		~425	~425		~425	~400

TABLE 1.3-1 (Sheet 3)

Parameter or Feature	Callaway FSAR Chapter/Section	Callaway	Comanche Peak	W. B. McGuire	Trojan
Initiation/completion temperature,°F		~350/140	~350/140	~350/140	~350/140
Component cooling water design temperature,°F		105	105	95	95
Cooldown time after initiation, hr		18.9	~16	~16	~16
Heat exchanger removal capacity, 10^6 Btu/hr		39.5(l)	39.1	34.15	34.2
Pressurizer	5.4.10				
Heatup rate using heaters,°F/hr		55	55	55	55
Internal volume, ft^3		1,800	1,800	1,800	1,800
Pressurizer safety valves	5.4.13				
Number		3	3	3	3
Maximum relieving capacity, lb/hr		420,000	420,000	420,000	420,000
Accumulators	6.3				
Number		4	4	4	4
Operating pressure, minimum, psig		600	600	600	600
Minimum operating water volume, each, ft^3		810	950	950	870
ECCS Centrifugal charging pumps	6.3				
Number		2	2	2	2
Design flow, gpm		150(m)	150	150	150
Design head, ft		5,800	5,800	5,800	5,800
Safety injection pumps	6.3				
Number		2	2	2	2
Design flow, gpm		425	425	425	425
Design head, ft		2,680	2,680	2,500	2,500
Residual heat removal pumps	5.4.7, 6.3				
Number		2	2	2	2
Design flow, gpm		3,800	3,800	3,000	3,000

TABLE 1.3-1 (Sheet 4)

Parameter or Feature	Callaway FSAR Chapter/Section	Callaway	Comanche Peak	W. B. McGuire	Trojan
Design head, ft		350	350	375	375
Instrumentation and controls	7.0	(n)	(n)	(n)	(n)
New fuel storage racks center-to-center spacing, in.	9.1.1	21	21	21	21
Chemical and volume control	9.3.4				
Total seal water supply flow rate, nominal, gpm		32	32	32	32
Total seal water return flow rate, nominal, gpm		12	12	12	12
Letdown flow, normal/maximum, gpm		75/120	75/120	75/120	75/120
Charging flow, normal/maximum, gpm		55/100(o)	55/100	55/100	55/100

NOTES:

(a) RTDP design limitt DNBRs are reported for Callaway.

(b) 139.4×10^6 lb/hr based on 556.8°F core inlet at a pressurizer pressure of 2250 psia. See Tables 4.1-1, 4.4-1, and 5.1-1.

(c) Based on a 5% equivalent tube plugging level in the steam generators, thermal design flow (non-RTDP).

(d) Operation with RCS T_{avg} reduced as low as 570.7°F has been evaluated to meet all criteria for acceptable plant operation.

(e) Non-RTDP T_{in}.

(f) See Section 4.1.

(g) The heat flux hot channel factor, F_Q, is 2.50 as presented in the Core Operating Limits Report (COLR).

(h) See Tables 4.4-1.

(i) Refer to Table 4.3-1a.

TABLE 1.3-1 (Sheet 5)

(j) Based on 5% steam generator tube plugging, 15.96 x 10^6 lb/hr based on 0% tube plugging.

(k) 53 tensioned in modes 1-4 per Table 5.3-2.

(l) See Section 5.4.7.2.1.

(m) Includes miniflow.

(n) The instrumentation and control systems discussed in Chapter 7.0 for Callaway are functionally similar to those systems implemented in Comanche Peak, W. B. McGuire, and Trojan.

(o) Excludes seal water.

TABLE 1.3-2 MAJOR ANALYSES NOT INCLUDED IN TOPICAL REPORTS

	Analysis Description and Name	Applicable FSAR Section	Previously Reviewed on Other Projects
Control Room Habitability			
a.	Control room air intake X/Q due to accidental releases of radiological materials	2.3	Partial use in Calvert Cliffs and Grand Gulf
b.	All other accidents, e.g., explosions, toxic chemical spills, fire, etc.	2.2	Grand Gulf
Reactor Building			
a.	Tendon Gallery [CE 901 (STRUDL)] [CE 800 (BSAP)]	3.8	Grand Gulf (1)
b.	Base Slab Bending [CE 779 (SAP 1.9)]	3.8	Grand Gulf (1)
c.	Containment Wall-Flexure [CE 779 (SAP 1.9)]	3.8	San Onofre Units 2 and 3 (1)
Reactor Building Internals			
a.	Secondary Shield Walls [CE 779 (SAP 1.9)]	3.8	San Onofre Units 2 and 3
b.	Refueling Pool (CE 779 (SAP 1.9)]	3.8	San Onofre Units 2 and 3
c.	Compartment Analysis [CE 901 (STRUDL)]	3.8	Grand Gulf (1)
Other Category I Structures			
a.	Structural Steel Framing [CE 901 (STRUDL)]	3.8	Grand Gulf Units 1 and 2
b.	Reinforced Concrete Analysis [CE 901 (STRUDL)] [CE 800 (BSAP)]	3.8	Grand Gulf Units 1 and 2
Seismic			
a.	Impedance Functions for Layered Soils [CE 970 (LUCON)]	3.7(B)	Palo Verde

TABLE 1.3-2 (Sheet 2)

Analysis Description and Name	Applicable FSAR Section	Previously Reviewed on Other Projects
b. Floor Response Spectral (FLUSH). Although not specifically named, a description of this program is included in BC-TOP-4-A	3.7(B)	None
c. Seismic Displacement Analysis [CE 933 (FASS)] (DISCOM)	3.7(B)	None
Piping Analysis		
a. ME-101 ME-632	3.9(B)	Grand Gulf, Farley
Used to calculate the stresses and loads in piping systems due to restrained thermal expansion; deadweight and seismic anchor movements, and earthquake		
b. ANSYS	3.9(B)	Grand Gulf, Farley
General static, thermal, and dynamic analysis for linear elastic and plastic analysis		
c. ME-210	3.9(B)	Grand Gulf, Farley
Computes local stresses in piping due to external loads		
d. ICES/STRUDL (CE 901)	3.9(B)	Grand Gulf, Farley
Analysis of indeterminate frame structures, both spatial and plane. Used to evaluate reactions, deflections, stresses, and code check		

(1) Although this program was not necessarily used for analysis of the same structure on another plant, it was used for similar applications.

TABLE 1.3-3 SIGNIFICANT DESIGN CHANGES FROM THE PSAR

Item	FSAR Chapter/Section	Reason for Change
Redesign of main steam tunnel for pipe break accident	3.6	The changes resulted from the NRC requirement to design for environmental effects of temperature and pressure.
Use of finite-element method, FLUSH computer program, in lieu of lumped-mass impedance method (except for structure displacement analysis).	3.7(B)/3.7(B).2	To use an improved method of seismic analysis.
Removal of T-signal (containment isolation) from valve LCV-1003 in the liquid waste processing system and the addition of valve 7176 to serve as a containment isolation valve	3.9(N).3, 11.2	Modulating control valves cannot be classified as active, because they will not meet the leakage and other technical requirements for active valves.
Deletion of part length control rods, incorporation of reduced T_{avg} control, and D bank changes from 9 rods to 5 rods	4.0, 7.7	To provide improved power distribution control during load follow operations; to comply with regulatory restrictions on use of part length rods
Incorporation of reactor coolant system cold overpressurization prevention	5.2.2, 7.6	To provide for the mitigation of potential reactor coolant system cold overpressurization transients, utilizing existing power-operated relief valves with modifications to their actuation logic
Baffle-to-barrel region configuration has been changed from downflow to upflow	5.3, 15.6.5	To reduce baffle plate and baffle bolt loading and to minimize the potential for excessive baffle joint jetting

TABLE 1.3-3 (Sheet 2)

Item	FSAR Chapter/Section	Reason for Change
Steam generators have been changed from Westinghouse model F to Framatome model 73/19T.	5.4.2	To increase the reliability of the steam generators
Thermal sleeves in the reactor coolant loop branch nozzles have been deleted.	5.4.3	To simplify the nozzle design, remove uncertainty, and show technical improvement
Residual heat removal interlocks have been revised to reflect a control room alarm setpoint if an RHR suction isolation valve is not fully closed and RCS pressure is greater than RHR system design pressure.	5.4.7	To avoid spurious auto-closures and provide operator annunciation for corrective action.
Deletion of reactor trip from reactor coolant pump breaker trip	7.2	Replace reactor trip from reactor coolant pump breaker trip with under-voltage and underfrequency sensors which are located in a seismic Category I structure
Automatic feedwater control at low power has been incorporated.	7.7	To provide the capability to automatically control steam generator water level from 0 to 25 percent power
Deletion of reactor trip following turbine trip below 50 percent power	7.2	To increase plant availability
Elimination of the low feedwater flow reactor trip	7.2, 15.0	To increase plant availability
An improved steam line break protection system has been incorporated.	7.2, 15.0	To increase plant availability by preventing spurious safety injection actuation

TABLE 1.3-3 (Sheet 3)

Item	FSAR Chapter/Section	Reason for Change
Changing spent fuel rack design to high density	9.1.2	The change to high density spent fuel racks increases the facility's capacity to accommodate 2363 spent fuel assemblies in the spent fuel pool and up to approximately 2642 assemblies ultimately in the fuel storage pool (279 assemblies in the cask loading pool with storage racks installed). The fuel storage pool consists of the spent fuel pool and the cask loading pool (with fuel storage racks installed)
Changes to the ventilation systems	9.4, 6.2	Addition of a minipurge system to the containment purge system for purging during power operation.
Elimination of the number 1 seal bypass on the reactor coolant pumps	9.3.4	In the pump design, seal injection is physically above the main radial bearing, and the seal bypass function is no longer required to maintain bearing cooling.
Revised fire protection system	9.5.1	Overall update of the fire protection system to meet current requirements.
Reactor cavity neutron shielding	12.3.2	Improved cavity seal ring design to provide operators with greater operational flexibility.
Changes to cold shutdown systems	5.4, 9.3.4	To improve the capability to achieve and maintain a cold shutdown.

TABLE 1.3-4 COMPLIANCE WITH NRC REGULATIONS, 10 CFR

Sheets 1 through 7 of this Table have been deleted.

TABLE 1.3-4 (Sheet 8)

Regulation (10 CFR)	Compliance
50.1	This regulation states the purpose of the Part 50 regulations and does not impose any independent obligations on licensees.
50.2	This regulation defines various terms and does not impose independent obligations on licensees.
50.3	This regulation governs the interpretation of the regulations by the NRC and does not impose independent obligations on licensees.
50.4	This regulation gives the address of the NRC and does not impose independent obligations on licensees.
50.10 50.11	These regulations specify the types of activities that may not be undertaken without a license from the NRC. The SNUPPS Utilities do not propose to conduct any such activities at Callaway or Wolf Creek without an NRC license.
50.12	This regulation provides for the granting of exemptions from 10 CFR Part 50 regulations, provided that such exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. It does not impose independent obligations on licensees.
50.13	This regulation says that a license applicant need not design against acts of war. It imposes no independent obligations on licensees.
50.20 50.21 50.22 50.23	These regulations describe the types of licenses that the NRC issues. They do not address the substantive requirements that an applicant must satisfy to qualify for such licenses.
50.30	This regulation sets forth procedural requirements for the filing of license applications concerning items such as place of filing, oath or affirmation, number of copies of application, application for operating license, filing fees, and an environmental report. The procedural requirements of this regulation have been met in the license application and will continue to be met for subsequent amendments to the license application.
50.31 50.32	These regulations permit more efficient organization of the license application and impose no independent obligations on licensees.
50.33	This regulation requires the licensee's application to contain certain general information, such as identification of the applicant, information about the applicant's financial qualifications, and a list of regulatory agencies with jurisdiction over the applicant's rates and services. This information is provided in the operating license application.

TABLE 1.3-4 (Sheet 9)

Regulation (10 CFR)	Compliance
50.33a	This regulation requires applicants for construction permits to submit information required for the antitrust review. The requirements set forth by this regulation were satisfied at the time the application for a construction permit was submitted.
50.34(a)	This regulation sets forth requirements which govern the content of technical information in the Preliminary Safety Analysis Report and is relevant to the construction permit stage. The requirements of this regulation were satisfied as part of the construction permit application.
50.34(b)	A Final Safety Analysis Report (FSAR) has been prepared and submitted which addresses in the chapters indicated the information required: 1. Site evaluation factors - Chapter 2.0 2. Structures, systems, and components - Chapters 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, 9.0, 10.0, 11.0, 12.0, and 15.0 3. Radioactive effluents and radiation protection - Chapters 11.0 and 12.0 4. Design and performance evaluation - ECCS performance is discussed and shown to meet the requirements of 10 CFR 50.46 in Chapters 6.0 and 15.0 5. Results of research program - Section 1.5 6. i. Organizational structure - Chapter 13.0 ii. Managerial and administrative controls - Chapter 13.0 and the OQAM. The OQAM discusses compliance with the quality assurance requirements of Appendix B. iii. Plans for preoperational testing and initial operations - Chapter 14.0 iv. Plans for conduct of normal operations - Chapters 13.0 and the OQAM. Surveillance and periodic testing is specified in the Technical Specifications. v. Plans for coping with emergencies - Emergency Plan (Site Addenda Chapter 13.0) vi. Technical specifications - will be submitted 1 year before scheduled fuel loading. vii. Potential hazards analysis (Appendix 3B) 7. Technical qualifications - Chapter 13.0 8. Operator requalification program - Site Addendum Chapter 13.0

TABLE 1.3-4 (Sheet 10)

Regulation (10 CFR)	Compliance
50.34(c) 50.34(d)	The information required in these sections has been submitted for both Wolf Creek and Callaway under separate covers pursuant to Paragraph 2.790(d) 10 CFR 2, “Rules of Practice”. This information includes both the physical security plans and the safeguards contingency plans.
50.34a	This regulation sets forth the requirements for including in the construction permit application a description of the design objectives and the preliminary design of equipment to control the release of radioactive material in nuclear power reactor effluents. The requirements of this regulation were satisfied as part of the construction permit application.
50.35	This regulation is relevant to the construction permit stage rather than the operating license stage.
50.36	Technical specifications are being prepared for implementation by SNUPPS and will include 1) safety limits and limiting safety settings, 2) limiting conditions for operations, 3) surveillance requirements, 4) design features, and 5) administrative controls. Technical specifications will take the form prescribed by NUREG 0452, Revision 3, dated November 1980 which are the "Standard Technical Specifications for Westinghouse Pressurized Water Reactors."
50.36(a)	Radiological Effluent Technical Specifications (RETS) are being prepared for implementation by SNUPPS, as required by this regulation. The RETS will take the form prescribed by NUREG 0472, Revision 2, dated July 1979.
50.37	This regulation requires the applicant to agree to limit access to restricted data. This requirement was satisfied at the time of application for the construction permit.
50.38	This regulation prohibits the NRC from issuing a license to any person who is a citizen, national, or agent of a foreign country or any corporation or other entity which is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The SNUPPS applicants are eligible to apply for and obtain a license as stated in their applications for operating licenses. Therefore, the requirements of this regulation are not applicable.
50.39	This regulation provides that applications and related documents may be made available for public inspection. This imposes no direct obligations on applicants and licensees.
50.40	This regulation provides considerations to "guide" the Commission in granting licenses, as follows:

TABLE 1.3-4 (Sheet 11)

Regulation (10 CFR)	Compliance
50.40(a)	The design and operation of the facility is to provide reasonable assurance that the health and safety of the public will not be endangered. The basis for the SNUPPS power block's assurance that the regulations will be met and the public protected is contained in this document and in the license application and the related correspondence over the years. Moreover, the lengthy process by which the plant is designed, constructed, and reviewed, including reviews by the architect-engineer, the NSSS vendor, the utilities' individual staffs, and the NRC Staff, provides a great deal of assurance that the public health and safety will not be endangered.
50.40(b)	This regulation requires that the applicant be both technically and financially qualified to engage in the proposed activities as specified in the license application. Technical and financial adequacy of the SNUPPS applicants was determined to be satisfactory during the hearing process at the construction permit stage. Additional information was provided in the operating license application.
50.40(c)	The issuance of a license to the SNUPPS applicants will not be inimical to the common defense and security or to the health and safety of the public. The individual showings of compliance with particular regulations contained in this section as well as the contents of the FSAR and related correspondence on the record, plus the lengthy process of design, construction, and review by the SNUPPS applicants, the architect engineer, the NSSS vendor, and the government ensure that the license will not be inimical to the health and safety of the public. Compliance with the requirements in 10 CFR 50.40(a) demonstrate that a license will not be inimical to the common defense and security.
50.40(d)	The requirements set forth in this regulation have been satisfied in that Environmental Reports have been submitted in accordance with 10 CFR 51 as part of the operating license application.
50.41	This regulation applies to class 104 licensees, such as those for devices used in medical therapy. SNUPPS has not applied for a class 104 license, and therefore 50.41 is not applicable.

TABLE 1.3-4 (Sheet 12)

Regulation (10 CFR)	Compliance
50.42	This regulation requires the Commission to consider additional standards in determining whether or not a license should be issued, i.e., 1) that the proposed activities will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized and 2) that due account will be taken of the antitrust advice provided by the Attorney General. Information pertinent to these standards was made known to the Commission at the construction permit stage 1) by the licensing board verification of the need for power and 2) by the Attorney General's satisfactory review of the antitrust information. An update of this information has been provided with the operating license application, in accordance with Regulatory Guide 9.3.
50.43	This regulation imposes certain duties on the NRC and addresses the applicability of the Federal Power Act and the right of government agencies to obtain NRC licenses. It imposes no direct obligations on licensees.
50.44	The SNUPPS power block combustible gas control system is described in FSAR Section 6.2.5.2. The system is designed to ensure a mixed atmosphere in containment as specified in 10 CFR 50.44(b)(1). The system consists of a hydrogen monitoring subsystem, a hydrogen mixing subsystem, and hydrogen recombiners.
50.45	This regulation provides standards for construction permits rather than operating licenses and is therefore not pertinent to this operating license proceeding.
50.46	FSAR Section 6.3 describes the emergency core cooling system and the methods used to analyze ECCS performance following the course of an accident. The results of the loss-of-coolant accident analyses presented in FSAR Section 15.6.5 demonstrate conformance with 50.46
50.50	This regulation provides that the NRC will issue a license upon determining that the application meets the standards and requirements of the Atomic Energy Act and the regulations and that the necessary notifications to other agencies or bodies have been duly made. It imposes no direct obligations on the licensees.
50.51	This regulation specifies the maximum duration of licenses. Compliance will be affected by the Commission's issuing the license in order to comply.

TABLE 1.3-4 (Sheet 13)

Regulation (10 CFR)	Compliance
50.52	This regulation provides for the combining in a single license of a number of activities. It imposes no independent obligation on the licensee.
50.53	This regulation provides that licenses are not to be issued for activities that are not under or within the jurisdiction of the United States. The operation of the SNUPPS' plants will be within the United States and subject to the jurisdiction of the United States, as is evident from the description of the facility in Part A of the operating license application.
50.54	This regulation specifies certain conditions that are incorporated in every license issued. Compliance is effected by including these conditions in the license when it is issued.
50.55	This regulation addresses conditions of construction permits, not operating licenses, and therefore it is not applicable to this application.
50.55a(a)(1)	Section 5.2 of the FSAR describes compliance with this regulation.
50.55a(a)(2)	This paragraph is general in nature leading into Paragraphs (c) through (i) of the regulation.
50.55a(b)(1) 50.55a(b)(2)	These paragraphs provide guidance concerning the approved edition and addenda of Sections III and XI of the ASME B&PV Code.
50.55a(c)	Design and fabrication of the reactor vessel were carried out in accordance with ASME Section III (1971).
50.55a(d)	Reactor coolant system piping meets the requirements of ASME Section III (1974).
50.55a(e)	Reactor coolant pumps meet the requirements of ASME Section III (1971).
50.55a(f)	Reactor coolant system valves comply with the requirements found in ASME Section III (1974).
50.55a(g)	Inservice inspection (ISI) requirements delineated in this part are specified in the Technical Specifications and the ISI Program Plan.
50.55a(h)	As discussed in Chapter 7.0, Section 7.1, the protection systems meet IEEE 279-1971.
50.55a(i)	Fracture toughness requirements are set forth in Appendices G and H of 10 CFR 50. Section 5.3 of the FSAR details vessel material parameters.
50.55b	This regulation has been revoked. 43 Fed. Reg. 49775.

TABLE 1.3-4 (Sheet 14)

Regulation (10 CFR)	Compliance
50.56	This regulation provides that the Commission will, in the absence of good cause shown to the contrary, issue an operating license upon completion of the construction of a facility in compliance with the terms and conditions of the construction permit. This imposes no independent obligations on the applicant.
50.57(a)	This regulation requires the Commission to make certain findings prior to the issuance of an operating license.
50.57(b)	The license, as issued, will contain appropriate conditions to ensure that items of contruction or modification are completed on a schedule acceptable to the Commission.
50.57(c)	This regulation provides for a low-power testing license.
50.58	This regulation provides for the review and report of the Advisory Committee on Reactor Safeguards.
50.59	This regulation provides for the licensing of certain changes, tests, and experiments at a licensed facility. Technical Specifications and procedures provide implementation of this regulation.
50.70	The Commission has assigned resident inspectors to the SNUPPS plants and space will be provided in conformance with 50.70(b)(1) through (3).
50.71	Records are and will be maintained in accordance with the requirements of sections (a) through (e) of this regulation and the license.
50.80	This regulation provides that licenses may not be transferred without NRC consent. No application for transfer has been made by the SNUPPS utilities.
50.81	This regulation permits the creation of mortgages, pledges, and liens on licensed facilities, subject to certain provisions. The regulation prohibits secured creditors from violating the Atomic Energy Act and the Commission's regulations.
50.82	This regulation provides for the termination of licenses. It does not apply to SNUPPS' because no termination of licenses has been requested.
50.90	This regulation governs applications for amendments to licenses. Future request for license amendments will be made in accordance with these requirements.
50.91	This regulation provides guidance to the NRC in issuing license amendments.

TABLE 1.3-4 (Sheet 15)

Regulation (10 CFR)	Compliance
50.100 50.101 50.102 50.103	These regulations govern the revocation, suspension, and modification of licenses by the Commission under unusual circumstances. No such circumstances are present and these regulations are not applicable.
50.109	This regulation specifies the conditions under which the NRC may require the backfitting of a facility. This regulation imposes no independent obligations on a licensee unless the NRC proposes a backfitting requirement and, therefore, this regulation is not applicable.
50.110	This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders. No enforcement action is at issue and, therefore, this regulation is not applicable.
Appendix A	FSAR Section 3.1 discusses the extent to which the design criteria for SNUPPS' plant structures, systems, and components important to safety comply with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC). As presented in Section 3.1, each criterion is first quoted and then discussed in enough detail to demonstrate SNUPPS' compliance with each criterion. In some cases, detailed evaluations of compliance with the various general design criteria are incorporated in more appropriate FSAR sections, and are located by reference.
Appendix B	Chapter 17.0 in each Site Addendum for each SNUPPS unit describes in detail the provisions of the quality assurance program which have been implemented to meet all applicable requirements of Appendix B.
Appendix C	This appendix provides a guide for establishing an applicant's financial qualifications. Financial qualifications were established at the construction permit stage, and it was found that there is reasonable assurance that the funds needed to operate the facility in compliance with the Commission's regulations are available. Updated information addressing financial qualification was submitted with the operating license application.
Appendix D	This appendix has been superseded by 10 CFR Part 51. As noted in the discussion for 10 CFR 50.40(d), the requirements of Part 51 have been satisfied.

TABLE 1.3-4 (Sheet 16)

Regulation (10 CFR)	Compliance
Appendix E	This appendix specifies requirements for emergency plans. Emergency plans are being developed to provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect the public's health and safety and prevent damage to property. The new criteria for emergency planning developed subsequent to the event at Three Mile Island, Unit 2 are factored into the emergency plans for the SNUPPS' utilities.
Appendix F	This appendix applies to fuel reprocessing plants and related waste management facilities, not to power reactors such as those found in SNUPPS' plants and is, therefore, not applicable.
Appendix G	Fracture toughness compliance can be found in FSAR Section 5.3.1.5. Assurance of adequate fracture toughness of ferritic materials in the reactor coolant pressure boundary (ASME Code, Section III, Class 1 components) is provided by compliance with the requirements for fracture toughness testing included in NB-2300 to Section III of the ASME Code and Appendix G of 10 CFR 50.
Appendix H	Reactor vessel material surveillance program requirements are delineated in this part. Technical Specifications and operating procedures have been established to implement their requirements. Further information is provided in FSAR Chapter 5.0.
Appendix I	This appendix provides numerical guides for design objectives and limiting conditions for operation to meet the criteria "as low as is reasonably achievable" for radioactive material in light water-cooled nuclear power reactor effluents. FSAR Chapters and Site Addendum Chapters 2.0, 11.0, and 12.0 discuss the extent to which the criteria for Appendix I are met.
Appendix J	Reactor containment leakage testing for water-cooled power reactors is delineated in this appendix. These requirements are given in the Technical Specifications. Additional information concerning compliance can be found in FSAR Chapter 6.0, Sections 6.2.3 and 6.2.6.
Appendix K	This appendix specifies features of acceptable ECCS evaluation models. As stated in FSAR Section 6.3, the ECCS subsystem functional parameters are integrated so that the Appendix K requirements are met over the range of anticipated accidents and single failure assumptions. In addition, the ECCS evaluation model used to demonstrate conformance with 10 CFR 50.46 (see FSAR Section 15.6.5) is in conformance with Appendix K requirements.

TABLE 1.3-4 (Sheet 17)

Regulation (10 CFR)	Compliance
Appendix L	This appendix identifies the information required to be submitted by the applicant to the Attorney General to satisfy the requirements when applying for a facility license. The requirements of this appendix were satisfied prior to the time of application for the operating license.
Appendix M	This appendix lists guidelines for the licensing of plants whose site requirements are not considered in the design of the plant structures. Since all SNUPPS' sites are considered in the plant design, this appendix is not applicable.
Appendix N	This appendix dictates the requirements applicable to duplicate plant designs on multiple sites. As allowed in this regulation, SNUPPS uses a common Safety Analysis Report; however, where site specific needs are addressed Addenda are included for reference.
Appendix 0	Appendix O dictates guidelines for the Staff in reviewing standardization of design. No independent obligation on the licensee is required.
Appendix P	Reserved.
Appendix Q	Appendix Q dictates guidelines for the staff in early review of the site and does not deal with operating license review.
51.1	This regulation states the general purpose and scope for which the Part 51 regulations are established and does not impose any independent obligations on licensees.
51.2	The definitions contained in this regulation are adhered to in all appropriate documents.
51.3	This regulation governs the interpretation of regulations and does not impose independent obligations on licensees.
51.4	This regulation specifies the authority of the NRC in granting exemptions and does not impose independent obligations on licensees.
51.5	This regulation specifies items to be completed by the staff and imposes no independent obligations on licensees.
51.6 51.7	These regulations specify that a notice of intent will be published in connection with an environmental impact statement.
51.20	This regulation sets forth the requirements for an applicant for filing an Environmental Report at the construction permit stage. These requirements were satisfied during the review of the Environmental Reports that were submitted with the applications for construction permits.

TABLE 1.3-4 (Sheet 18)

Regulation (10 CFR)	Compliance
51.21	This regulation sets forth requirements similar to those in 10 CFR 51.20 for an applicant for filing an Environmental Report at the operating license stage. These requirements were satisfied by the Environmental Reports that were submitted with the applications for operating licenses.
51.22	This regulation specifies that the Commission prepare a draft environmental impact statement and imposes no independent obligation on licensees.
51.23	This regulation sets forth the requirements regarding the contents of draft environmental statements. This regulation does not impose independent obligations on licensees.
51.24	This regulation sets forth the distribution that the Commission is required to follow for issuance of environmental impact statements. This regulation does not impose independent obligations on licensees.
51.25	This regulation sets forth the requirements regarding requests for comments on draft environmental impact statements. This regulation does not impose independent obligations on licensees.
51.26	This regulation specifies that the Commission prepare a final environmental impact statement and imposes no independent obligation on licensees.
51.40	This regulation sets forth requirements concerning the submittal of Environmental Reports that are related to the issuance of materials licenses. These requirements were satisfied by the information included in the applications for operating licenses.
51.41	This regulation sets forth the requirements concerning issuance of environmental impact statements in regard to materials licenses. This regulation does not impose independent obligations on licensees.
51.50 - 51.54	These regulations provide guidelines on public hearings on the environmental impact statements and impose no independent obligation on licensees.
51.55	This regulation makes the documents in environmental proceedings available to the public upon request without charge. The requirements of this regulation have been anticipated in that a specified portion of the production copies of Environmental Reports have been reserved for distribution to the public upon request.
51.56	This regulation makes the requirements of Part 51 applicable to this license application proceeding. Compliance with the Part 51 regulations is individually described in this section.

TABLE 1.3-4 (Sheet 19)

Regulation (10 CFR)	Compliance
70.1	This regulation states the general purpose for which Part 70 regulations are established and does not impose any independent obligations on licensees.
70.2	This regulation states the general scope of Part 70 and does not impose any independent obligations on licensees.
70.3	This regulation gives the Commission the power to authorize licenses for the shipment and possession of special nuclear material.
70.4	The definitions contained in this regulation are adhered to in all appropriate documents.
70.5	This regulation gives the address of the NRC and does not impose any independent obligations on licensees.
70.6	This regulation governs the interpretation of regulations and does not impose any independent obligations on licensees.
70.11 - 70.14	These regulations specify those persons exempted from complying with Part 70. The SNUPPS' applicants are not exempt from complying with the applicable requirements of Part 70.
70.15	Reserved.
70.18 - 70.20a	These regulations list types of licenses issued for special nuclear material. SNUPPS plans to adhere to all applicable requirements.
70.21	This regulation sets forth the requirements concerning the filing of special nuclear material license applications. At such time, when a special nuclear material license is applied for, the requirements of this regulation will be satisfied.
70.22	This regulation sets forth the requirements concerning the contents of special nuclear material license applications. At such time, when a special nuclear material license application is submitted, the requirements of this regulation will be satisfied.
70.23	This regulation defines the requirements for the approval of an application for a license to possess special nuclear material. It does not impose independent obligations on licensees.
70.24	This regulation requires licensees to install monitors which have the capability of initiating audible alarms in the event of accidental criticality. SNUPP's compliance with the requirements of this regulation is discussed in detail in Sections 9.1 and 12.3.4.
70.31	This regulation lists guidelines for the Commission to follow in issuing a license.

TABLE 1.3-4 (Sheet 20)

Regulation (10 CFR)	Compliance
70.32	This regulation defines the conditions by which the licensee must abide in order to keep the special nuclear materials license. The radiation protection program found in Chapter 12.0, Section 12.5 of the SNUPPS' Site Addendum provides information relating to the compliance of this regulation.
70.33 - 70.35	These regulations dictate procedural requirements for renewing or amending a license. The SNUPPS' utilities shall follow these guidelines when the need to renew or amend arises.
70.36	This regulation prohibits the transfer of the license. No such transfer is planned by SNUPPS.
70.37	This is a disclaimer of warranty and does not affect the SNUPPS' utilities.
70.39	This regulation sets guidelines for the manufacture of source material and does not apply to power units such as those found in the SNUPPS' plants.
70.41 70.42	These regulations provide guidance on the transfer of special nuclear material. SNUPPS will commit to follow these guidelines as appropriate.
70.44	This regulation sets forth the requirements in regard to creditors concerning special nuclear material. Information concerning creditors has been included, as applicable, in the information submitted with the operating license applications. The primary financial constituents have been identified and their relationships described.
70.51	This regulation sets forth the requirements in regard to licensees of special nuclear material that require them to maintain records and establish procedures for inventory of special nuclear material. At such a time when this regulation applies, records will be established and kept and procedures established to satisfy this regulation.
70.52	This regulation sets forth the requirements concerning reporting procedures in the event of an accidental criticality, loss or theft or attempted theft of special nuclear material. When applicable, the requirements of this regulation will be satisfied.
70.53	This regulation sets forth the requirements for submitting Material Status Reports. Where applicable, proper procedures will be developed and submitted to properly account for quantities of special nuclear material and to describe appropriate actions that should be taken in the event that material is unaccounted for.

TABLE 1.3-4 (Sheet 21)

Regulation (10 CFR)	Compliance
70.54	This regulation sets forth the requirements for the reporting of special nuclear material transfers in the Nuclear Material Transaction Report. When applicable, the proper transfer documentation will be completed.
70.55	This regulation sets forth the requirements regarding the responsibilities of the licensees with respect to affording support and access to NRC inspection personnel. Provisions will be made to satisfy the requirements of this regulation in conjunction with granting approval on an application for license for special nuclear material.
70.56	This regulation sets forth the requirements for testing the administration of the regulations in 10 CFR 71. The SNUPPS Utilities will support such testing to the extent practicable under the regulation.
70.57	This regulation sets forth the requirements for operations other than those involved in the operation of a nuclear reactor licensed to Part 50, waste disposal operations or sealed sources. No such operations are contemplated; therefore, the requirements of this regulation are not applicable.
70.58	This regulation sets forth the requirements concerning use of special nuclear material other than licensed by Part 50 and in a waste disposal operation and as sealed sources. No such use is contemplated; therefore, the requirements of this regulation are not applicable.
70.59	This regulation sets forth the requirements for effluent monitoring reporting for special nuclear material. This regulation pertains to fuel processing and fabrication and is not applicable to a utilization facility.
70.61 70.62	These regulations allow the Commission to revoke any license for special nuclear material. It does not impose independent obligations on licensees.
70.71	This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders. No enforcement action is at issue and, therefore, this regulation is not applicable.
71.1	This regulation states the general purpose for which the Part 71 regulations are established and does not impose any independent obligations on licensees.
71.2	This regulation establishes the applicability of the Part 71 regulations and imposes no independent obligations on licensees.

TABLE 1.3-4 (Sheet 22)

Regulation (10 CFR)	Compliance
71.3	This regulation prohibits delivery or transport of licensed material, except as authorized by the Commission.
71.4	The definitions contained in this regulation are adhered to in all appropriate documents.
71.5	This regulation specifies that transportation of licensed materials be done per the requirements of the Department of Transportation and Postal Service. This regulation shall be complied with per the revision of March 25, 1980.
71.6 - 71.10	These regulations specify exemptions to Part 71. SNUPPS is not exempt from Part 71 and thus is obligated to conform with all applicable requirements.
71.11	This regulation issues a general license to deliver licensed material for transport without complying with the packaging standards under certain conditions. Shipments of this nature are not anticipated and, therefore, the requirements of this regulation are not applicable.
71.12	This regulation issues a general license for shipment in certain containers and packages provided the licensee has an approved QA program. QA programs for the SNUPPS' utilities are filed with the NRC during the operations phase as part of the license application.
71.13	This regulation gives the address of the NRC and does not impose independent obligations on licensees.
71.14	This regulation prohibits interpretation of Part 71 regulations.
71.15	This regulation allows the Commission to impose additional requirements and does not impose independent obligations on licensees.
71.16	This regulation amends licenses prior to June 30, 1973, and does not affect the SNUPPS’ utilities.
71.21 - 71.63	These regulations set forth the requirements for packaging radioactive material for transport. Compliance with each of the individual parts and paragraphs will be demonstrated and documented in the license application proceedings. The requirements of these regulations will be referenced as the standards.
71.64	This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders. No enforcement action is at issue and, therefore, this regulation is not applicable.

TABLE 1.3-4 (Sheet 23)

Regulation (10 CFR)	Compliance
Appendix A	This regulation sets forth the requirements for the normal design conditions that a package must meet before it can be transported. Packages and containers will comply with these requirements.
Appendix B	This regulation sets forth the requirements concerning compliance of transport package design with hypothetical accident conditions. Packages and containers will be designed to withstand the hypothetical accident conditions described in this regulation.
Appendix C	Appendix C is a table of transport grouping of radionuclides and is to be used as a reference.
Appendix D	This regulation sets forth additional conditions that packages should be designed to endure for special form licensed material. Compliance with the requirements in this regulation will be considered when such conditions are applicable.
Appendix E	This regulation sets forth the quality assurance criteria for shipping packages for radioactive material. The quality assurance programs for the SNUPPS utilities are filed with the NRC during the operations phase as part of the license application.
73.1	This regulation states the general purpose and scope of Part 73 and does not impose independent obligations on the licensee.
73.2	The definitions contained in this regulation are adhered to in all appropriate documents.
73.3	This regulation governs the interpretation of regulations by the NRC and does not impose independent obligations on licensees.
73.4	This regulation gives the address of the NRC and does not impose any independent obligations on licensees.
73.5	This regulation allows the Commission to grant exemptions as long as they will not endanger life or property or the common defense and security. It does not impose independent obligations on licensees.
73.6	This regulation enumerates specific exemptions, including an exemption for the following: U-235 contained in uranium enriched to less than 20 percent in the U-235 isotope. Since all SNUPPS units comply with this provision, they are exempt from the requirements of 73.20, 73.25, 73.26, 73.27, 73.45, and 73.46.
73.20	The licensee is exempt from the requirements of this regulation. See 73.6.

TABLE 1.3-4 (Sheet 24)

Regulation (10 CFR)	Compliance
73.24	This regulation sets forth the requirements concerning transport of special nuclear material in passenger aircraft and in quantities in excess of formula quantities. Shipments of special nuclear material will use the requirements of this regulation for reference when such requirements are applicable.
73.25 - 73.27	The licensee is exempt from the requirements of these regulations. See 73.6.
73.30 - 73.36	These requirements have been deleted.
73.37	This regulation sets forth the requirements regarding the transport of irradiated nuclear material. This regulation places the requirements on the carrier for special nuclear material and impose no obligations on the licensee.
73.40	This regulation sets forth the requirements regarding the establishment of and maintenance of physical security systems that provide physical protection against radiological sabotage and against theft of special nuclear material. Physical security systems are provided and will be maintained to provide adequate physical protection against sabotage and theft of special nuclear material. In addition, a safeguards contingency plan will be prepared in accordance with the criteria in Appendix C of this part, submitted for Commission approval and implemented at such time as the requirements of this regulation become applicable.
73.45 73.46	The licensee is exempt from the requirements of these regulations. See 73.6.
73.50	This regulation sets forth the requirements for physical protection of licensed activities at other than nuclear power reactors.
73.55	This regulation sets forth the requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage. Features will be implemented at each site to provide for physical barriers, access control, detection aids, and communications along with a physical security organization that will ensure physical protection. The requirements, as prescribed by this regulation, will be satisfied at such time when the requirements become applicable and to the extent practicable.
73.60	This regulation applies to non-power reactors and thus does not apply to SNUPPS.

TABLE 1.3-4 (Sheet 25)

Regulation (10 CFR)	Compliance
73.67	This regulation sets forth licensee fixed site and in-transient requirements for the physical protection of special nuclear material of moderate and low strategic significance. The requirements of this regulation will be met in a manner similar to that described in the response to Paragraph 73.55.
73.70	This regulation sets forth the requirement for records for licensees subject to various Paragraphs in part 73. The requirements of this regulation will be met in conjunction with providing compliance with other paragraphs in this part.
73.71	This regulation sets forth requirements for reporting unaccounted for shipments, suspected theft, unlawful diversion, or radiological sabotage. The requirements of this regulation will be followed at such time as they become applicable.
73.72	This regulation sets forth the requirements for making advanced notice of shipment of special nuclear material. The requirements of this regulation will be met at such time when they are applicable.
73.80	This regulation governs enforcement of the Atomic Energy Act, the Energy Reorganization Act of 1974, and the NRC's regulations and orders. No enforcement action is at issue and so this regulation is not applicable.
Appendix A	This appendix groups each state into regions to be supervised by the USNRC Inspection and Enforcement and requires no obligations by the licensee.
Appendix B	The general criteria for security personnel are outlined in this appendix. The principles in this regulation are being factored into security plans for implementation at such time as when the requirements of this part become applicable.
Appendix C	This regulation sets forth the requirements for licensee safeguards contingency plans. Such plans are being developed and will be implemented at such time as they become applicable.
Appendix D	This appendix requires that licensees who transport or deliver to a carrier for transport irradiated reactor fuel assure that shipment escorts have completed a training program. These requirements were satisfied at the time of submittal of the operating license application.
100.1	This regulation is explanatory and does not impose independent obligations on licensees.
100.2	This regulation is explanatory. SNUPPS is not novel in design and is not unproven as a prototype or pilot plant.

TABLE 1.3-4 (Sheet 26)

Regulation (10 CFR)	Compliance
100.3	This regulation is explanatory and does not impose independent obligations on licensees.
100.10	The factors listed related to both the unit design and the site have been provided in the application. Site specifics, including seismology, meteorology, geology, and hydrology, are presented in Chapter 2.0 of the FSAR. The exclusion area, low population zone, and population center distance are provided and described. The FSAR also describes the characteristics of reactor design and operation.
100.11	Exclusion areas have been established, as described in each FSAR Site Addendum Section 2.1. The low population zone for each unit has been established in accordance with this requirement. The FSAR accident analyses, particularly those in Chapters 6.0 and 15.0, demonstrate that offsite doses resulting from postulated accidents would not exceed the criteria in this section of the regulation.
Appendix A	Appendix A to 10 CFR Part 100 provides seismic and geologic siting criteria for nuclear power plants. Site suitability was determined at the construction permit stage.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 APPLICANTS

This FSAR supports the operating license applications for the facilities and applicants listed in Section 1.1.1.

1.4.1.1 Description of Business

Union Electric Company (UE) is an independent investor-owned utility with its general offices located in St. Louis, Missouri, employing approximately 6,950 persons as of December, 1990. UE serves a 24,000 square mile area which includes the greater part of the St. Louis metropolitan area and the eastern third of the State of Missouri, and an area reaching into Illinois in the St. Louis area. The aggregate population of the service area is estimated at approximately 2,669,000 of which 2,394,000 reside in Missouri and 230,000 in Illinois. The electric customers in this total area number approximately 1,053,000.

UE operates one nuclear-fuel and five fossil-fueled, steam electric generating plants containing a total of 19 units and having a net generating capacity of approximately 6,680,000 kilowatts. In addition, two hydroelectric plants, one pumped storage plant, nine combustion turbine units and several small diesel units provide a net generating capacity of approximately 1,010,000 kilowatts. Aggregate net generating capacity of UE is approximately 7,690,000 kilowatts. In addition, UE owns 40% of Electric Energy Incorporated, which in turn owns the Joppa Plant in Joppa, Illinois. UE's summer share from the surplus capacity of the Joppa Plant is scheduled one year in advance.

The record peak (gross instantaneous) electrical load carried by UE was 8,085,000 kilowatts in the summer of 1996.

1.4.1.2 Description of Corporate Organization

Union Electric Company is the owner and operator of Callaway Plant. Engineering responsibility, technical cognizance during design and construction and responsibility for operation rests with the Senior Vice President and Chief Nuclear Officer. Overall responsibility for the design, construction and operation of the Callaway Plant rests with Union Electric Company.

1.4.2 SNUPPS

Several electric utility companies joined together to share costs and manage a project to design, purchase, and license two nuclear power plants of standardized design, the Standardized Nuclear Unit Power Plant System (SNUPPS) units. The three SNUPPS utilities were Kansas City Power & Light Company, Kansas Gas and Electric Company, and Union Electric Company.

The SNUPPS utilities were signatories of an agreement for standardization for nuclear generating facilities known as the SNUPPS Agreement. The Agreement specified objectives of the undertaking, defined the responsibilities of the utility participants, and establishes a method of sharing the costs. Participation in the agreement was open to any entity proposing to construct or participate with others in constructing nuclear generating facilities of approximately 1,100 electrical megawatts on a site without active earthquake potential. The SNUPPS Agreement provided for cost sharing of the duplicate portions of the plant, established an organizational structure for management of the project, and defined a mechanism for reaching decisions on joint actions on the basis of one share and one vote per unit.

The basic shared activities were: (1) design of the standardized portion of the plants, known as the power block; (2) procurement of the NSSS; (3) procurement of the turbine generators; (4) procurement of essentially all other equipment and materials for the power block; and (5) design and fabrication of the first fuel loading. Activities which were the responsibility of each individual applicant utility were: (1) design and procurement of equipment and materials for nonstandardized facilities outside of the power block; (2) construction of both standardized and nonstandardized facilities; and (3) procurement of certain power block materials to standard specifications.

The SNUPPS utilities controlled the project through a management committee composed of one officer of each company. The members elected a chairman annually and held regular meetings. This senior executive group had overall responsibility for administration of the project and for resolution of technical, contractual, and schedular problems during evolution of the project.

The SNUPPS utilities entered into individual, basically identical contracts with four contractors to purchase the materials and services for the shared activities: 1) Bechtel Power Corporation provided architect-engineering services for the power block; 2) Westinghouse supplied two identical NSSSs and, under a separate contract, supplied the first fuel loading; 3) General Electric supplied two identical turbine generators and directly related auxiliaries; and 4) Nuclear Projects, Inc., project management and furnished the technical and administrative staff to represent the utility owners and engaged consulting services and contractors, as required. These contracts were administered as one project.

Nuclear Projects, Inc., established in May 1974, furnishes services to the SNUPPS utilities for management of the SNUPPS project, as authorized and directed by the management committee. The SNUPPS Executive Director, appointed by the management committee, and the SNUPPS technical and administrative staff are employees of and consultants to Nuclear Projects, Inc.

They had the responsibility to act for the management committee and the utilities in the day-to-day administration of work under the lead architect-engineer contract. The lead architect-engineer, in turn, was delegated responsibility for administration of the turbine generator and NSSS procurement. The lead architect-engineer was responsible for the power block design and authority to procure equipment and materials for the utilities. The Executive Director also had authority to administer the contracts for design and fabrication of the first core.

Various utility committees augmented the SNUPPS staff and provided communication links between SNUPPS activities and each individual utility. Committees included a technical committee, quality assurance committee, operations committee, legal committee, construction review group, licensing coordination group, and committees for records management, spare parts, finance and accounting, public relations, and numerous ad hoc groups and task forces for special problems.

Outside of the shared activities, each utility managed site unique activities and construction. Each utility retains a site architect-engineer (identified in Section 1.4.6) to design non-standardized facilities. Construction management at the Callaway Plant was by Daniel International Corporation. Bechtel staff was located at each construction site to interpret plans and specifications and expedite procurements. A SNUPPS staff member was at each active site to ensure that construction experience was made available to later plants.

1.4.3 NUCLEAR STEAM SUPPLY SYSTEM MANUFACTURER

Westinghouse Electric Corporation (Westinghouse) was responsible for suppling the NSSS and first fuel load for Callaway Plant.

Westinghouse has designed, developed, and manufactured nuclear facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which has produced power since 1957. Completed or presently contracted commercial nuclear capacity totals in excess of 97,000 MW. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Among the company's own related manufacturing facilities are the Columbia Plant, Nuclear Fuel Division, the largest commercial nuclear fuel fabrication facility in the world, and the Tampa Division Plant, the world's most modern heat transfer equipment production facility.

1.4.4 STANDARD PLANT (LEAD) ARCHITECT/ENGINEER

The Eastern Power Division of Bechtel Power Corporation (Bechtel) was retained by the SNUPPS utilities to provide architect/ engineer services, including procurement, for the standardized portions of the nuclear electric generating facilities.

The Bechtel Corporation, the parent of Bechtel Power Corporation, has been continuously engaged in construction and engineering activities since 1898. Since the close of World War II, Bechtel has placed strong emphasis on electrical power generation projects. During this period, Bechtel has been responsible for the design of over 204 thermal generating units, representing more than 126,860 MW of new generating capacity. Of this number, a nuclear capacity of more than 65,800 MW has been or is being engineered by the company itself.

The ratings of thermal generating plants designed by Bechtel range up to 1,470 MW per unit and include most types of station designs and arrangements, such as reheat and nonreheat, indoor and outdoor stations, single and multiple units, and wide ranges of steam conditions up to 3,500 psig, 1,050/1,000°F. Also, some of the larger units are fully automated and computer controlled. The majority of contracts for these facilities provided Bechtel with complete responsibility for both engineering and construction, although several contracts have been engineering design assignments only.

For over 25 years, Bechtel has been actively working on nuclear projects involving power plants, as well as such facilities as nuclear accelerators, research laboratories, hot cells, experimental reactors, and nuclear fuel processing plants. Its responsibilities have covered design, construction, site surveys, license applications, feasibility studies, and equipment procurement.

1.4.5 TURBINE-GENERATOR MANUFACTURER

The General Electric Company was responsible for the design, fabrication, and delivery of the turbine generators, and provided technical assistance for installation, startup, and operation of this equipment.

General Electric has a long history in the application of turbine generators for nuclear power plants.

1.4.6 SITE ARCHITECT/ENGINEER

For the site-related work Sverdrup & Parcel and Associates, Inc. (S&P) was retained as site architect engineer to provide engineering and architectural services for the plant systems and facilities which were not the responsibility of the standard plant architect/engineer. In general, the responsibilities of S&P included the site layout, the location of the power block, the design of yard and construction facilities, and the location and design of the circulating water systems. S&P was responsible for the design of site-related systems and facilities which are nonseismic Category I and for seismic Category I dams, canals, ponds (except for Category I UHS Retention Pond), and earthworks.

1.4.7 CONSULTANTS

1.4.7.1 Principal Consultants

Principal consultants to Callaway Plant and their related responsibilities were:

a. Quadrex Corporation (formerly Nuclear Services Corporation)

This consultant assisted the Callaway Plant staff to coordinate the preparation of power block operating procedures and review and approval action of Bechtel-prepared flush, hydrostatic, preoperation, and special test procedures. The compilation of specific data lists, useful for operating procedure preparation, power plant operation, and maintenance, was assigned to this consultant on an as-needed basis. This consultant also performed third-level design reviews of selected systems for compliance with codes and regulations.

b. Southwest Research Institute

This consultant reviewed portions of the Callaway Plant design to assure that adequate provision was made for preservice and inservice inspection, including access engineering, and to verify the performance of mechanical equipment. In the latter category, this consultant has performed analog simulation of the reactor charging system and recommended the design of pulsation suppressors chosen for use in Callaway Plant.

c. NUS Corporation

The nuclear engineering, plant design, and nuclear power plant licensing skills and experience of this consultant were drawn upon on an as-needed basis to perform a number of activities. Examples include drafting specifications for a loose parts monitor, carrying out an independent review of Bechtel's calculations for shielding the reactor cavity, and reviewing the Callaway Plant's cold shutdown capability.

d. Nuclear Water & Waste Technology, Inc.

This consultant, a specialist in water chemistry, reviewed the design and assisted in the selection of fluid systems and equipment, particularly the condensate polisher liquid radwaste systems, and process control instrumentation.

e. Pickard, Lowe and Garrick, Inc.

This consultant was utilized early in the Callaway Plant project to assist in bid evaluations and selection of the Standard Plant A/E. This consultant remains available on an as-needed basis, and provides occasional assistance in matters related to nuclear design and performance, such as reviewing the performance of nuclear fuel designs.

f. Professional Loss Control, Inc.

This consultant reviewed the Callaway Plant fire protection system and assisted in making related design decisions.

g. Energy Research & Consultants Corporation

This consultant reviewed design and operation of pumps and other rotating equipment, included advising Callaway Plant during the bid evaluation for several pumps, and performing tests necessary to evaluate the auxiliary feedwater pumps.

h. Dr. James Halitsky

This consultant developed calculations of atmospheric dispersion parameters for the control room fresh air intake for use in control room accident dose calculations.

i. Energy Incorporated

This consultant was engaged to assist the Callaway Plant in developing an independent plant transient and analysis capability using the RETRAN computer code.

j. Essex Corporation

This consultant was engaged to perform an independent design evaluation of the Callaway Plant control room, emphasizing human factors considerations.

These consultants may continue to be used, if necessary, as reviewers and sources of further information to respond to licensing and design questions.

1.4.7.2 Other Consultants

Consultants were hired to provide services in the areas of geology, meteorology, demography, hydrology, aquatic and terrestrial aspects, population, land use, thermal and chemical effects, and biological factors and in the preparation of Environmental Reports. The consultants for Callaway Plant include Dames & Moore, Nuclear Services Corporation, VT Technologies, Inc., and Radiation Management Corporation. A brief description of each follows.

1.4.7.2.1 Dames & Moore

Dames & Moore, (D & M) as Consultants in the Applied Earth and Environmental Sciences, is active in performing geotechnical investigations and environmental studies related to the siting, design, construction and operation of nuclear power plants. The firm assisted Union Electric in performing a comprehensive plant siting study which resulted in the selection of the Callaway Plant site. Their services have also included the procurement of data and the preparation of the sections of the Preliminary and Final Safety Analysis Reports dealing with demography, land use, site meteorology, hydrology, geology, seismology and soils and foundation engineering. Dames & Moore was also responsible for the data acquisition and preparation of the Environmental Report - Construction Permit Stage as well as the preparation of selected sections of the Environmental Report - Operating License Stage. During construction, D & M served Union Electric in monitoring all activities related to earthwork construction.

1.4.7.2.2 Nuclear Services Corporation

Nuclear Services Corporation (NSC) (name has now been changed to Quadrex Corporation) provided a wide range of capabilities including assistance in nuclear power plant engineering, quality assurance services, nuclear fuel service, and plant startup and operations planning. The organization consists of a broadly-based professional staff covering many engineering and quality assurance fields with particular emphasis on quality assurance. Union Electric employed NSC as a quality assurance consultant. NSC assisted UE in formulating a complete design and construction quality assurance program and in implementing a Quality Assurance Procedures package.

1.4.7.2.3 VT Technologies, Inc.

VT Technologies, Inc. is engaged in the design, construction, installation, and service of security systems for nuclear power plants and other installations. They are also active in providing specialized engineering and security planning for the nuclear industry. Union Electric employed VT Technologies, Inc. as a security consultant. They assisted Union Electric in developing design criteria, system description, security building layout, and technical specifications for equipment in the area of security outside the Power Block.

1.4.7.2.4 Radiation Management Corporation

Radiation Management Corporation (RMC) engages in a wide variety of activities which include radiation physics, health physics, chemistry instrumentation, biology, medicine, environmental sciences, and emergency support. Union Electric employed RMC as a consultant to assist in developing our Radiological Emergency Response Plan.

1.4.8 CONSTRUCTOR

The constructor for Callaway Plant was Daniel International Corporation. Daniel's scope of work consists of receiving design information as prepared by Bechtel, Westinghouse, and Sverdrup & Parcel; receiving manufactured items and materials as procured by Bechtel, Westinghouse, Union Electric and Daniel; procuring additional bulk materials and consumable items; procuring the services of various sub-contractors; planning and scheduling the activities of Daniel and sub-contractor forces to perform construction, testing, and inspection activities in accordance with the design.

Daniel International Corporation, a subsidiary of Fluor Corporation, is headquartered in Greenville, South Carolina. Charles E. Daniel established Daniel Construction Company in Anderson, S.C. in 1934. In 45 years Daniel expanded over much of the United States and into foreign countries. Basically an engineering, construction, and industrial maintenance company, Daniel performs services on an annual average of 230 projects. More than 90 percent of Daniel's projects are in the area of power, chemicals, fibers, food and other types of industrial plants.

The power group of Daniel International Corporation specializes in providing management and technical expertise to utility clients for power plant construction. The power group has recently constructed, or has current contracts for eleven nuclear units, four fossil units and two hydroelectric plants, with a capacity in excess of 16,000 MW. This total capacity is divided as follows:

Nuclear	10,700 MW	(10 PWR's - 1 BWR)
Fossil	2,900 MW	
Hydro	2,500 MW	

1.4.9 DIVISION OF RESPONSIBILITIES

1.4.9.1 Utility Company

The ultimate responsibility for the proper design, construction, and operation for the entire spectrum of safety of Callaway Plant rests with Union Electric Company.

1.4.9.2 Standard Plant Architect/Engineer

Bechtel Power Corporation was responsible for the design, engineering, and procurement of the standard power block, which includes the following:

a. Turbine building

b. Reactor building

c. Auxiliary building

d. Fuel building

e. Radwaste building

f. Diesel generator building

g. Control building

Bechtel was also responsible for the design of the standard plant storage tanks and transformer vaults. However, Union Electric arranged to procure this equipment. In addition, main power transformers and certain seismic Category I structures, systems, and equipment outside the power block, were designed by Bechtel.

All systems, equipment, and structures within the standard power block were designed or specified by Bechtel. The NSSS portion of the facility was procured by an individual contract between Union Electric and the NSSS supplier. Similarly, the turbine generator was obtained by direct contract between the turbine generator supplier and Union Electric. However, Bechtel Power Corporation (acting as agent) retained responsibility for monitoring the design and integrating the system into the power block to ensure that the NSSS and turbine generator components being supplied were consistent with the needs of the facility. Other equipment and material for areas within their scope were procured by Bechtel Power Corporation.

1.4.9.3 SNUPPS Staff

The SNUPPS Staff functioned as an extension of the management, engineering, and operations organization of Callaway Plant. During the design and construction phases, the SNUPPS Staff performed day-to-day administration of all of the shared activities, primarily by interfacing with and providing written direction to Bechtel Power Corporation. This required a close relationship between the SNUPPS Staff and Callaway Plant, which was achieved by frequent communications and regularly scheduled meetings of the various committees and groups.

1.4.9.4 Site Architect/Engineer

As indicated in Subsection 1.4.6, Sverdrup & Parcel provided engineering and technical services for all systems, equipment, and structures outside the Standard Power Block except for the switchyard and safety-related structures and equipment connected with the Essential Service Water System. Interfaces were established by Bechtel Power Corporation, the Standard Plant A/E, to assure compatibility of design between Power Block and site-related facilities.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

One of the design bases for the SNUPPS project has been to utilize well-developed and proven design concepts, systems, and equipment, in order to minimize the potential for cost and schedule overruns and to enhance the reliability of operation. As a consequence, there have been few requirements, as delineated by 10 CFR 50.34(a)(8), for research and development programs to confirm the adequacy of the design. Two such programs were identified at the construction permit stage. Those programs have been satisfactorily completed, as described in Sections 1.5.1 and 1.5.2. Other programs were identified at the construction permit stage as not required but as valuable to define margins of conservatism or possible design improvements. Relevant programs in this latter category are described in Section 1.5.3.

1.5.1 17 x 17 FUEL ASSEMBLY

A comprehensive test program for the 17 x 17 assembly has been successfully completed by Westinghouse. Reference 1 contains a summmary discussion of the program. The following sections present specific references documenting individual portions of the program.

1.5.1.1 Rod Cluster Control Spider Tests

Rod cluster control spider tests have been completed. For a further discussion of these tests, refer to Section 4.2.4.3.

1.5.1.2 Grid Tests

Verification tests of the structural adequacy of the grid design have been completed. Refer to Section 4.2.3.4 and Reference 2 for a discussion of these tests.

1.5.1.3 Fuel Assembly Structural Tests

Fuel assembly structural tests have been completed. Refer to References 2 and 3 for a discussion of these tests.

1.5.1.4 Guide Tube Tests

Verification tests of the structural adequacy of the guide tubes have been completed. Refer to References 3 and 4 for a discussion of these tests.

1.5.1.5 Prototype Assembly Tests

Verification tests of the integrated fuel assembly and rod cluster control performance have been completed. Refer to References 3 and 4 for a discussion of these tests.

1.5.1.6 Departure from Nucleate Boiling Tests

The test program for experimentally determining the effect of the fuel assembly geometry on the departure from nucleate boiling (DNB) heat flux has been completed. Refer to Reference 5 for a discussion of these tests.

1.5.1.7 Incore Flow Mixing

The experimental test program to determine the effects of the fuel assembly geometry on mixing has been completed. Refer to Reference 6 for a discussion of these tests.

1.5.2 FIRE STOPS

A test program to determine the adequacy of various fire stop designs has been completed. Penetration seals compatible with the SNUPPS design were successfully tested, using silicone foam sealant. Details of the fire stop designs are provided in Section 9.5.1.

1.5.3 OTHER PROGRAMS

1.5.3.1 Generic Programs of Westinghouse

Reference 7 summarizes ongoing safety-related research and development programs that are being carried out for, or by, or in conjunction with the Westinghouse Nuclear Energy System Division and that are applicable to Westinghouse pressurized water reactors. These programs are applicable to SNUPPS and may lead to changes in safety analyses or modes of operation. Further progress on these programs is not required for safe operation of the SNUPPS plants.

Experimental test programs to determine the thermal-hydraulic characteristics of 17 x 17 fuel assemblies and to obtain experimental reflooding heat transfer data under simulated LOCA conditions have been completed. Refer to Reference 8 for a discussion of these tests. A single rod burst test program to quantify the maximum assembly flow blockage which is assumed in the LOCA analyses has been completed. Refer to Reference 9 for a discussion of these tests. The results of these two test programs have been used in the ECCS analyses in Section 15.6.5.

1.5.3.2 Generic Programs of Bechtel

SNUPPS has contributed, with other utilities, to tests of prototypical cable trays under seismically induced loads. A primary objective of the tests has been evaluation of damping coefficients under SSE conditions. Mechanical bracing of cable trays in the SNUPPS plants is verified by the results of this test program.

1.5.3.3 Test of a Wolf Creek Steam Generator

One of the steam generators in the Wolf Creek plant is being equipped with special pressure and temperature instrumentation that will enable thermal-hydraulic performance characteristics to be measured during the early stages of power operation. The objective of these tests is primarily to confirm Westinghouse's design analyses. The test results will be proprietary but will be available for NRC review.

1.5.4 REFERENCES

1. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Spring 1976," WCAP-8768, June, 1976.

2. Gesinski, L. and Chiang, D., "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8236 (Proprietary) and WCAP-8288 (Non-Proprietary), December, 1973.

3. DeMario, E. E., "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8278 (Proprietary) and WCAP-8279 (Non-Proprietary), February, 1974.

4. Cooper, F. W., Jr., "17 x 17 Driveline Component Tests - Phase IB, II, III, D-Loop Drop and Deflection," WCAP-8446 (Proprietary) and WCAP-8449 (Non-Proprietary), December, 1974.

5. Hill, K. W., et al., "Effect of 17 x 17 Fuel Assembly Geometry on DNB," WCAP-8296-P-A (Proprietary) and WCAP-8297-A (Non-Proprietary), February, 1975.

6. Cadek, F. F., Motley, F. E. and Dominicis, D. P., "Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid," WCAP-7941-P-A (Proprietary) and WCAP-7959-A (Non-Proprietary), January, 1975.

7. Eggleston, F. T., "Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 - Summer 1978," WCAP-8768, Revision 2, October, 1978.

8. "Westinghouse ECCS Evaluation Model - October 1975 Version," WCAP-8622 (Proprietary) and WCAP-8623 (Non-Proprietary), November, 1975.

9. Kuchirka, P. J., "17 x 17 Design Fuel Rod Behavior During Simulated Loss-of-Coolant Accident Conditions," WCAP-8289 (Proprietary) and WCAP-8290 (Non-Proprietary), November, 1974.

1.6 MATERIAL INCORPORATED BY REFERENCE

The Standard Plant FSAR incorporates, by reference, various topical reports as part of the applications. Bechtel topical reports are listed in Table 1.6-1, and Westinghouse topical reports are listed in Table 1.6-2. The Bechtel and Westinghouse topical reports have been filed separately in support of this and similar applications.

TABLE 1.6-1 BECHTEL TOPICAL REPORTS INCORPORATED BY REFERENCE

Bechtel Topical Report No.	Title	Revision Number	FSAR Section Reference	Report Submitted to the NRC	Review(1) Status
BC-TOP-1	Containment Building Liner Plate Design Report	Rev. 1	3.7(B).3 3.8	1/73	A
BC-TOP-3-A	Tornado and Extreme Wind Design Criteria for Nuclear Power Plants	Rev. 3	3.3 3.8	8/74	A
BC-TOP-4-A	Seismic Analyses of Structures and Equipment for Nuclear Power Plants	Rev. 3	3.7(B).2 3.7(B).3 3.8	11/74	A
BC-TOP-5-A	Prestressed Concrete Nuclear Reactor Containment Structures	Rev. 3	3.8 3A	2/75	A
BC-TOP-7	Full Scale Buttress Test for Prestressed Nuclear Containment Structures	Rev. 0	3.8 3A	9/72	A
BC-TOP-8	Tendon End Anchor Reinforcement Test	Rev. 0	3.8 3A	9/72	A
BC-TOP-9-A	Design of Structures for Missile Impact	Rev. 2	3.8 3.5.3.2	9/74	A

TABLE 1.6-1 (Sheet 2)

Bechtel Topical Report No.	Title	Revision Number	FSAR Section Reference	Report Submitted to the NRC	Review(1) Status
BN-TOP-1	Test Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants	Rev. 1	3.8 6.2	11/72	A
BN-TOP-2	Design for Pipe Break Effects	Rev. 2	3.6 3.8	5/74	A
BN-TOP-3	Performance and Sizing of Dry Pressure Containments	Rev. 3	6.2.1	8/75	P
BN-TOP-4	Subcompartment Pressure and Temperature Transient Analysis	Rev. 1	6.2.1 3.6	10/77	A
BP-TOP-1	Seismic Analysis of Piping Systems	Rev. 3	3.7(B).2 3.7(B).3 3.9(B).7	1/76	A

(1) A legend for the review status code letter follows:

A - Approved by the NRC

P - Pending approval by the NRC

TABLE 1.6-2 WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-2048	"The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements"	Rev. 0	4.3	7/62	O
WCAP-2850-L(P) WCAP-7916	"Single Phase Local Boiling And Bulk Boiling Pressure Drop Correlations"	Rev. 0	4.4	5/66	O
WCAP-2923	"In-Pile Measurement of UO_2 Thermal Conductivity"	Rev. 0	4.4	3/66	O
WCAP-3269-8	"Hydraulic Tests of the San Onofre Reactor Model"	Rev. 0	4.4	6/64	O
WCAP-3269-26	"LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM - 7094"	Rev. 0	4.3, 15.0, 15.4	9/63	O
WCAP-3385-56	"Saxton Core II Fuel Performance Evaluation,"WCAP-3385-56, Part II "Evaluation of Mass Spectrometric and Radiochemical Materials Analyses of Irradiated Saxton Plutonium Fuel"	Rev. 0	4.3, 4.4	7/70	O
WCAP-3680-20	"Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors" (EURAEC-1974)	Rev. 0	4.3	3/68	O

TABLE 1.6-2 (Sheet 2)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-3680-21	"Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors" (EURAEC-2111)	Rev. 0	4.3	2/69	O
WCAP-3680-22	"Xenon-Induced Spatial Instabilities in Three-Dimensions"(EURAEC-2116)	Rev. 0	4.3	9/69	O
WCAP-3696-8	"Pressurized Water Reactor pH Reactivity Effect Final Report" (EURAEC-2074)	Rev. 0	4.3	10/68	O
WCAP-3726-1	"PuO_2 - UO_2 Fueled Critical Experiments"	Rev. 0	4.3	7/67	O
WCAP-6065	"Melting Point of Irradiated UO_2"	Rev. 0	4.4	2/65	O
WCAP-6069	"Burnup Physics of Heterogeneous Reactor Lattices"	Rev. 0	4.4	6/65	O
WCAP-6073	"LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS"	Rev. 0	4.3	4/66	O
WCAP-6086	"Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium"	Rev. 0	4.3	8/69	0

TABLE 1.6-2 (Sheet 3)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7015	"Subchannel Thermal Analysis of Rod Bundle Cores"	Rev. 1	4.4	2/14/69	O
WCAP-7048-P-A(P) WCAP-7757-A	"The PANDA Code"	Rev. 0	4.3	1/9/75	A
WCAP-7198-L(P) WCAP-7825	"Evaluation of Protective Coatings for use in Reactor Containment"	Rev. 0	6.1	4/23/69 12/16/71	O
WCAP-7213-P-A(P) WCAP-7758-A	"The TURTLE 24.0 Diffusion Depletion Code"	Rev. 0	4.3, 15.0, 15.4	1/9/75	A
WCAP-7240(P)	"An Experimental Investigation of the Effect of Open Channel Flow on Thermal-Hydrodynamic Flow Stability"	Rev. 0		7/7/72	B
WCAP-7308-L(P) WCAP-7810	"Evaluation of Nuclear Hot Channel Factor Uncertainties"	Rev. 0	4.3	7/9/70 12/16/71	U
WCAP-7359-L(P) WCAP-7838	"Application of the THINC Program to PWR Design"	Rev. 0	4.4	9/8/69 1/17/72	O
WCAP-7397-L(P) WCAP-7817	"Seismic Testing of Electrical and Control Equipment"	Rev. 0	3.10(N)	2/6/70 12/16/71	U
WCAP-7397-L(P) WCAP-7817	"Seismic Testing of Electrical And Control Equipment (WCID Process Control Equipment)"	Supp. 1	3.10(N)	1/27/71 12/16/71	U

TABLE 1.6-2 (Sheet 4)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7477-L(P) WCAP-7735	"Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems"	Rev. 0	5.2	3/26/70 8/12/71	A
WCAP-7488-L(P) WCAP-7672	Solid State Logic Protection System Description	Rev. 0	7.2, 7.3	3/24/71 5/27/71	B
WCAP-7558	"Seismic Vibration Testing with Sine Beats"	Rev. 0	3.10(N)	9/25/72	U
WCAP-7588	"An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods"	Rev. 1A	15.4	1/7/75	A
WCAP-7667-P-A(P) WCAP-7755-A	"Interchannel Thermal Mixing with Mixing Vane Grids"	Rev. 0	4.4	1/27/75	A
WCAP-7695-P-A(P) WCAP-7958-A	"DNB Tests Results for New Mixing Vane Grids (R)"	Rev. 0	4.4	1/21/75	A
WCAP-7695 Addendum 1-P-A(P) WCAP-7985, Addendum 1-A	"DNB Test Results for R Grid Thimble Cold Wall Cells"	Rev. 0	4.4	1/21/75	A

TABLE 1.6-2 (Sheet 5)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7705	"Testing of Engineered Safety Features Actuation System"	Rev. 2		5/5/76	B
WCAP-7706-L(P) WCAP-7706	"An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients"	Rev. 0	4.6, 7.1, 7.2, 7.3	9/2/71	U
WCAP-7709-L(P) WCAP-7820	"Electrical Hydrogen Recombiner for Water Reactor Containments"	Rev. 0	6.2.5	7/14/71 12/16/71	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments - Final Development Report"	Supp. 1	6.2.5	5/23/72 5/31/72	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments - Equipment Qualification Report"	Supp. 2	6.2.5	9/24/73 11/2/73	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments - Long-Term Tests"	Supp. 3	6.2.5	1/23/74 3/22/74	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments"	Supp. 4	6.2.5	4/21/74	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner Special Tests"	Supp. 5	6.2.5	1/7/76	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner IEEE 323-1974 Qualification"	Supp. 6	6.2.5	11/5/76	A

TABLE 1.6-2 (Sheet 6)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner LWR Containments - Supplemental Test Number 2"	Supp. 7	6.2.5	9/21/77	A
WCAP-7750	"A Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant (SATAN-IV Digital Code)"	Rev. 0	3.6.3	8/31/71	O
WCAP-7769(2)	"Overpressure Protection for Westinghouse Pressurized Water Reactors"	Rev. 1	5.2, 15.2	7/5/72	U
WCAP-7798-L(P) WCAP-7803	"Behavior of Austenitic Stainless Steel in Post Hypothetical Loss-of-Coolant Environment"	Rev. 0	6.1	12/6/71 1/4/72	O
WCAP-7800	"Nuclear Fuel Division Quality Assurance Program Plan"	Rev. 4A	3A 4.2	4/28/75	A
WCAP-7806	"Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods"	Rev. 0	4.3	12/16/71	B
WCAP-7811	"Power Distribution Control of Westinghouse Pressurized Water Reactors"	Rev. 0	4.3	12/16/71	O

TABLE 1.6-2 (Sheet 7)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)"	Supp. 2	3.10(N)	1/17/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Westinghouse Solid State Protection System) (Low Seismic Plants)"	Supp. 3	3.10(N)	1/17/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (WCIDNUCANA 7300 Series) (Low Seismic Plants)"	Supp. 4	3.10(N)	12/14/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Instrument Bus Distribution Panel) (Low Seismic Plants)"	Supp. 5	3.10(N)	3/19/75	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Type DB Reactor Trip Switchgear)"	Supp. 6	3.10(N)	8/74	U
WCAP-7832	"Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate Under Combined LOCA Plus SSE Conditions"	Rev. 0	5.4	12/26/73	A
WCAP-7836	"Inlet Orificing of Open PWR Cores"	Rev. 0	4.4	1/17/72	B
WCAP-7870	"Neutron Shielding Pads"	Rev. 0	3.9(N)	7/17/72	A

TABLE 1.6-2 (Sheet 8)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7907	"LOFTRAN Code Description"	Rev. 0	5.2, 15.0, 15.1, 15.2, 15.3, 15.4, 15.5, 15.6	10/11/72	U
WCAP-7908	"FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod"	Rev. 0	15.0, 15.3, 15.4	9/20/72	U
WCAP-7909	"MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System"	Rev. 0		10/11/72	U
WCAP-7912-P-A(P) WCAP-7912-A	"Power Peaking Factors"	Rev. 0	4.3, 4.4	1/16/75	A
WCAP-7913	"Process Instrumentation for Westinghouse Nuclear Steam Supply Systems (4-Loop Plant Using WCID-7300 Series Process Instrumentation)"	Rev. 0	7.2, 7.3	3/9/73	B
WCAP-7921-AR	"Damping Values of Nuclear Power Plant Components"	Rev. 0	3.7(N), 3A	7/11/74	A
WCAP-7924-A	"Basis for Heatup and Cooldown Limit Curves"	Rev. 0		4/28/75	A

TABLE 1.6-2 (Sheet 9)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-7941-P-A(P) WCAP-7959-A	"Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Van Grid"	Rev. 0	1.5, 4.4	1/27/75	A
WCAP-7956	"THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores"	Rev. 0	4.4	10/22/73	A
WCAP-7964	"Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor"	Rev. 0	4.3	6/15/71	O
WCAP-7979-P-A(P) WCAP-8028-A	"TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code"	Rev. 0	15.0, 15.4	1/7/75	A
WCAP-8054 (P) WCAP-8195	"Application of the THINC-IV Program to PWR Design"	Rev. 0	4.4	12/7/73 1/11/74	A
WCAP-8082-P-A(P) WCAP-8172-A	"Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop"	Rev. 0	3.6.3	1/16/75	A
WCAP-8099	"A Summary Analysis of the April 30 Incident at the San Onofre Nuclear Generation Station, Unit 1"	Rev. 0		4/20/73	B
WCAP-8163	"Reactor Coolant Pump Integrity in LOCA"	Rev. 0	3A, 5.4	9/20/73	U

TABLE 1.6-2 (Sheet 10)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8170 (P) WCAP-8171	"Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)"	Rev. 0	15.6	7/3/74	AE
WCAP-8183	"Operational Experience with Westinghouse Cores (up to December 31, 1977)"	Rev. 7	4.2	4/20/78	B
WCAP-8200(P) WCAP-8261	"WFLASH - A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR"	Rev. 2 Rev. 1	15.6	7/3/74	AE
WCAP-8218 P-A (P) WCAP-8219-A	"Fuel Densification Experimental Results and Model for Reactor Application"		See WCAP-10851-P-A.		
WCAP-8236 (P) WCAP-8288	"Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident"	Rev. 0	1.5, 4.2	2/28/74 3/1/74	U
WCAP-8236 (P) WCAP-8288	"Safety Analysis of the 8-Grid 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident"	Addendum 1	3.7(N)	4/15/74	A
WCAP-8252	"Documentation of Selected Westinghouse Structural Analysis Computer Codes"	Rev. 1	3.6.3, 3.9(N)	7/19/77	U

TABLE 1.6-2 (Sheet 11)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8253	"Source Term Data for Westinghouse Pressurized Water Reactors"	Amendment 1		2/13/76	B
WCAP-8255	"Nuclear Instrumentation System"	Rev. 0	7.2, 7.7	4/9/74	B
WCAP-8278 (P) WCAP-8279	"Hydraulic Flow Test of the 17 x 17 Fuel Assembly"	Rev. 0	1.5, 4.2, 4.4	2/28/74 3/1/74	U
WCAP-8289 (P) WCAP-8290	"17 x 17 Design Fuel Rod Behavior During Simulated Loss-of-Coolant Accident Conditions"	Rev. 0	1.5	11/18/74	A
WCAP-8296-P-A(P) WCAP-8297-A	"Effect of 17 x 17 Fue Assembly Geometry on DNB"	Rev. 0	1.5	2/6/75	A
WCAP-8298-P-A(P) WCAP-8299-A	"The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing"	Rev. 0	4.4	1/28/75	A
WCAP-8301(P) WCAP-8305	"LOCTA-IV Program: Loss-of-Coolant Transient Analysis"	Rev. 0	15.0, 15.6	7/12/74	AE
WCAP-8302 (P) WCAP-8306	"SATAN-VI Program: Comprehensive Space-Time DependentAnalysis of Loss-of-Coolant"	Rev. 0	15.0, 15.6	7/12/74	AE
WCAP-8303-P-A(P) WCAP-8317-A	"Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests"	Rev. 0	3.9(N)	7/18/75	A

TABLE 1.6-2 (Sheet 12)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8324-A	"Control of Delta Ferrite in Austenitic Stainless Steel Weldments"	Rev. 0	5.2	6/23/75	A
WCAP-8327 (P) WCAP-8326	"Containment Pressure Analysis Code (COCO)"	Rev. 0	15.6	7/3/74	AE
WCAP-8330	"Westinghouse Anticipated Transients Without Trip Analysis"	Rev. 0	4.3, 4.6	9/25/74	U
WCAP-8339	"Westinghouse Emergency Core Cooling System Evaluation Model - Summary"	Rev. 0	15.6	7/3/74	AE
WCAP-8340 (P) WCAP-8356	"Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies"	Rev. 0	15.6	8/1/74	AE
WCAP-8341 (P) WCAP-8342	"Westinghouse Emergency Core Cooling System Evaluation Model-Sensitivity Studies"	Rev. 0	15.6	7/3/74	AE
WCAP-8359	"Effects of Fuel Densification Power Spikes on Clad Thermal Transients"	Rev. 0	4.3	8/2/74	A
WCAP-8370	"Quality Assurance Plan Westinghouse Nuclear Energy Systems Divisions"	Rev. 7A	3A	2/5/75	A
WCAP-8370	"Westinghouse Water Reactor Divisions Quality Assurance Plan"	Rev. 8A	3A	11/14/77	A

TABLE 1.6-2 (Sheet 13)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8370	"Westinghouse Water Reactor Divisions Quality Assurance Plan"	Rev. 9A	3A		
WCAP-8373	"Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974"	Rev. 0	3.10(N)	8/23/74	U
WCAP-8377 (P) WCAP-8381	"Revised Clad Flattening Model"	Rev. 0	4.2	8/7/74 8/6/74	A
WCAP-8385 (P) WCAP-8403	"Power Distribution Control and Load Following Procedures"	Rev. 0	4.3, 4.4	10/9/74	A
WCAP-8424	"An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs"	Rev. 1	15.3	5/30/75	U
WCAP-8446 (P) WCAP-8449	"17 x 17 Driveline Components Tests Phase IB,II, III D-Loop Drop and Deflection"	Rev. 0	1.5, 3.9(N)	12/31/74	A
WCAP-8453-A	"Analysis of Data from the Zion (Unit 1) THINC Verification Test"	Rev. 0	4.4	5/10/76	A
WCAP-8471 (P) WCAP-8472	"Westinghouse ECCS Evaluation Model - Supplementary Information"	Rev. 0	15.6	2/10/75 2/11/75	AE

TABLE 1.6-2 (Sheet 14)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[(1)] Status
WCAP-8485	"Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Fall 1974"	Rev. 0		4/2/75	B
WCAP-8498	"Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors, Program Summaries - Fall 1974"	Rev. 0	4.3	7/22/75	U
WCAP-8510	"Method for Fracture Mechanics Analysis of Nuclear Reactor Vessels Under Severe Thermal Transients"		5.3	12/75	
WCAP-8516-P(P) WCAP-8517	UHI Plant Internals Vibration Measurement Program and Pre and Post Hot Functional Examinations"	Rev. 0	3.9(N)	4/11/75	A
WCAP-8536(P) WCAP-8537	"Critical Heat Flux Testing of 17 x 17 Fuel Assembly Geometry with 22-Inch Grid Spacing"	Rev. 0	4.4	5/30/75	A
WCAP-8565-P-A(P) WCAP-8566-A	"Westinghouse ECCS-Four Loop Plant (17 x 17) Sensitivity Studies"	Rev. 0	15.6	7/17/75	A
WCAP-8577	"The Application of Preheat Temperatures after Welding Pressure Vessel Steels"	Rev. 0	6.1	2/3/76	A

TABLE 1.6-2 (Sheet 15)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8584 (P) WCAP-8760	"Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Features Actuation System"	Rev. 1	4.6, 7.3	3/20/80	U
WCAP-8587	"Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment"	Rev. 6A	3.10(N), 3.11(N), 3A	11/83	U
WCAP-8587	"Equipment Qualification Data Packages"	Rev. 1 Supp. 1	3.11(N)	4/17/78	U
WCAP-8622 (P) WCAP-8623	"Westinghouse ECCS Evaluation Model - October 1975 Version"	Rev. 0	1.5, 15.6	11/20/75	AE
WCAP-8624(P)	"General Method of Developing Multi-Frequency Biaxial Test Inputs for Bistables"		3.10(N)		
WCAP-8682 (P) WCAP-8683	"Experimental Verification of Wet Fuel Storage Criticality Analyses"	Rev. 0	4.3	3/18/76	B
WCAP-8691 (P) WCAP-8692	"Fuel Rod Bow Evaluation"	Rev. 0	4.2, 4.4	1/9/76	U
WCAP-8693	"Delta Ferrite in Production Austenitic Stainless Steel Weldments"	Rev. 0	5.2	3/16/76	B

TABLE 1.6-2 (Sheet 16)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8708-P-A (P), Volumes I and II WCAP-8709-A, Volumes I and II	"MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics"	Rev. 0	3.6.3, 3.9(N)	9/16/77	A
WCAP-8720 (P) WCAP-8785	"Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations"		See WCAP-10851-P-A.		
WCAP-8768	"Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 - Summer 1978"	Rev. 2	1.5, 4.2, 4.3, 5.4	9/28/78	B
WCAP-8766 (P) WCAP-8780	"Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant"	Rev. 0	3.9(N)	5/21/76	A
WCAP-8865-A	"Westinghouse ECCS - Four Loop Plant (17 x 17) Sensitivity Studies with Upper Head Fluid Temperature at T_{hot}"	Rev. 0		5/6/77	A

TABLE 1.6-2 (Sheet 17)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-8872	"Design, Inspection, Operation and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures as Low as Reasonably Achievable."	Rev. 0	12.1	4/27/77	B
WCAP-8892-A	"Westinghouse 7300 Series Process Control System Noise Tests"	Rev. 0	7.1	6/15/77	A
WCAP-8929	"Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program"	Rev. 0	3.9(N)	5/26/77	U
WCAP-8963 (P) WCAP-8964	"Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis"	Rev. 0	4.2	3/31/71 8/11/77	A
WCAP-8970(P) WCAP-8971	"Westinghouse Emergency Core Cooling System Small Break October 1975 Model"		15.6	4/77	
WCAP-8976	"Failure Mode and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System"	Rev. 0	4.6, 7.7	10/26/77	U

TABLE 1.6-2 (Sheet 18)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-9166	"Westinghouse Emergency Core Cooling System Evaluation Model for Large LOCAs Driving Operation With One Loop Out of Service for Plants Without Loop Isolation Valves"		15.6	2/78	
WCAP-9168 (P) WCAP-9169	"Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version"	Rev. 0	15.6	9/27/77	U
WCAP-9179 (P) WCAP-9224	"Properties of Fuel and Core Component Materials"	Rev. 1	4.2	8/2/78 10/80	U
WCAP-9207(P) WCAP-8966	"Evaluation of Mispositioned ECCS Valves"	Rev. 0	6.3	3/21/78	U
WCAP-9220-P-A(P) WCAP-9221-P-A	"Westinghouse ECCS Evaluation Model-February 1978 Version"		15.6	2/78	
WCAP-9226(P) WCAP-9227	"Reactor Core Response to Excessive Secondary Steam Releases"		15.1	7/78	
WCAP-9230(P) WCAP-9231	"Report on the Consequences of a Postulated Main Feedline Rupture"	Rev. 0	15.2	1/27/78	U

TABLE 1.6-2 (Sheet 19)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-9279	"Combination of Safe Shutdown Earthquake and Loss-of-Coolant Accident Responses for Faulted Condition Evaluation of Nuclear Power Plants"	Rev. 0	3.9(N)	3/21/78	U
WCAP-9283	"Integrity of the Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events"	Rev. 0	3.9(N)	3/21/78	U
WCAP-9292	"Dynamic Fracture Toughness of ASME SA508 Class 2a and ASME SA533 Grade A Class 2 Base and Heat Affected Zone Material and Applicable Weld Metals"	Rev. 0	5.2	3/17/78	U
WCAP-9346	"Electric Hydrogen Recombiner Qualification Testing for Model B"		6.2.5	7/78	
WCAP-9714(P) WCAP-9750-A	"Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equipment"		3.10(N)		
WCAP-9944(P) WCAP-9945	"Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests on Sequoyah 1 Power Plant"		3.9(N)	7/81	

TABLE 1.6-2 (Sheet 20)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-10043	Steam Generator Tube Plugging Analysis for the Westinghouse Standardized Nuclear Power Plant (P) System	Rev. 0	5.4.2.5	12/3/82	U
WCAP-10297-P-A	"Dropped Rod Methodology for Negative Flux Rate Trip Plants"		15.4	6/83	
WCAP-10851-P-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations	Rev. 0	3A, 4.2, 4.3, 4.4	6/85	A
WCAP-10961-P	Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment	Rev. 1	3B.4.2	1/17/86	A
SCP-97-116	Releases for Equipment Environmental Qualification Outside Containment				
WCAP-12472-P-A	BEACON Core Monitoring and Operations Support System	Rev. 0	4.3.2.2.7	4/90	A
WCAP-12476[3]	Evaluation of LOCA During Mode 3 and Mode 4 Operation for Westinghouse NSSS	Rev. 1	Table 15.0-8	11/27/91	Not approved-withdrawn on 4/28/99

TABLE 1.6-2 (Sheet 21)

Westinghouse Topical Report No.	Title	Revision Number	FSAR Section Reference	Submitted to the NRC	Review[1] Status
WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves	Rev. 4	5.3.1.6.1		A
WCAP-15151	Westinghouse Archived Reactor Vessel Materials	12/98	5.3.1.6.1		
WCAP-12472-P-A	Addendum 1-A	Rev. 0	4.3.2.2.7	1/00	A
WCAP-15400	Analysis of Capsule X from Callaway Unit 1 Reactor Vessel Radiation Surveillance Program	6/00	5.3.4 Table 5.3-10		A
WCAP-14565-P-A Addendum 2-P-A	Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications	4/08	4.4.1.1		A

TABLE 1.6-2 (Sheet 22)

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(1) A legend to the review status code letters follows:

A - NRC review complete; NRC acceptance letter issued.

AE - NRC accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.

B - Submitted to NRC as background information; not undergoing formal NRC review.

O - On file with NRC; older generation report with current validity; not actively under formal NRC review.

U - Actively under formal NRC review.

(2) Portions of WCAP 7769 are superceded by the Callaway overpressure protection report included in Westinghouse letter SCP 94-143 dated 8/30/94. This report was reviewed by the NRC during the review of OL-1186, Amendment 128, MSSV tolerance change.

(3) Westinghouse letter SCP-10-31, “Transmittal of Mode 4 Small Break LOCA (SBLOCA) RHR Flow Evaluation for Callaway (SCP), Phase 3, Rev. 1,” dated May 11, 2010, provides a Callaway-specific evaulation of WCAP-12476.

(P) - Proprietary

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The engineering drawings listed in Tables 1.7-1, 1.7-2 and 1.7-3 reflect the detailed design configuration as described in FSAR text and tables. Earlier revisions of engineering drawings may be used as FSAR figures when they do not affect the safety review.

1.7.1 Electrical, Instrumentation and Control Drawings

Table 1.7-1 has been deleted.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of each piping and instrumentation diagram and the corresponding FSAR figure number as it appears at the end of the respective text section. The P&ID legend, Figure 1.1-1, provides an explanation of Bechtel symbols and characters used in these FSAR figures.

Various valves and dampers are designated as being normally locked in position (locked open, locked closed, locked throttled, etc.) on the P&ID's. These valves or dampers may be required to be in the designated locked position to support various analyses. For this reason, the normally locked position designations on P&ID's are considered to be part of the description of the plant in this safety analysis report. Otherewise, the position of a particular valve or damper may vary depending on plant or system conditions. The functional level of detail regarding valve or damper positions is described, as needed, in sections (text, tables, or figures) of the FSAR other than the P&ID. Thus, a change to a valve or damper normal position designation on a P&ID is not considered to be a change to the plant as described in the FSAR provided that it is consistent with other descriptions in the FSAR, does not involve a locked position valve or damper, and is not credited in other FSAR sections as describing the position or lineup. Adding a normally locked position designation to a passive valve or damper that previously was not designated as locked would not be a change to the FSAR description.

1.7.3 Other Drawings and Diagrams

Table 1.7-3 contains a list of electrical, instrumentation, and control drawings and piping and instrumentation diagrams for site safety-related systems.

TABLE 1.7-1 DELETED

The information provided in Table 1.7-1 has been deleted. This table provided a listing of the electrical, instrumentation, and control drawings that were provided to the NRC for their use in evaluating the offsite and onsite electrical power system designs to see if they met the requirements of the Standard Review Plan. This Table provided information that has been determined to be historical and is no longer needed in the FSAR.

TABLE 1.7-2 PIPING AND INSTRUMENTATION DIAGRAMS

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-220101	1.1-1	1	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams	6
10466-M-220102	1.1-1	2	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams	5
10466-M-220103	1.1-1	3	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams	3
10466-M-220104	1.1-1	4	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams	5
10466-M-22AB01(Q)	10.3-1	1	Main Steam System	15
10466-M-22AB02(Q)	10.3-1	2	Main Steam System	14
10466-M-22AB03	10.3-1	3	Main Steam System	17
10466-M-22AB04	10.3-1	4	Main Steam System	9
10466-M-22AB05	10.3-1	5	Main Steam System	3
10466-M-22AB06	10.3-1	6	Main Steam System	3
10466-M-22AB07	10.3-1	7	Main Steam System	3
10466-M-22AC01	10.2-1	1	Main Turbine	12
10466-M-22AC02	10.2-1	2	Main Turbine	16
10466-M-22AC03	10.2-1	3	Main Turbine	16
10466-M-22AC04	10.2-1	4	Main Turbine	14
10466-M-22AD01	10.4-2	1	Condensate System	16
10466-M-22AD02	10.4-2	2	Condensate System	31
10466-M-22AD03	10.4-2	3	Condensate System	9
10466-M-22AD04	10.4-2	4	Condensate System	3
10466-M-22AD05	10.4-2	5	Condensate System	8
10466-M-22AD06	10.4-2	6	Condensate System	12
10466-M-22AD07	10.4-2	7	Condensate System	12
10466-M-22AD08	10.4-2	8	Condensate System	16
10466-M-22AE01	10.4-6	1	Feedwater System	42
10466-M-22AE02(Q)	10.4-6	2	Feedwater System	24
10466-M-22AF01	10.4-6	3	Feedwater Heater Extraction Drains & Vents	32
10466-M-22AF02	10.4-6	4	Feedwater Heater Extraction Drains & Vents	40
10466-M-22AF02A	10.4-6	10	Feedwater Heater Extraction Drains & Vents	1
10466-M-22AF02B	10.4-6	11	Feedwater Heater Extraction Drains & Vents	3
10466-M-22AF02C	10.4-6	9	Feedwater Heater Extraction Drains & Vents	3
10466-M-22AF03	10.4-6	5	Feedwater Heater Extraction Drains & Vents	20
10466-M-22AF04	10.4-6	6	Feedwater Heater Extraction Drains & Vents	10
10466-M-22AK01	10.4-5	1	Condensate Demineralizer System	25
10466-M-22AK02	10.4-5	2	Condensate Demineralizer System	14

TABLE 1.7-2 (Sheet 2)

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-22AK03	10.4-5	3	Condensate Demineralizer System	23
10466-M-22AK04	10.4-5	4	Condensate Demineralizer System	9
10466-M-22AL01(Q)	10.4-9		Auxiliary Feedwater System	32
10466-M-22AN01	9.2-4		Demineralized Water Storage and Transfer System	34
10466-M-22AP01	9.2-12		Condensate Storage and Transfer System	22
10466-M-22AQ01	10.4-7	1	Condensate Chemical Addition System	15
10466-M-22AQ02	10.4-7	2	Feedwater Chemical Addition System	12
10466-M-22BB01(Q)	5.1-1	1	Reactor Coolant System	26
10466-M-22BB02(Q)	5.1-1	2	Reactor Coolant System	26
10466-M-22BB03(Q)	5.1-1	3	Reactor Coolant System	13
10466-M-22BB03A	5.1-1	3A	Reactor Coolant System	9
10466-M-22BB03B	5.1-1	3B	Reactor Coolant System	8
10466-M-22BB03C	5.1-1	3C	Reactor Coolant System	7
10466-M-22BB03D	5.1-1	3D	Reactor Coolant System	6
10466-M-22BB04(Q)	5.1-1	4	Reactor Coolant System	15
10466-M-22BG01(Q)	9.3-8	1	Chemical and Volume Control System	27
10466-M-22BG02(Q)	9.3-8	2	Chemical and Volume Control System	18
10466-M-22BG03(Q)	9.3-8	3	Chemical and Volume Control System	50
10466-M-22BG04(Q)	9.3-8	4	Chemical and Volume Control System	20
10466-M-22BG05(Q)	9.3-8	5	Chemical and Volume Control System	22
10466-M-22BL01(Q)	9.2-13		Reactor Makeup Water System	21
10466-M-22BM01(Q)	10.4-8	1	Steam Generator Blowdown System	31
10466-M-22BM02	10.4-8	2	Steam Generator Blowdown System	15
10466-M-22BM03	10.4-8	3	Steam Generator Blowdown System	5
10466-M-22BM04	10.4-8	4	Steam Generator Blowdown System	6
10466-M-22BM05	10.4-8	5	Steam Generator Blowdown System	3
10466-M-22BN01(Q)	6.3-1	1	Borated Refueling Water Storage System	24
10466-M-22CA01	10.4-4		Steam Seal System	3
10466-M-22CB01	10.2-1	7	Turbine Lube Oil System	3
10466-M-22CC01	10.2-1	8	Generator System	7
10466-M-22CD01	10.2-1	9	Shaft Sealing System	6
10466-M-22CE01	10.2-1	10	Generator Stator Cooling Storage System	11
10466-M-22CF01	10.2-1	5	Lube Oil Storage, Transfer and Purification System	9
10466-M-22CF02	10.2-1	6	Lube Oil Storage, Transfer and Purification System	2
10466-M-22CG01	10.4-3	1	Condenser Air Removal	11
10466-M-22CH01	10.2-1	11	Turbine Hydraulic Control System	6
10466-M-22DA01	10.4-1	1	Circulating Water & Water Box Drains System	12

TABLE 1.7-2 (Sheet 3)

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-22DA02	10.4-1	3	Circulating Water & Water Box Venting System	15
10466-M-22EA01	9.2-1	1	Service Water System	20
10466-M-22EA02	9.2-1	2	Service Water System	25
10466-M-22EB01	9.2-14		Closed Cooling Water System	16
10466-M-22EC01(Q)	9.1-3	1	Fuel Pool Cooling and Cleanup System	19
10466-M-22EC02(Q)	9.1-3	2	Fuel Pool Cooling and Cleanup System	23
10466-M-22EF01(Q)	9.2-2	1	Essential Service Water System	40
10466-M-22EF02(Q)	9.2-2	2	Essential Service Water System	47
10884-M-U2EF01(Q)	9.2-2	3	Essential Service Water System	46
10466-M-22EG01(Q)	9.2-3	1	Component Cooling Water System	6
10466-M-22EG02(Q)	9.2-3	2	Component Cooling Water System	14
10466-M-22EG03(Q)	9.2-3	3	Component Cooling Water System	19
10466-M-22EJ01(Q)	5.4-7		Residual Heat Removal System	52
10466-M-22EM01(Q)	6.3-1	2	High Pressure Coolant Injection System	30
10466-M-22EM02(Q)	6.3-1	3	High Pressure Coolant Injection System	18
10466-M-22EM03	6.3-1	4	High Pressure Coolant Injection System	12
10466-M-22EN01(Q)	6.2.2-1		Containment Spray System	9
10466-M-22EP01(Q)	6.3-1	5	Accumulator Safety Injection	14
10466-M-22FA01	9.5.9-1	1	Auxiliary Boiler System	14
10466-M-22FB01	9.5.9-1	2	Auxiliary Steam System	16
10466-M-22FB02	9.5.9-1	3	Auxiliary Steam System	10
10466-M-22FC02(Q)	10.4-10		Auxiliary Feedwater Pump Turbine	20
10466-M-22FC03	10.4-6	7	S.G.F.P. Turbine "A"	14
10466-M-22FC04	10.4-6	8	S.G.F.P. Turbine "B"	16
10466-M-22FE01	9.5.9-1	4	Auxiliary Steam Chemical Addition System	4
10466-M-22GA01	9.4-9	1	Plant Heating System	8
10466-M-22GA02	9.4-9	2	Plant Heating System	3
10466-M-22GB01	9.4-10		Central Chilled Water System	17
10884-M-U2GD01(Q)	9.4-8	2	ESW Pump House & UHS Elec. Room HVAC	9
10466-M-22GE01	9.4-4	1	Turbine Building HVAC	10
10466-M-22GE02(Q)	9.4-4	2	Turbine Building HVAC	12
10466-M-22GE03	9.4-4	3	Turbine Building HVAC	10
10466-M-22GE04	9.4-4	4	Turbine Building HVAC	1
10466-M-22GE05	9.4-4	5	Turbine Building HVAC	6
10466-M-22GF01(Q)	9.4-3	1	Miscellaneous Buildings HVAC	9
10466-M-22GF02	9.4-3	2	Miscellaneous Buildings HVAC	3
10466-M-22GG01(Q)	9.4-2	1	Fuel Building HVAC	12

TABLE 1.7-2 (Sheet 4)

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-22GG02(Q)	9.4-2	2	Fuel Building HVAC	9
10466-M-22GH01	9.4-5	1	Radwaste Building HVAC	7
10466-M-22GH02	9.4-5	2	Radwaste Building HVAC	12
10466-M-22GK01(Q)	9.4-1	1	Control Building HVAC	12
10466-M-22GK02(Q)	9.4-1	2	Control Building HVAC	16
10466-M-22GK03(Q)	9.4-1	3	Control Building HVAC	18
10466-M-22GK04(Q)	9.4-1	4	Control Building HVAC	16
10466-M-22GL01(Q)	9.4-3	3	Auxiliary Building HVAC	29
10466-M-22GL02(Q)	9.4-3	4	Auxiliary Building HVAC	23
10466-M-22GL03(Q)	9.4-3	5	Auxiliary Building HVAC	15
10466-M-22GM01(Q)	9.4-7		Diesel Generators Building HVAC	1
10466-M-22GN01(Q)	9.4-6	1	Containment Cooling System	17
10466-M-22GN02	9.4-6	2	Containment Cooling System	5
10466-M-22GP01(Q)	6.2.6-1		Containment Integrated Leak Rate Test	9
10466-M-22GS01(Q)	6.2.5-1		Containment Hydrogen Control System	7
10466-M-22GT01(Q)	9.4-6	4	Containment Purge System HVAC	23
10466-M-22HA01	11.3-1	1	Gaseous Radwaste System	7
10466-M-22HA02	11.3-1	2	Gaseous Radwaste System	8
10466-M-22HA03	11.3-1	3	Gaseous Radwaste System	6
10466-M-22HB01(Q)	11.2-1	1	Liquid Radwaste System	31
10466-M-22HB02	11.2-1	2	Liquid Radwaste System	18
10466-M-22HB03	11.2-1	3	Liquid Radwaste System	12
10466-M-22HB04	11.2-1	4	Liquid Radwaste System	13
10466-M-22HB05	11.2-1	5	Liquid Radwaste System	20
10466-M-22HC01	11.4-1	1	Solid Radwaste System	31
10466-M-22HC02	11.4-1	2	Solid Radwaste System	6
10466-M-22HC03	11.4-1	3	Solid Radwaste System	11
10466-M-22HC04	11.4-1	4	Solid Radwaste System	10
10466-M-22HC05	11.4-1	5	Solid Radwaste System	11
10466-M-22HD01(Q)	12.3-4		Decontamination System	8
10466-M-22HE01	9.3-11	1	Boron Recycle System	11
10466-M-22HE02	9.3-11	2	Boron Recycle System	13
10466-M-22HE03	9.3-11	3	Boron Recycle System	10
10466-M-22HF01	10.4-12	1	Secondary Liquid Waste System	14
10466-M-22HF02	10.4-12	2	Secondary Liquid Waste System	18
10466-M-22HF03	10.4-12	3	Secondary Liquid Waste System	22
10466-M-22HF04	10.4-12	4	Secondary Liquid Waste System	16

TABLE 1.7-2 (Sheet 5)

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-22HF05	10.4-12	5	Hollow Fiber Filter System	9
10466-M-22JE01(Q)	9.5.4-1		Emergency Fuel Oil System	17
10466-M-22KA01(Q)	9.3-1	1	Compressed Air System	28
10466-M-22KA02(Q)	9.3-1	2	Compressed Air System (Service Air)	24
10466-M-22KA03	9.3-1	3	Instrument Air System	12
10466-M-22KA04	9.3-1	4	Instrument Air System	10
10466-M-22KA05(Q)	9.3-1	5	Compressed Air System	13
10466-M-22KA06	9.3-1	6	Compressed Air System	15
10466-M-22KA07	9.3-1	7	Compressed Air System	4
10466-M-22KA08	9.3-1	8	Instrument Air System	11
10466-M-22KA09	9.3-1	9	Instrument Air System	19
10466-M-22KB01	9.5.10-1	1	Breathing Air System	4
10466-M-22KB02	9.5.10-1	2	Breathing Air System	14
10466-M-22KB03	9.5.10-1	3	Breathing Air System	7
10466-M-22KD01	9.2-5	1	Domestic Water System	17
10466-M-22KD02	9.2-5	2	Domestic Water System	30
10466-M-22KH01	9.3-9	1	Service Gas System	22
10466-M-22KH02	9.3-9	2	Service Gas System	9
10466-M-22KJ01(Q)	9.5.5-1	1	Standby Diesel Generator "A" Cooling Water System	18
10466-M-22KJ04(Q)	9.5.5-1	2	Standby Diesel Generator "B" Cooling Water System	17
10466-M-22KJ02(Q)	9.5.6-1	1	SDG "A" Intake, Exh., F.O. & Starting Air System	17
10466-M-22KJ05(Q)	9.5.6-1	2	SDG "B" Intake, Exh., F.O. & Starting Air System	20
10466-M-22KJ03(Q)	9.5.7-1	1	Standby Diesel Generator "A" Lube Oil System	18
10466-M-22KJ06(Q)	9.5.7-1	2	Standby Diesel Generator "B" Lube Oil System	17
10466-M-22LA01	9.3-5	1	Sanitary Lift Station & Turb. Bldg. Sanitary Drainage System	2
10466-M-22LA02	9.3-5	2	Comm. Corridor & Control Bldg. Sanitary Drainage System	4
10466-M-22LD01	9.3-5	3	Chemical and Detergent Waste	10
10466-M-22LE01	9.3-5	4	Turb. Bldg. & Aux. Feedwater Pump Rooms Oily Waste System	8
10466-M-22LE02(Q)	9.3-5	5	Control & Diesel Gen. Bldg. Oily Waste System	3
10466-M-22LE03	9.3-5	6	Turb. Bldg. & Aux. Boiler Room Oily Waste System	3
10466-M-22LE04	9.3-5	7	Tendon Access Gallery & Turb. Bldg. Oily Waste System	5
10466-M-22LF01(Q)	9.3-5	8	Aux. Bldg. Floor and Equipment Drain System	5
10466-M-22LF02	9.3-5	9	Aux. Bldg. Floor And Equipment Drain System	3
10466-M-22LF03(Q)	9.3-5	10	Aux. Bldg. Floor and Equipment Drain System	5
10466-M-22LF04	9.3-5	11	Aux. Bldg. Floor and Equipment Drain System	5
10466-M-22LF05	9.3-5	12	Aux. Bldg. Floor and Equipment Drain System	20

TABLE 1.7-2 (Sheet 6)

Drawing Number	Figure Number	Sheet	Title	Rev.
10466-M-22LF06	9.3-5	13	Radwaste & Fuel Bldgs. FED System	14
10466-M-22LF07	9.3-5	14	Radwaste Bldg. FED System	3
10466-M-22LF08(Q)	9.3-5	15	Control and Fuel Bldgs. FED System	5
10466-M-22LF09(Q)	9.3-5	16	Reactor Bldg. & Laundry Decon Facility FED System	18
10466-M-22LF10	9.3-5	17	Radwaste Bldg. and Tunnel FED System	9
10466-M-22RM01	9.3-4	1	Process Sampling System	23
10466-M-22RM02	9.3-4	2	Process Sampling System	24
10466-M-22RM03	9.3-4	3	Process Sampling System	12
10466-M-22SJ01(Q)	9.3-2	1	Nuclear Sampling System Primary Sampling System	18
10466-M-22SJ02	9.3-3		Nuclear Sampling System Radwaste Sampling System	4
10466-M-22SJ03	9.3-2	2	Nuclear Sampling System Primary Sampling System	1
10466-M-22SJ04	18.2-15		Nuclear Sampling System Primary Sampling System	13

TABLE 1.7-3 SITE ELECTRICAL, PIPING, INSTRUMENTATION, AND CONTROL DRAWINGS

Title	Drawing No.
Single Line Diagram Essential Service Water System	E-U1001(Q)
Low Voltage System, Class 1E 480V. Single Line, Meter & Relay Diagram	E-U1NG01(Q)
Motor Control Center Summary	E-U1NG20(Q)
Low Voltage Sys Class 1E Motor Cost Cent Summary	E-U1NG21(Q)
Low Voltage System, Class 1E 480V. Three Line, Meter & Relay Diagram	E-U3NG01(Q)
Low Voltage System, Class 1E 480V. Three Line, Meter & Relay Diagram	E-U3NG02(Q)
Schematic Diagram 4.16kV XFMR Feeder Brkrs.	E-U3NG10(Q)
Schematic Diagram 4.16kV XFMR Feeder Brkrs.	E-U3NG11(Q)
Schematic Diagram Essential Service Water Pumps	E-U3EF01(Q)
Schematic Diagram Ultimate Heat Sink Clng. Twr. Fans	E-U3EF02A(Q)
Schematic Diagram UHS Clng. Twr. Fans Speed Selection	E-U3EF02B(Q)
Schematic Diagram UHS Clng. Twr. Fans Manual Control	E-U3EF02C(Q)
Schematic Diagram UHS Cooling Tower Fans	E-U3EF02D(Q)
Schematic Diagram ESW Self-Cleaning Strainers	E-U3EF03(Q)
Schematic Diagram Self-Cleaning Strainer, Trash Rack	E-U3EF04(Q)
Schematic Diagram Cooling Tower Inlet By-Pass Valves	E-U3EF05(Q)
Schematic Diagram ESW Pump Interposing Relays	E-U3EF06(Q)
Schematic Diagram ESW Aux. Relays, Inlet Clng. Twr. By-Pass Vlvs.	E-U3EF07(Q)

TABLE 1.7-3 (Sheet 2)

Title	Drawing No.
Schematic Diagram Cooling Tower Trouble Alarm	E-U3EF08(Q)
Schematic Diagram Miscellaneous Circuits	E-U3EF09(Q)
Schematic Diagram Miscellaneous Circuits	E-U3EF10(Q)
Schematic Diagram ESW Pump Disch. Line Air Disch. Valves	E-U3EF11(Q)
Schematic Diagram ESW Pump Room Supply Fans	E-U3GD01(Q)
Schematic Diagram ESW Electrical Room Supply Fans	E-U3GD02(Q)
Schematic Diagram ESW Pump Room & Elec. Room Unit Heaters	E-U3GD03(Q)
Schematic Diagram ESW Pump & Elec. Room Exhaust Valves	E-U3GD04(Q)
Schematic Diagram ESW Pump & Elec. Rooms, Miscellaneous Circuits	E-U3GD05(Q)
Schematic Diagram Lighting System Power Feeders	E-U3QB01(Q)
Essential Service Water System	M-U2EF01(Q)

Title	Drawing No.	Figure No.
Essential Service Water System	M-U2EF01(Q)	9.2-2
Essential Service Water Pumphouse HVAC	M-U2GD01(Q)	Std. Plt. 9.4-8, sht. 2

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

A discussion of the extent to which Union Electric complies with each of the NRC Division 1 Regulatory Guides is provided in Appendix 3A. Appendix 3A gives a brief statement of compliance and refers to the most appropriate section of the FSAR for the complete description of how the design complies with the regulatory recommendations.

1.9 RC REGULATORY REQUIREMENTS REVIEW COMMITTEE CATEGORY 2, 3, AND 4 MATTERS

The Office of Nuclear Reactor Regulation (NRR) has established a Regulatory Requirements Review Committee (RRRC) which reviews proposed changes to the regulatory requirements issued by the staff and recommends a course of action to the Office of NRR. The course of action includes an implementation schedule. The Director's approval is then used by the NRR staff as review guidance on individual licensing matters.

The RRRC has developed a categorization nomenclature to aid in the uniform implementation of new and revised regulatory staff concerns. The system includes four categories (1 through 4) which correspond to the evaluation by the RRRC of the need for applying the regulatory concerns to new and ongoing license applications. The four categories can be defined as follows:

Category 1: Matters whose applicability is to be applied to applications in accordance with the implementation section of the published guide. The RRRC considers it necessary to forward fit (on new applications) the requirements of these matters.

Category 2: A new position whose applicability is to be determined on a case-by-case basis. The NRC staff will give further consideration to the need for backfitting certain identified items of the regulatory concerns.

Category 3: Positions to which the NRC staff considers conformance necessary, either by direct implementation or by implementation of an acceptable alternative. These positions could be the cause of backfitting if an acceptable alternative is not available.

Category 4: Positions of concern to the NRC staff which have not been reviewed by the RRRC and subsequently categorized as Category 1, 2, or 3. Since these items are of concern to the NRC staff, for review purposes, they are to be considered on the same basis as Category 2, potential for backfitting certain identified regulatory concerns.

The Office of NRR, by letter dated November 21, 1978, transmitted to Union Electric a list of Category 2, 3, and 4 matters for consideration in preparing the FSAR. A discussion of how SNUPPS complies with each of the listed matters is contained in Tables 1.9-1 through 1.9-4. Since the NRC listings are expected to be revised periodically, the RRRC Category Designation columns in the tables correspond to those contained in the November 21, 1978 letter.

Table 1.9-1 - Lists all Category 2, 3, and 4 regulatory guides and references the location in which the regulatory guides are addressed.

Table 1.9-2 - Lists all Category 2, 3, and 4 branch technical positions (BTPs), provides remarks on the extent to which the recommendations of the BTPs are met, and references the location of more complete discussions of the RRRC matter.

Tables 1.9-3 and 1.9-4 - Address the Category 4 SRP criteria and other Category 4 positions, respectively.

TABLE 1.9-1 CATEGORY 2, 3, AND 4 REGULATORY GUIDES[1,2]

Regulatory Guide		RRRC Category		
Number	Revision	2	3	4
1.12	1			X
1.13	1			X
1.14	1			X
1.27	2	X		
1.52	1	X		
1.56	1		X	
1.59	2	X		
1.63	2	X		
1.68.2	1		X	
1.75	1			X
1.76	0			X
1.79	1			X
1.80	0			X
1.82	0			X
1.83	1			X
1.89	0			X
1.91	1	X		

1 All regulatory guides are addressed in Appendix 3A with the exception of Regulatory Guide 8.8, which is addressed in Section 12.1.

2 Revision numbers listed on this table are those listed in the November 21, 1978 letter from the Office of NRR to Union Electric.

TABLE 1.9-1 (Sheet 2)

Regulatory Guide		RRRC Category		
Number	Revision	2	3	4
1.93	0			X
1.97	1		X	
1.99	1		X	
1.101	1		X	
1.102	1	X		
1.104	0			X
1.105	1	X		
1.108	1	X		
1.114	1		X	
1.115	1	X		
1.117	1	X		
1.121	0		X	
1.124	1	X		
1.127	1		X	
1.130	0	X		
1.137	0	X		
1.141	0	X		
8.8	2	X		

TABLE 1.9-2 CATEGORY 2, 3, AND 4 BRANCH TECHNICAL POSITIONS

Branch Technical Position	Title	RRRC Cat.	Remarks
MTEB 5-7	Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary	2	The recommendations of this BTP are not applicable to the Callaway (PWR) design.
RSB 5-1, Rev. 1	Design Requirements of the Residual Heat Removal System	3	The recommendations of this BTP are met to the extent described in Appendix 5.4A.
RSB 5-2	Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures	3	The recommendations of this BTP are met to the extent described in Sections 5.2.2 and 7.6.6.
MTEB 5-3	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	4.B.1	The recommendations of this BTP are met. Refer to Sections 9.3.2 and 10.3.5.
CSB 6-1	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	4.B.2	The recommendations of this BTP are met. Refer to Sections 6.2.1 and 15.6.5.
CSB 6-2	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	4.B.3	The recommendations of this BTP are met. Refer to Section 6.2.5.
CSB 6-3	Determination of Bypass Leakage Paths in Dual Containment Plants	4.B.4	The recommendations of this BTP are not applicable to the Callaway design, since there is no dual containment.

TABLE 1.9-2 (Sheet 2)

Branch Technical Position	Title	RRRC Cat.	Remarks
CSB 6-4	Containment Purging During Normal Plant Operations	4.B.5	The recommendations of this BTP are met to the extent described in Table 9.4-13.
ASB 9.1	Overhead Handling Systems for Nuclear Power Plants	4.B.6	The recommendations of this BTP are met. No critical loads are handled. Refer to Section 9.1.4.
ASB 10.1	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR Plants	4.B.7	The recommendations of this BTP are met. Refer to Section 10.4.9.

TABLE 1.9-3 CATEGORY 4 SRP CRITERIA

SRP Section	RRRC Category	Title	Remarks
3.5.3	4.B.8	Procedures for Composite Section Local Damage Prediction (SRP Section 3.5.3, Par. II.1.C)	The recommendations of this SRP are met to the extent described in Section 3.5.3.
3.7.1	4.B.9	Development of Design Time History for Soil-Structure Interaction Analysis (SRP Section 3.7.1, Par. II.2)	The recommendations of this SRP are met to the extent described in Section 3.7(B).1.
3.7.2	4.B.10	Procedures for Seismic System Analysis (SRP Section 3.7.2, Par. II)	The recommendations of this SRP are met. Refer to Sections 3.7(B).2 and 3.7(N).2.
3.7.3	4.B.11	Procedures for Seismic Subsystem Analysis (SRP Section 3.7.3, Par. II)	The recommendations of this SRP are met. Refer to Sections 3.7(B).3 and 3.7(N).3.
3.8.1	4.B.12	Design and Construction of Concrete Containments (SRP Section 3.8.1, Par. II)	The design of the containment structure is described in Section 3.8.1. The load combinations used meet or exceed ACI 359/SRP criteria.
3.8.2	4.B.13	Design and Construction of Steel Containments (SRP Section 3.8.2, Par. II)	The recommendations of this SRP are not applicable to the Callaway Plant.
3.8.3	4.B.14	Structural Design Criteria for Category I Structures Inside Containment (SRP Section 3.8.3, Par. II)	The design meets or exceeds the load combinations of ACI 359/SRP criteria. Refer to Sections 3.8.3 and 5.4, respectively, for discussion of the Bechtel and Westinghouse component supports.

TABLE 1.9-3 (Sheet 2)

SRP Section	RRRC Category	Title	Remarks
3.8.4	4.B.15	Structural Design Criteria for Other Seismic Category I Structures (SRP Section 3.8.4, Par. II)	The design meets or exceeds the load combinations of ACI 359/SRP criteria. Refer to Section 3.8.4.
3.8.5	4.B.16	Structural Design Criteria for Foundations (SRP Section 3.8.5, Par. II)	The design meets or exceeds the load combinations of ACI 359/SRP criteria. Refer to Section 3.8.5. The safety factors for sliding are discussed in Section 3.8.5.5.
3.7 11.2 11.3 11.4	4.B.17	Seismic Design Requirements for Radwaste Systems and their Housing Structures (SRP Section 11.2, BTP ETSB 11-1, Par. B.v.)	The recommendations of this SRP are met as described in Appendix 3A in the response to Regulatory Guide 1.143. Refer to Chapter 11.0. Section 3.8.6 describes the seismic design capabilities of the radwaste building.
3.3.2	4.B.18	Tornado Load Effect Combinations (SRP Section 3.3.2, Par. II.2.d)	The recommendations of this SRP are met. Refer to Section 3.3.2.
3.4.2	4.B.19	Dynamic Effects of Wave Action (SRP Section 3.4.2, Par. II)	The recommendations of this SRP are met. Refer to Section 3.4.2.
10.4.7	4.B.20	Water Hammer for Steam Generators with Preheaters (SRP Section 10.4.7, Par. I.2.b)	The Callaway Plant steam generators (Model 73/19T) have no preheaters. Refer to Sections 5.4.2 and 10.4.7.
4.4	4.B.21	Thermal-Hydraulic Stability (SRP Section 4.4, Par. II.5)	The recommendations of this SRP are met as discussed in Section 4.4.4.6.

TABLE 1.9-3 (Sheet 3)

SRP Section	RRRC Category	Title	Remarks
5.2.5	4.B.22	Intersystem Leakage Detection (SRP Section 5.2.5, Par. II.4 and R.G. 1.45)	Intersystem leakage detection requirements and capabilities are discussed in Section 5.2.5.
3.2.2	4.B.23	Main Steam Isolation Valve Leakage Control System (SRP Section 10.3, Par. III.3 and BTP RSB-3.2)	The recommendations of this SRP are not applicable to the Callaway (PWR) design.

TABLE 1.9-4 OTHER CATEGORY 4 POSITIONS

SRP Section	RRRC Category	Title	Remarks
3.5.3	4.C.1	Ductility of Reinforced Concrete and Steel Structural Elements Subjected to Impactive or Impulsive Loads	The recommendations of this item are met to the extent described in Section 3.5.3.
3.7.1	4.C.2	Response Spectra in Vertical Direction	The recommendations of this item are met. Refer to Section 3.7(B).1. Westinghouse utilizes the damping values of WCAP 7921-AR. See also the response to Regulatory Guide 1.60 in Appendix 3A.
3.8.1 3.8.2	4.C.3	BWR Mark III Containment Pool Dynamics	The recommendations of this item are not applicable to the Callaway (PWR) design.
3.8.4	4.C.4	Air Blast Loads	Air blast loads from transportation are less than the external pressure design capabilities described in Section 3.8.
3.5.3	4.C.5	Tornado Missile Impact	The recommendations of this item are met. Refer to Section 3.5.3.1.
6.3	4.C.6	Passive Failures During Long-Term Cooling Following LOCA	The recommendations of this item are met to the extent described in Sections 3.1 and 6.3.
6.3	4.C.7	Control Room Position Indication of Manual (Handwheel) Valves in the ECCS	The recommendations of this item are met. Refer to Sections 7.5.2.2.1 and 7.5.2.2.2.

TABLE 1.9-4 (Sheet 2)

SRP Section	RRRC Category	Title	Remarks
15.1.5	4.C.8	Long-Term Recovery from Steamline Break: Operator Action to Prevent Overpressurization	The recommendations of this item are met to the extent described in Section 15.0.13.
5.4.6 5.4.7 6.3	4.C.9	Pump Operability Requirements	The recommendations of this item are met. Refer to Section 6.2.2.1.2.2 and Section 6.3.2.5.
3.5.1	4.C.10	Gravity Missiles, Vessel Seal Ring Missiles Inside Containment	The recommendations of this item are met. Refer to Appendix 3B. Section 9.1.4.2.2 discusses the reactor cavity seal ring.
4.4	4.C.11	Core Thermal-Hydraulic Analysis	The recommendations of this item are met. However, Westinghouse is generically reducing rod bow penalties through experience gained by test surveillance. Refer to Section 4.2.3.1.
8.3	4.C.12	Degraded Grid Voltage Conditions	The recommendations of this item are met to the extent described in Section 8.3.1.1.3 and the Tech Specs.
6.2.1.2	4.C.13	Asymmetric Loads on Components Located Within Containment Subcompartments	The recommendations of this item are met. Refer to Section 6.2.1.2.
6.2.6	4.C.14	Containment Leak Testing Program	The recommendations of this item are met. Refer to Section 6.2.6.

TABLE 1.9-4 (Sheet 3)

SRP Section	RRRC Category	Title	Remarks
6.2.1.4	4.C.15	Containment Response Due to Main Steamline Break and Failure of MSLIV to Close	The recommendations of this item are met. Refer to Sections 6.2.1, 3.11(B), and 3.11(N).
3.6.1 3.6.2	4.C.16	Main Steam and Feedwater Pipe Failures	The recommendations of this item are met. Refer to Sections 3.6.1 and 3.6.2.
9.2.2	4.C.17	Design Requirements for Cooling Water to Reactor Coolant Pumps	The recommendations of this item are met to the extent described in Sections 5.4.1, 9.2.2, and 9.3.4
10.4.7	4.C.18	Design Guidelines for Water Hammer in Steam Generators with Top Feedring Design (BTP ASB-10.2)	The design meets the recommendations of this item; however, no testing will be performed. Refer to Section 10.4.7.
3.11	4.C.19	Environmental Control Systems for Safety-Related Equipment	The recommendations of this item are met to the extent described in Section 3.11(B).

1.10 RESPONSES TO RESAR-3 QUESTIONS

The following provides the responses, as applicable to SNUPPS, to the November 17, 1977 NRC questions addressed to all construction permit applications that referenced RESAR-3.

Q31.1
3.10

Section 3.9.1.2 of RESAR-3 states that dynamic testing procedures concerning Westinghouse supplied safety-related mechanical equipment will be provided in the applicant's FSAR. It is our position that as a minimum you commit to conduct a seismic qualification program to conform to the criteria as contained in Attachment A. State your intent to employ the criteria as contained in Attachment A for all Westinghouse Category I mechanical equipment in order to confirm the functional operability of such equipment during and after a seismic event up to and including the SSE.

RESPONSE

Refer to Section 3.9(N).2.2.

Q31.2
3.10

Section 3.9.2.4.1 of RESAR-3 states that the pump motor and vital auxiliary electrical equipment will be qualified by meeting the requirements of IEEE Standard 344-1971. Since the standard has undergone a major revision, state your intent to meet the requirements of the 1975 version of IEEE Standard 344. IEEE Standard 344-1975 includes requirements which are applicable to all plants with C.P. applications docketed after October 1972.

RESPONSE

Refer to Section 3.9(N).3.2.

Q31.3
3.10

The seismic qualification criteria for electrical equipment as stated in Section 3.10 of the proposed Amendment 6 to RESAR-3 is not completely acceptable because it is only applicable to certain specific conditions when single frequency input to an individual axis is justifiable. A broader criterion to account for overall considerations should be provided. The major concern is the possible directional coupling and the concurrent multi-mode response. An acceptable response is to conduct a seismic qualification program as recommended by the 1975 version of IEEE-344 standard. State your intent to use this recommended criteria.

RESPONSE

Refer to Section 3.10(N).

Q31.4 3.11	The lists of safety related equipment and components provided in Section 3.11.1 of RESAR-3 are not complete. Identify all individual components and complete the lists.

RESPONSE

Refer to Section 3.11(N).

Q31.5 3.11	Section 3.11.2 of RESAR-3 does not give a complete and acceptable description of the qualification tests and analyses for each type of safety related equipment and component. Provide this information for each item.

RESPONSE

Refer to Section 3.11(N).

Q31.6 3.11	RESAR-3 Section 7.1.2.5. Describe how your design complies with IEEE Standard 323-1971, or IEEE Standard 323-1974, for all applications for which the construction permit safety evaluation report was issued July 1, 1974 or later. Identify and justify all exceptions.

RESPONSE

Refer to Section 3.11(N).

Q31.7 7.1	In accordance with the implementation dates (noted in parentheses) and as they apply to your application, describe the extent to which the recommendations of the following regulatory guides will be met. Identify and justify any exception. Regulatory Guide 1.22 (Safety Guide 22), "Periodic Testing of Protection System Actuation Functions" (Guide dated 2/17/72) Regulatory Guide 1.29, "Seismic Design Classifications;" (Revision 1 dated August 1973) Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment;" (Guide dated August 11, 1972) Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants;" (Guide dated 3/16/73)

Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;" (Guide dated May 1973)

Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems;" (Guide dated June 1973)

Regulatory Guide 1.62, "Manual Initiation of Protective Actions;" (Guide dated October 1973)

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants;" (Guide dated October 1973)

Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors;" (Guide dated November 1973)

Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants;" (Guide dated January 1974)

Regulatory Guide 1.75, "Physical Independence of Electric Systems." The physical identification of safety-related equipment should also be addressed in this section; (Guide dated February 1974)

Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems;" (Guide dated June 1974) and

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants." (Applicable to all plants with an SER issued after July 1, 1974).

RESPONSE

Refer to Appendix 3A for the applicability of these regulatory guides.

Q31.8(1)
7.1

Provide a discussion and the results of an analysis showing how your design of the test and calibration features of the safety systems meets the requirements of Section 4.10 of IEEE std 279-1971.

RESPONSE

Refer to Sections 7.1.2.5.2, 7.1.2.6.2, 7.2.2.2.3, and 7.3.8.2 item b.5 and Figures 7.3-2 and 7.3-3.

Q31.8(2)
7.2

Based on Figure 7.2-1, Sheet 7 of 17, of RESAR-3 we have concluded that the proposed design for the steamline differential pressure circuits does not conform to the requirements of IEEE Standard 279-1971. Specifically, during operation with a loop isolated, the logic for the operable steamlines is effectively changed to 2-out-of-2 which does not meet the single failure criterion. Our position is that in order to comply with IEEE Std 279-1971, the design should incorporate positive means of assuring that these circuits continue to meet the single failure criterion during operation with a coolant loop isolated. Discuss your intent to comply with this position and describe the necessary design changes, or justify any exceptions by discussing your reasons for concluding that such exceptions are in accordance with the requirements of IEEE Standard 279-1971. In addition as committed on Page 7.2-30 of RESAR-3, provide the results of an analysis that will determine whether automatic tripping of the steamline differential pressure bistables is required for N-1 loops operating.

RESPONSE

Refer to Figure 7.2-1 (Sheet 7) and Table 7.3-13.

Q31.9
3.7.2

RESAR-3 Section 7.2.1.1.2(1)(d) and Figure 7.2-1 Sheet 3 address a power range high neutron flux rate "Positive" trip. This trip is used as protection against a rod ejection accident. The referenced Westinghouse Topical Report WCAP-7380-L (pages 2-8 and 3-12) provides a diagram and a description for the "Negative" flux rate trip but does not provide for the "Positive" flux rate trip. Provide a description and diagram covering "Positive" flux rate trip.

RESPONSE

WCAP-7380-L is no longer referenced. It is replaced with WCAP-8255, as listed in Section 7.2.4. WCAP-8255 discusses both the positive and negative rate trips and provides diagrams for both.

Q31.10
7.2

The reactor trip system contains logic circuits that can initiate trips for the purpose of anticipating the approach to a limiting condition for operation. Specifically, these reactor trips are:

(1) Generation of a reactor trip by tripping the main coolant pump breakers,

(2) Generation of a reactor trip by tripping the turbine,

(3) Generation of reactor trip by underfrequency conditions on reactor coolant pump bus, and

(4) Generation of reactor trip by undervoltage conditions on reactor coolant pump bus.

Our position requires that all inputs to the reactor trip system be designed to meet IEEE Standard 279-1971, with an exception for anticipatory trips (trips not required for safety actions in the accident analysis - Chapter 15). The exception is that sensors for anticipatory trips are not required to be located in a qualified seismic Category I structure. Discuss your intent to comply with this position or justify any exceptions you may have in this regard. Your response should include a discussion of the testability of these circuits while the reactor is at power.

RESPONSE

(1) The design is changed. A reactor trip caused by the main coolant pump breaker opening is one condition of the under-voltage trip. Refer to Section 7.2.1.1.2, item d.2.

(2) Refer to Section 7.2.1.1.2, item f.

(3) Refer to Section 7.2.1.1.2, item d.3.

(4) Refer to Section 7.2.1.1.2, item d.2.

Q31.11
7.2, 7.3

Testing of the reactor trip system and the engineered safety feature actuation system to verify that the "systems" response times are equal to or less than the values assumed in the accident analysis is discussed on Pages 7.1-19, 7.2-24, and 7.3-13 of RESAR-3. In addition to the proposed response time testing during preoperational start-up testing and following the replacement of a component that affects response time, our position requires that these systems be designed to permit periodic verification that the response times are within the values assumed in the accident analysis. Discuss your intent to comply with this position or justify any exceptions. It is stated in RESAR-3 on Page 7.3-26 that the response time specified in Paragraph 4.1 of IEEE Standard 338-1971 is not checked periodically as is the setpoint accuracy. Provide justification for the exception to this requirement.

RESPONSE

Refer to Section 7.1.2.6.2.

Q31.12
7.3

With regard to the motor operated accumulator isolation valves, we require that the proposed design include the following features in order to conform to the requirements of IEEE Std 279-1971:

(1) Automatic opening of the accumulator valves when either (a) the primary coolant system pressure exceeds a preselected value (to be specified in the Technical Specifications) or (b) a safety injection signal has been initiated. Both signals shall be provided to the valves.

(2) Visual indication in the control room of the open or closed status of the valve, actuated by sensors on the valve.

(3) An audible alarm, independent of Item (2), that is actuated by a sensor on the valve when the valve is not in the fully open position.

(4) Utilization of a safety injection signal to automatically remove (override) any bypass feature that may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with the provisions of the proposed Technical Specifications). Discuss your intent to comply with these requirements or justify any exceptions to these requirements.

RESPONSE

Refer to Section 7.6.4 and Figure 7.2-1 (Sheet 6).

Q31.13
7.3

Based on the information provided in Section 7.3 of RESAR-3, we conclude that the proposed design for manual initiation of steamline isolation does not conform with the requirements of Section 4.17 of IEEE Standard 279-1971. In addition, there is not sufficient information on the design provision for manual initiation of containment isolation and containment depressurization to determine whether these functions are designed in accordance with Section 4.17 of IEEE Standard 279-1971. Our position is that a design which meets the following is an acceptable means of meeting the requirements of Section 4.17 of IEEE Standard 279-1971:

(1) Means should be provided for manual initiation of each protective action (e.g., reactor trip, containment isolation) at the system level, regardless of whether or not means are also provided to initiate the protective action at the component or channel level (e.g., individual control rod, individual isolation valve).

(2) Manual initiation of a protective action at the system level should perform all actions performed by automatic initiation such as starting auxiliary or supporting systems, sending signals to appropriate valves to assure their correct position, and providing the required action-sequencing functions and interlocks.

(3) The switches for manual initiation of protective actions at the system level should be located in the control room and be easily accessible to the operator so that action can be taken in an expeditious manner.

(4) The amount of equipment common to both manual and automatic initiation should be kept to a minimum. It is preferable to limit such common equipment to the final actuation devices and the actuated equipment. However, action-sequencing functions and interlocks (of Position 2) associated with the final actuation devices and actual equipment may be common providing individual manual initiation at the component or channel level is provided in the control room. No single failure within the manual, automatic, or common portions of the protection system should prevent initiation of protective action by manual or automatic means.

(5) Manual initiation of protective actions should depend on the operation of a minimum of equipment consistent with 1, 2, 3, and 4 above.

(6) Manual initiation of protective action at the system level should be so designed that once initiated, it will go to completion as required in Section 4.16 of IEEE Standard 279-1971.

Discuss your intent to comply with this position or justify any exceptions by discussing your reasons for concluding that such exceptions are in accordance with the requirements of IEEE Standard 279-1971.

RESPONSE

Refer to Section 7.3.8.2, item b.7.

Q31.14
7.4

General Design Criterion 37 requires, in part, that the emergency core cooling system be designed to permit testing the operability of the system as a whole. On Page 7.3-26 of RESAR-3, it is stated that the safety injection and residual heat removal pumps are made inoperable during the system tests. Our position is that in order to comply with the requirements of Criterion 37, these pumps must be included in the system test. Discuss your intent to comply with this position or justify any exception.

RESPONSE

Refer to Section 6.3.4.2.

Q31.15
7.3, 6.3

Section 6.3.5.1 of RESAR-3 states that only "one temperature detector which provides heater control for the immersion heater, control room alarm and control room indication" is provided for the boron injection surge tank. Provide the results of an analysis which addresses the effect of a single failure in this system. This analysis should include possible boron dilution during recirculation. Also, it is our position that the monitoring system for the boron injection system meet IEEE Standard 279-1971. Discuss your intent to comply with this position or justify any exceptions you may have in this regard.

RESPONSE

Not applicable. Referenced systems have been deleted.

Q31.16
7.3.1

The description of the Emergency Safety Feature systems provided in Section 7.3.1 of RESAR-3 is incomplete in that it does not provide all of the information requested in Section 7.3.1 of the Standard Format for those safety related systems, interfaces and components supplied by the applicant which match with the RESAR-3 scope systems. Provide all of the descriptive and design basis information requested in the Standard Format for these systems. In addition, provide the results of an analysis, as requested in Section 7.3.2 of the Standard Format, to demonstrate how the requirements of the General Design Criteria and IEEE Standard 279-1971 are satisfied and the extent to which the recommendations of applicable Regulatory Guides are satisfied. Identify and justify each exception.

RESPONSE

Refer to Section 7.3.8.

Q31.17 7.3.1	Provide analyses showing that no adverse effects will occur or a discussion of such adverse effects that could occur as a result of power interruption to the Engineered Safety Features Actuation System at any time following the onset of a LOCA or other accident conditions in the plant.

RESPONSE

A power interruption to the engineered safety features system, in conjunction with a LOCA or other postulated accident, is believed to be a highly improbable event. To satisfy the requirements of GDC-35, the accident analyses assume a loss of offsite power coincident with certain postulated events, such as a LOCA. In addition, it is assumed that a single failure occurs which causes the loss of one of the two onsite emergency diesel generators.

The assumption of a loss of offsite power (LOSP) at any time following the onset of a LOCA is unnecessarily conservative. However, if one assumes that a LOSP occurs following the onset of a LOCA, the load shedder emergency load sequencer (LSELS) will function properly to load the ESF busses with loads required for a LOCA in the proper sequence if the safety injection signal (SIS) is still present.

If one assumes that SIS has been reset prior to the LOSP, the LSELS will function to load the ESF busses with only those loads required for LOSP. Operator action would be required to actuate the loads required for LOCA. These operator actions to actuate the loads required by the LOCA are addressed in the emergency operating procedures. See Section 6.3.2.8 and Tables 6.3-8 and 15.0.8 for further discussion.

Q31.18 7.4 15.3.6	General Design Criterion 25 requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded from an accidental withdrawal of a single rod control cluster assemby (not ejection). In the accident analysis, presented in Section 15.3-6 of RESAR, it is stated that "no single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single rod control cluster assembly." However, Chapter 7.0 does not describe how the design prevents such an occurrence. Provide a detailed description of the control circuitry and discuss how the design meets the requirements of Criterion 25.

RESPONSE

Refer to Section 7.7.2.2 and Figure 7.7-15.

Q31.19
7.4, 7.5
7.6

Provide a discussion which supplements those in Sections 7.4, 7.5 and 7.6 of RESAR-3 and which addresses the Standard Format information requirements for the safe shutdown systems, the safety-related display instrumentation and other safety systems and equipment outside the RESAR-3 scope which are assumed in the RESAR-3 and the PSAR Chapter 15 accident analyses.

RESPONSE

The safety-related systems are identified in Section 7.1.1. The safe shutdown safety-related system and other safety-related systems are discussed in Sections 7.4, 7.5, and 7.6.

Q31.20
7.6.2

In addition to the design features discussed in Section 7.6.2 of RESAR-3, it is our position that the design of the RHR isolation valves satisfy the following:

(1) The interlocks shall utilize diverse equipment, and

(2) The interlocks shall be designed in accordance with the intent of IEEE Standard 279-1971.

The information presented in Section 7.6.2 of RESAR-3 does not address the requirement for diverse equipment and describes a degree of testability that conflicts with the requirements of IEEE Standard 279-1971. In addition it is stated that the position indications for the RHR valves differ from those for the accumulator isolation valves but these differences are not identified. Discuss your intent to comply with the requirements that the design shall utilize diverse equipment and shall include complete on-line test capability without opening the isolation valves, or justify any exceptions. In addition identify the differences in the position indications provided for the RHR valves compared to the accumulator valves and discuss the reasons for the differences.

RESPONSE

For the response to this question, refer to Section 5.4.7.

5.1

Provide the list of transients that were analyzed in determining the maximum steam system pressure transient for sizing the steam generator safety valves.

RESPONSE

Refer to Section 5.2.2.

5.2 In reference to Section 5.3.4, provide Reactor Coolant System Temperature - Percent Power Map for plant with loop stop valves if different from Figure 5.3-1.

RESPONSE

Since the Callaway Plant does not incorporate loop stop valves, this question is not applicable.

5.2.2 Provide a discussion of the consequences of inadvertent overpressurization resulting from a malfunction or operator error when the reactor coolant system is water-solid during startup or shutdown. The discussion should include consideration of the pressure-temperature operating limitations on the reactor vessel to protect against brittle fracture. In addition, discuss any design provisions that will be incorporated into the facility design to prevent overpressurization incidents that would exceed allowable pressures in this particular plant condition.

RESPONSE

Refer to Section 5.2.2.

5.3 Justify the fouling factor resistance specified in Section 5.5.2.3.1. Correct the difference between Section 5.5.2.3.1 and Table 5.5-3 with regard to the fouling factor.

RESPONSE

The fouling factor is discussed in Section 5.4.2.5.1 and is consistent with the value reported in Table 5.4-3.

5.4 Provide pressurizer relief and safety valve capacities when discharging water liquid.

RESPONSE

Liquid flow rates assumed in the analysis are based on the homogeneous equilibrium saturated flow model which gives the most conservative relief rate. Accident analysis demonstrates that water relief through the pressurizer safety valves does not occur during the feedline rupture event.

6.1 Item 6.3.2.11 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 1, October 1972) indicates the need to distinguish between true redundancy incorporated in a system and multiple components. To complement the SAR discussions in this regard, provide a summary of a systematic core cooling functional analysis of components required over the complete range of coolant pipe break inside the containment. The summary should be shown in the form of simple block diagrams beginning with the event (pipe break), branching out to the various possible sequences for the different size breaks, continuing through initial core cooling and ending with extended to long-term core cooling. When complete, the diagram should clearly identify each safety system required to function to cool the core for all coolant pipe breaks inside the containment during any plant operating state. The attached Figure 6-1 is provided as a guide.

RESPONSE

System reliability of the ECCS, including a discussion of redundancy and compliance with the single failure criteria, is provided in Section 6.3.2.5. Functioning of the various ECCS components for various accidents, including large and small LOCAs, is discussed in Section 6.3.3. The actual LOCA analyses are discussed in Sections 6.2 and 15.6.5.

Also refer to the Response to Question 15.0(1).

6.2 For each engineered safety feature identified in Question 6.1, list the auxiliaries required for its operation.

RESPONSE

Refer to Section 6.3.2.2 and the Response to Question 15.0(1).

15.0(1) For each transient and accident analyzed in Chapter 15, provide the following information:

(1) The step-by-step sequence of events from event initiation to the final stabilized condition. This listing should identify each significant occurrence on a time scale, including for example: flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, containment isolated, etc.

All required operator actions should also be identified.

(2) The extent to which normally operating plant instrumentation and controls are assumed to function.

(3) The extent to which plant and reactor protection systems are required to function.

(4) The credit taken for the functioning of normally operating plant systems.

(5) The operation of engineered safety systems that is required.

RESPONSE

The sequence of events listed for each transient is provided in tables in Chapter 15.0. The assumptions for instrumentation, controls, protection systems, and ESF systems are described for each transient analyzed in Chapter 15.0.

Figures of the step-by-step sequence of events for each transient are also provided in Chapter 15.0.

5.2.7 and 6.3 Discuss the ability to assure that the operational capability of the valves that are required to function in the short and long term LOCA modes of ECCS operation are not impaired by potential crystallization of boric acid solutions on the valve stem due to leakage. Appropriate methods may include the ability to detect individual valve stem leakoff or periodic operational testing of the valves.

RESPONSE

Refer to Section 6.3.2.2.

15.0(2) Section 15.2.4 of RESAR-3 UNCONTROLLED BORON DILUTION, analyzes the effects of a dilution at power. The analysis discusses the causes of the incident, and the automatic actions of the Reactor Protection System and the manual actions prompted by alarms and instrumentation that would mitigate the consequences of the accident.

However, there is a possible situation, involving the loss of offsite power, where a dilution incident may not be as readily apparent as that described in Section 15.2.4 and where no automatic Reactor Protection System action is available.

In order to assess the potential severity of a dilution accident after a loss of offsite power, provide the results of an analysis that assumes the anticipated equipment configurations in normal use prior to the event that results in the most severe consequences. The analysis should include a dilution operation in progress with the Chemical and Volume Control System mode selector switch being in the DILUTE position (or ALTERNATE DILUTE mode). The loss of offsite power is then assumed to occur with the minimum shutdown reactivity insertion due to control rods. Both diesel generators start and sequence the loss of offsite power loads.

The concerns are that the charging pumps again automatically start running after being loaded to the diesel generators and from electrical schematics of control circuits for the reactor makeup water pumps, that the reactor makeup water pumps would also again automatically start with the mode selector switch in DILUTE. Therefore, a dilution of the Reactor Coolant System is again in progress which could potentially result in a return to critical.

If the reactor makeup water batch integrator is assumed to malfunction by not automatically cutting off flow at the pre-selected value, provide the time available for manual action before the total shutdown margin is lost due to this dilution. If operator action is to be prompted by alarms, describe the features that will alert the operator to this specificaction at a time when alarms from many plant systems are occurring simultaneously.

RESPONSE

This question is not applicable since the reactor makeup water pumps cannot be supplied by the emergency diesel generators.