WOLF CREEK
UPDATED SAFETY
ANALYSIS REPORT
(USAR)

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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 (KG&E) 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. The terminology "the Operating Agent" is used throughout this report to identify the managing corporation for WCGS. At this time the Operating Agent is Wolf Creek Nuclear Operating Corporation (WCNOC).

1.1.1 LICENSE REQUESTED

The Safety Analysis Report was submitted to the Nuclear Regulatory Commission (NRC) in support of the application by the Operating Agent for a Class 103 license to operate a nuclear power facility.

The participants in the Wolf Creek project and their portions of ownership are: Kansas City Power & Light Company (47 percent), Kansas Electric Power Cooperative, Incorporated (6 percent), and Kansas Gas and Electric Company (47 percent). Great Plains Energy Incorporated (Great Plains) and Westar Energy, Inc. (Westar) through subsidiaries Kansas City Power and Light (KCP&L) and Kansas Gas and Electric Company (KG&E), respectively owns 47% of WCNOC and WCGS. The Great Plains and Westar merger was finalized June 4, 2018, and Westar became a wholly-owned subsidiary of Great Plains. As a result of this merger, Great Plains owns a combined 94% of WCNOC and WCGS. The remaining 6% ownership interest is held by Kansas Electric Power Cooperative, Inc. (KEPCO). See Section 1.4.1 for additional discussion of plant ownership.

This report was originally submitted in two parts, the SNUPPS FSAR and the Wolf Creek Site Addendum. It was combined into one report, the Wolf Creek Updated Safety Analysis Report, in the first update after receipt of the Operating License. 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 Wolf Creek Generating Station (WCGS) can be operated without undue risk to the health and safety of the public.

The Licensees received a low power (less than five percent) license to operate the Wolf Creek Generating Station on March 11, 1985. The full power license was issued on June 4, 1985.

1.1.2 PLANT UNITS

The application was for a single pressurized water reactor nuclear unit. The power block was built to the SNUPPS duplicate plant design. The ESW Vertical Loop Chase design is not included in the SNUPPS duplicate plant design. Evaluations of the site characteristics and the design of the cooling system and other site-related systems and facilities have considered the installation of a second nuclear unit at a later date.

The WCGS power block, consists of these structures, including enclosed systems and components:

a. Reactor building (containment)

b. Turbine building

c. Control building

d. Auxiliary building

e. Diesel generator building

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

j. ESW Vertical Loop Chase

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 were 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) for work not yet completed 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.

The ESW Vertical Loop Chase design was analyzed using the design envelopes applied to the original SNUPPS standard plant design.

1.1.3 PLANT LOCATION

The site for the Wolf Creek Generating Station, Unit No. 1, is located approximately 3.5 miles northeast of the town of Burlington, in Coffey County, Kansas. The site is situated approximately 3.5 miles east of the Neosho River and the John Redmond Reservoir. The nearest population center is Emporia, Kansas, located 28 miles west-northwest of the site. It is approximately 75 miles southwest of Kansas City, Kansas.

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 Wolf Creek is a pressurized water reactor (PWR) which was designed and supplied by the Westinghouse Electric Corporation.

The reactor core was designed for an output of 3,411 MWt. When the reactor coolant pump input of 14 MWt was added to the core output, the warranted nuclear steam supply system output was 3,425 MWt, which was defined as the rated power in the license application. The engineered safety features were designed for a core power of 3,565 MWt. An additional 2 percent conservatism was added for some analyses to give a maximum accident analysis power of 3,636 MWt. Analyses were performed in 1992 to uprate the reactor core power to 3565 MWt.

The turbine generator is rated for operation at the NSSS output of 3,425 MWt. The corresponding turbine generator electrical output is 1,186 MWe. The turbine generator has a valve wide open capability of 1,234 MWe, assuming an NSSS output of approximately 105 percent of the rated steam flow. The turbine generator was designed and supplied by the General Electric Company.

The Wolf Creek Power Rerate Program increases the licensed reactor core power level from 3411 $MW_{(th)}$ to 3565 $MW_{(th)}$. The estimated turbine-generator output is 1228 $MW_{(e)}$ at the Power Rerate condition, which is based on an NSSS output of 3579 $MW_{(th)}$ and a Reactor Coolant System hot leg (T_{hot}) temperature of 618.2 °F.

The turbine has been upgraded with new monoblock rotors on both the high pressure (HP) and low pressure (LP) turbines. The new HP turbine is a dense pack design (increase in stages from 7 to 9). The last stage buckets in the LP turbines have been increased from 38” to 43”. These efficiency enhancements resulted in an expected turbine-generator output of 1268 MWe.

1.1.6 SCHEDULE FOR FUEL LOADING AND OPERATION

A low power operating license was issued for WCGS on March 11, 1985. WCGS first entered commercial operation on September 3, 1985.

1.1.7 DESIGN BASES

As used within this USAR, 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 after a design basis accident and maintain it in a post-accident 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 USAR

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
AMSAC	(ATWS) Mitigation System Activation Circuitry
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
CHF	Critical Heat Flux
CIS	Containment Isolation Signal
ClCWS	Closed Cooling Water System
CM	Center of Mass
CMAA	Crane Manufacturing Association of America
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

TABLE 1.1-1 (Sheet 2)

CSD	Cold Shutdown
CST	Condensate Storage Tank
CSTS	Condensate Storage and Transfer System
CtCS	Containment Cooling System
CVCS	Chemical and Volume Control System
CWS	Circulating Water System
DAC	Derived Air Concentration
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
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 Makeup 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
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
ESWVLC	Essential Service Water Vertical Loop Chase
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	Floor Response Spectra
FSAR	Final Safety Analysis Report
FSF	Fuel Storage Facility
GDC	General Design Criteria

TABLE 1.1-1 (Sheet 3)

GRWS	Gaseous Radwaste System
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air (filter)
HEX	Heat Exchanger
Hga	Inches of Mercury Absolute
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
ISI	Inservice Inspection
LCO	Limiting Condition of Operation
LEFM	Linear Elastic Fracture Mechanics
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor
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
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
PA	Public Address
PAMS	Post-Accident Monitoring System
PCT	Peak Cladding Temperature
PHS	Plant Heating System

TABLE 1.1-1 (Sheet 4)

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
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
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SACF	Single Active Component Failure
SAR	Safety Analysis Report
SAS	Secondary Alarm Station
SGB	Steam Generator Blowdown
SGBIS	Steam Generator Blowdown Isolation System/Signal
SGBS	Steam Generator Blowdown System
SIS	Safety Injection Signal
SIT	Structural Integrity Test
SJAE	Steam Jet Air Ejectors
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
USAR	Updated Safety Analysis Report
USGS	U.S. Geological Survey

TABLE 1.1-1 (Sheet 5)

UT	Ultrasonic Testing
VWO	Valves Wide Open
W	Westinghouse
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation

1.2 GENERAL PLANT DESCRIPTION

This section describes the plant site, general arrangement of plant structures, design criteria and general design of major systems. Comparisons between the WCGS design and to the design at other plants were made at the time of application for the Operating License. These comparisons are considered historic references and will not be updated to reflect changes in other utilities designs.

1.2.1 PLANT SITE DESCRIPTION

1.2.1.1 Site Location

The WCGS site is located approximately 3.5 miles northeast of the town of Burlington in Coffey County, Kansas. The site is situated approximately 3.5 miles east of the Neosho River and the John Redmond Reservoir. It is approximately 75 miles southwest of Kansas City, Kansas. Site location is discussed in more detail in Section 2.1.1.

1.2.1.2 Site Ownership

The Licensees have either purchased or have obtained easements on the necessary land within the site boundary. Full ownership control of the exclusion area is presently exercised by the Operating Agent and will continue including all mineral rights with full authority to determine all activities within the exclusion area including exclusion or removal of personnel and property from the area. This is in accordance with the exclusion area requirements of 10 CFR 100.3(a).

1.2.1.3 Access to the Site

There are no public highways, county and/or township roads, public waterways, or public railroads that traverse the exclusion area. There are no persons living in the exclusion area. There is to be no one working in the exclusion area except employees of the Applicants and their authorized agents. The exclusion area is patrolled periodically by plant guards to ensure awareness of access to the area by individuals. See Section 2.1.2 for further details on access to the exclusion area and details concerning lake use. Controlled access to the protected area is monitored by guards on a 24 hour per day basis.

1.2.1.4 Environs

The area within 10 miles of the site is rural and of low population; in 1980 the population within the 10-mile radius was 6,652. The only incorporated places,as defined by the U.S. Bureau of Census, within 10 miles of the site are Burlington and New Strawn. In 1980, Burlington recorded a population of 2,901 and New Strawn, which was incorporated in 1971, had a 1980 population of 457. Topeka, located 53 miles north of the site, recorded a metropolitan population of 115,266 in 1980 and Emporia, 28 miles west-northwest of the site, in 1980 recorded a population of 25,287. Most of the land surrounding the site is used for agricultural purposes with the exception of such rural service centers as Burlington. All recreational facilities are city-owned parks with the exception of the John Redmond Reservoir area, the main recreational facility in the area which is federally operated. Use of the Wolf Creek lake for recreation is discussed in Section 2.1.2. The region is expected to retain its distinctly rural character.

The Low Population Zone is chosen as the area within a 2.5-mile radius of the plant site.

1.2.1.5 Geology

The WCGS site is located within the Central Stable Region of the North American Continent. This region was subjected to gentle structural uparching and down-warping during Mesozoic and Paleozoic time. These structural movements resulted in the formation of broad-scale basins and arches which have been modified locally by folding and faulting. Geotechnical investigations at the site during construction excavation have identified the presence of localized zones of penecontemporaneous deformation in the bedrock. However, the investigations have established the last age of deformation as Pennsylvanian, and there is no known macroseismic activity associated with these zones and no structural association with capable faults (Reference 1). The faulting, shearing and deformation, therefore, are noncapable as defined by Appendix A to 10 CFR 100.

The surface bedrock in the site area consists of alternating layers of Pennsylvanian age shales, limestones, sandstones, and a few thin coal seams. These bedrock units dip gently to the west and northwest and have been folded locally into small-scale plunging anticlines and synclines. At the site, the Precambrian basement is present at a depth of approximately 2,500 feet. The Precambrian rocks consist of approximately 1,000 feet of sedimentary deposits which rest on a granitic basement complex.

The site area has been submaturely to maturely dissected by the Neosho River and its tributaries to form flat to gently rolling uplands with a maximum topographic relief of 100 feet or less from the uplands to the valley floors. Residual soils ranging in thickness from 0 to 16 feet are developed on the Pennsylvanian strata. Quaternary alluvium reaches a thickness of approximately 25 feet in the Wolf Creek valley. Scattered Tertiary age deposits of clayey gravel cap some of the higher hills in the site area. Glacial deposits are not present at the site. The alternating Pennsylvanian strata forming the bedrock surface consist of competent rock units with a low amount of structural discontinuities in the rock mass. No major geologic features have been identified which could adversely affect the stability of subsurface materials at seismic Category I facilities. Minor geologic features, such as jointing, the zones of penecontemporaneous deformation, and the weathering profile in the rock, were considered during design and construction of facilities. Comprehensive geotechnical investigations of the site have determined the subsurface conditions in adequate detail to provide design criteria for foundation support of safety-related facilities. Major seismic Category I structures are supported on competent rock. Only minor, localized modifications to foundation materials were required in design and construction to provide uniform support of safety-related facilities.

1.2.1.6 Seismology

The plant site is located in a relatively seismic stable region of the central United States. No earthquake epicenter has been reported closer than 40 miles to the site, and the nearest shocks have had epicentral intensities no greater than Intensity III. At distances of 85 and 105 miles from the site, earthquakes of Intensity VII to VII-VIII have been recorded. Since 1800, only seven earthquakes of Intensity V or greater have occurred within 100 miles of the site, and 16 events of Intensity VI or greater have been recorded within 200 miles. Previously recorded earthquakes probably have not generated intensities greater than VI at the site, and none of the buildings in the vicinity of the site have sustained any known structural damage due to earthquakes.

An Operating Basis Earthquake corresponding to a horizontal acceleration of six percent of gravity and a Safe Shutdown Earthquake corresponding to a horizontal acceleration of 12 percent of gravity was selected for the site. However, a seismic evaluation of these structures, systems, and components using the Lawrence Livermore Laboratories spectrum anchored at 0.15g for structures supported on bedrock is contained in Appendix 3C.

1.2.1.7 Hydrology

1.2.1.7.1 Surface Water Hydrology

The plant site is located within the Wolf Creek watershed northeast of Burlington, Kansas. The topography within the watershed varies from undulating hills upstream of the plant site to a floodplain area shared with the Neosho River with a drainage area within Kansas of 6,300 sq. miles near the mouth of Wolf Creek with a drainage area of 35 sq. miles. The cooling lake alters the draining pattern of the watershed, but safety-related facilities are protected from severe hydrological events.

The cooling lake is designed to supply adequate cooling water to the plant during a one in fifty year drought. Makeup water is supplied to the cooling lake from the Wolf Creek watershed runoff and from makeup water pumped from John Redmond Reservoir. The region surrounding the site is not characterized by events such as tsunamis, surge activity, or severe ice flooding. Major dam failures on the Neosho River above Wolf Creek watershed will not affect safety-related facilities.

The flow of the Neosho River is controlled by three reservoirs above the site. The Maximum flood design elevation of 1097.5 ft. msl., resulting from the probable maximum flood routed through the cooling lake with coincident wave activity, is below the plant site grade of 1099.5 ft. msl.

1.2.1.7.2 Groundwater Hydrology

Only small quantities of groundwater are available within a 50-mile radius of the plant site. The groundwater is produced from three types of aquifers: the alluvial deposits in the river valleys, the weathered bedrock including the shallow soil, and the unweathered bedrock.

The alluvial aquifers are composed of silts, sands, and gravel. Yields from wells in the alluvial aquifers are up to 100 gallons per minute. Recharge to such aquifers occurs from precipitation and from rivers during periods of high flow. Regionally, discharge from the alluvial aquifers normally flows into the rivers.

The weathered bedrock aquifer consists of weathered shales, siltstones, sandstones, and limestones. Pressure tests indicate that this aquifer is sufficiently permeable to yield up to 10 gallons per minute for livestock and domestic wells. Recharge occurs from precipitation and locally from downward percolation through the overlying alluvium. Discharge occurs into both alluvium and streams.

The consolidated bedrock aquifers are composed of sandstones and limestones which are limited to yields ranging from about 1 to 10 gallons per minute. Recharge to such aquifers occurs by precipitation and infiltration of surface water at the outcrops. Where overlain by shales and siltstones, which act as aquitards and aquicludes, vertical recharge to the limestones and sandstones is minimal.

There is no anticipated use of groundwater at the plant site. The operation of the plant will not have any detrimental effect on the groundwater environment, nor will local groundwater use affect the operation of the plant.

1.2.1.8 Meteorology

The continental location of the site ensures a wide seasonal range of temperature and frequent day to day temperature changes due to frequent passage of cyclonic systems through the vicinity. The maximum temperature was 117 degrees Fahrenheit recorded at Burlington, Kansas. The lowest extreme temperature was -26 degrees Fahrenheit. The prevailing winds are from the south to southeast except during the winter when north to northwest winds prevail. There are no meteorologically significant terrain features or bodies of water within 50 miles of the site.

The site vicinity is subject to occasional severe thunderstorms and the possibility of a tornado from early spring until autumn. The world record 42 minute rainfall of 12 inches occurred at Holt, Missouri, approximately 120 miles from the Wolf Creek Site. However, precipitation is generally moderate throughout the year and snowfall ranges from very little during some winters to substantial during others.

The fastest wind, excluding tornadoes was 86 mph.

The diffusion climatology is generally favorable due to the frequent passage of cyclonic storm systems. The poorest diffusion conditions occur during (1) nighttime inversions which become most developed during winters and (2) dominance of the site area by stagnant anticyclonic systems which may persist for several consecutive days, especially during late summer and autumn.

1.2.2 GENERAL ARRANGEMENT OF STRUCTURES

The principal structures located on the Wolf Creek Generating Station site are listed below.

a. Reactor Building - houses the reactor, reactor coolant piping, 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 fuel storage pool, the fuel handling system, and a portion of the spent fuel pool cooling and cleanup system;

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 IE switchgear, the Class IE battery rooms, the access control area, cable spreading rooms, and portions of the main control room emergency ventilation systems;

g. Storage Tanks - include the condensate storage tank, the refueling water storage tank, the reactor makeup water storage tank, the demineralized water tank, and the emergency fuel oil storage tanks;

h. Diesel Generator Building - houses the diesel generators and associated equipment;

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

j. Communication Corridor;

k. Deleted

l. Cooling lake and ultimate heat sink;

m. Circulating Water Screenhouse - houses traveling screens, service water pumps and strainers, circulating water pumps, fire protection pumps, and chemical injection systems for raw water treatment;

n. Essential Service Water Pumphouse - houses pumps and strainers for the essential service water system;

o. Deleted

p. Hot Machine Shop;

q. Administration Building;

r. Shop building;

s. Materials Center (Warehouse);

t. Technical Support Center;

u. Prescreening and Main Security Buildings;

v. Covered Walkway;

w. Education Center simulator/training complex);

x. Non Discharging Sewage Lagoon;

y. Switchyard;

z. Make-up Water Screenhouse (located below John Redmond Dam);

aa. Make-up Discharge Structure;

bb. Outage Processing Center;

cc. Support Building West;

dd. General Office Building

ee. Waste Water Treatment Facility;

ff. Waste Water Treatment Facility -- houses the recirculation pumps, chemical reagent storage tanks and feedpumps for the wastewater treatment system;

gg. Waste Water Retention Basins -- two 300,000 gallon open top concrete basins used for retaining and neutralizing secondary regenerative wastewaters prior to discharging to the WCGS cooling lake.

hh. Owens Corning Building

ii. Cathodic Protection Building (Rectifier Shelter #1)

jj. Cable Reel Yard Building

kk. X-Ray Building

ll. Water Treatment Building North

mm. Chemical Addition Building

nn. Station Blackout Diesel Generator Missile Barrier

oo. Cathodic Protection Building (Rectifier Shelter #2, near Firing Range)

pp. ESW Vertical Loop Chase – houses both trains of the ESW vertical loops.

qq. Primary Flex Storage Building

rr. Emergency Operations Facility

NOTE: The above list gives a general use description of the principal structures. The actual name of the structure may differ.

The general arrangement of these and other structures and equipment is shown in Figures 1.2-1 through 1.2-43. The site area layout is shown in Figure 1.2-44.

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 USAR.

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.

1.2.3.1 SNUPPS Design Envelope

The Wolf Creek power block was designed and evaluated to the SNUPPS design envelope which was established by:

a. A design criterion which is 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. The initial reactor core design for WCGS is essentially identical to the design for the Comanche Peak units (Docket Nos. 50-445 and 50-446).

In the initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, some of the fuel assemblies (e.g., one-third) are discharged. The remaining fuel assemblies are, in general, moved to other positions in the core, and fresh fuel assemblies are 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 93Al vertical, single stage, centrifugal pumps of the shaft-seal type. WCGS was one of the first domestic operating units to utilize Model 93Al reactor coolant pumps. However, Westinghouse Model 93Al 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 93Al 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 pumps utilized at WCGS are identical to those used at Callaway.

The steam generators are Westinghouse Model F vertical U-tube units, which contain thermally treated Inconel tubes. The Model F steam generator includes features (discussed in detail in Section 5.4.2) such as improved tube support plate design and high circulation ratio, which are designed to minimize most forms of corrosion, sludge buildup, and chemical attack. Integral moisture separation equipment reduces the moisture content of the existing steam to one-quarter percent or less. WCGS was the second domestic operating unit to utilize Model F steam generators, (Callaway was the first domestic operating unit to utilize the Westinghouse Model F steam Generator). Westinghouse Model F steam generators were previously reviewed by the NRC in conjunction with the Sundesert PSAR application (Docket Nos. 50-582 and 50-583). An interim Safety Evaluation Report (NUREG-0469) was issued on the Sundesert PSAR application in October 1978.

Essentially all of the metal surfaces in contact with the reactor water are stainless steel, except the steam generator tubes (Inconel), fuel assembly skeleton (stainless steel, Inconel and zirconium ally) and fuel rod cladding (zirconium ally).

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 WCGS is essentially identical to those utilized in the Comanche Peak units, Callaway and many other 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 and identical to the structure at Callaway.

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. Sodium hydroxide is added to the spray to remove iodine from the containment atmosphere in the post-LOCA condition.

The containment spray system is similar in design and function to the reactor building spray system at the Midland Plant, Units 1 and 2. Although the WCGS containment spray system utilizes two containment sumps, versus one for Midland, both systems function under equivalent conditions. The WCGS containment spray system is identical to the system used at Callaway.

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, three fan cooler units are required to remove sensible 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, Palo Verde, and Midland and are identical to components used in the system at Callaway.

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 combustible gas control system including the hydrogen recombiner system, hydrogen monitoring system, and the hydrogen purge system has components that are similar in design and function to the combustible gas control system used at Midland Plant, Units 1 and and 2.

The hydrogen recombiners utilized at WCGS are essentially identical to those utilized in the Comanche Peak units, Callaway, and many other 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. The containment isolation system used at WCGS is essentially identical to that utilized at Callaway.

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 WCGS is functionally identical to the ECCS design for the Comanche Peak units and Callaway.

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° degrees 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, if necessary, be used to supply feedwater to the steam generators for removal of decay and sensible heat from the reactor coolant system. See section 7.3.6.1.1 for a description of this operation.

The auxiliary feedwater system has a design that is similar to the auxiliary feedwater system design on the Midland Plant, Units 1 and 2. Both designs utilize steam-driven and ac-powered motor-driven auxiliary feedwater pumps. However, the WCGS design utilizes an additional motor-driven pump for reliability. The auxiliary feedwater system utilized at WCGS is identical to that used at Callaway.

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 shutdown 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 post-accident or post-fire 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 WCGS 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 and are essentially identical to those systems utilized at Callaway.

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 was designed to initiate automatically protective action whenever the monitored parameters reach a limiting safety system setting. Redundancy was provided to assure that no single failure would prevent protective action when it was required. The plant protection system was 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 was 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 would automatically be shut down.

1.2.6.1.2 Engineered Safety Features Actuation System

The engineered safety features actuation system was 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 startup and shutdown; (3) and by a soluble neutron absorber, boron, in the form of boric acid inserted during cold shutdown, partially removed at startup, 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 increases of up to 10 percent and ramp load increases of up to 5 percent per minute over the load range of 15 to 100 percent of full power. Equal step and ramp load reductions are possible, over the range of 100 to 15 percent of full power.

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 analog Electro-Hydraulic Control (EHC) system has been replaced with a new digital Turbine Control System (TCS). The new system utilizes an Ovation-Based Distributed Control System (DCS). Two redundant sets of controllers are used in the turbine control system. The TCS architecture is based on combined functional and hardware redundancy to create a robust and reliable system.

Additional instrumentation and controls allow manual or automatic control of various temperatures, pressures, flows, and liquid levels throughout the plant. 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.

1.2.7 PLANT ELECTRIC POWER SYSTEM

1.2.7.1 Transmission and Generation Systems

The generating units are connected to the respective utility transmission systems. The transmission system voltage is 345 kV for Kansas Gas and Electric Company (KG&E) and Great Plains. The utilities have integrated transmission networks and interconnections with neighboring systems. A description of system network and interconnection for each utility 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 coupled directly to the generator shaft.

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

1.2.7.2 Electric Power Distribution System

Electric power is supplied from the switchyard to the onsite power system for the electrical auxiliaries of each unit 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 buses 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 IE and non-Class IE ac and dc power systems. The Class IE power system supplies equipment used to shut down the reactor and limit the release of radioactive material following a design basis event.

The Class IE ac system for each unit 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 for each unit, one for each of the two Class IE 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 IE 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 IE equipment is designed to accept the expected range in voltage.

The Class IE electrical systems are similar to Class IE 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. For instance, Class IE system and components have a similar design and function to the ac systems and components at the Midland Plant, Units 1 and 2. In addition, the Class IE dc systems and components are similar to the dc systems and components at Palo Verde. The Class IE system used at WCGS is similar to the system used at Callaway.

Direct current power for the Class IE dc loads of each unit is supplied by four independent Class IE 125-V dc batteries and associated battery chargers. One 250-V and four 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.

The Station Blackout Diesel Generator (SBO DG) system consists of three (3) non-safety related diesel generators that are capable of supplying essential loads on bus NB001 or bus NB002 required to reliably and safely shut down the unit following a station blackout event. The SBO DG system is also capable of supplying power to the non-safety auxiliary feedwater pump (NSAFP). Station blackout means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system).

The SBO DG system is not credited for coping with a station blackout per NRC Regulatory Guide 1.155 and NUMARC 87-00, but is instead installed to provide plant MSPI/PRA margin.

The SBO DGs are located within a missile barrier designed to limit the average wind speed downstream of the barrier entrance to less than or equal to 150 mph during a 230 mph tornado event in accordance with NRC Regulatory Guide 1.76, Rev. 1.

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 were 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 atmospheric relief valve. The main steam supply system is similar to the main steam supply system at Palo Verde and identical to the main steam supply system at Callaway.

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. The system used at WCGS is identical to the system used at Callaway.

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 and identical to the system used at Callaway.

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 startup, 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 and identical to the system at Callaway.

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 Wolf Creek site utilizes a large cooling lake for its source of circulating water and cooling mechanism as does Comanche Peak.

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 is similar in design and function to the cleanup system utilized at Midland Plant, Units 1 and 2. The WCGS design includes additional components, such as a waste collection tank and sluice water pumps. The system used at WCGS is identical to the system used at Callaway.

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 and identical to the system used at Callaway.

During normal plant cooldown and startup, the feedwater system is used to supply feedwater to the steam generators for removal of decay and sensible heat from the reactor coolant system.

1.2.8.8 Steam Generator Blowdown System

The steam generator blowdown system functions to maintain the secondary side water within the NSSS supplier's water chemistry specifications. This system includes a flash tank, 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 blow-down 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 re-use 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.

1.2.8.10 Wastewater Treatment System

The wastewater treatment facility processes wastewater discharges from the makeup demineralized water system (WM), the condensate demineralized system (AK), the secondary liquid waste system (HF), water treatment plant acid and caustic skid drains, the water treatment plant chemical spill catch basin, and the wastewater treatment facility chemical spill catch basin prior to discharge to the WCGS cooling lake. This system is designed to ensure that the above mentioned wastewater are in compliance with the pH limitations set forth in the NPDES permit. This system shall also process wastewater as allowed by the Offsite Dose Calculation Manual.

1.2.9 AUXILIARY SYSTEMS

1.2.9.1 Chemical and Volume Control System

The chemical and volume control 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 WCGS is similar to the CVCS design for the Comanche Peak units. The major difference is that WCGS includes provisions in the CVCS and residual heat removal system (see Section 7.4) to improve the capability to achieve and maintain cold shutdown. The CVCS system at WCGS is identical to the system used by Callaway.

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 425 psig and the temperature is from 350 degrees F to 140 degrees F, and to maintain the proper reactor coolant temperature during refueling.

The design of the RHRS includes two motor-operated isolation valves that are closed during normal operations. They are provided with both a “prevent-open” interlock and “RHRS-Iso-Valve-Open” alarm which are designed to prevent possible exposure of the RHRS to normal RCS operating pressure.

The isolation valves are opened for residual heat removal during a plant cooldown after the RCS temperature is reduced to approximately 350°F and RCS pressure is less than approximately 360 psig. During a plant startup, the inlet isolation valves are shut after drawing a bubble in the pressurizer and prior to increasing RCS pressure above approximately 425 psig (alarm setpoint).

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 lowpressure injection header.

The RHRS design for WCGS is similar to the RHRS design for the Comanche Peak units. The major difference is that at Wolf Creek, provisions are included in the CVCS and RHRS (see Section 7.4) to improve the capability to achieve and maintain cold shutdown. The RHRS used at WCGS is identical to the system used by Callaway.

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 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 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 reactor cavity and the fuel storage pool 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 spent fuel 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 spent fuel pool, the fuel is removed from the transfer system and placed into storage racks. After a suitable decay period, the fuel is removed from storage and loaded into a shipping cask for transfer.

The fuel handling system and new fuel storage racks utilized at WCGS are similar to those utilized in the Comanche Peak units and many other facilities that are currently in operation.

1.2.9.4 Service Water Systems

1.2.9.4.1 Service Water System

During normal plant operation, the service water system (SWS) supplies cooling water to the turbine building closed cooling water heat exchangers, central chiller condensers, and pump out units, condenser vacuum pump seal water coolers, steam packing exhauster, generator stator coolers, generator hydrogen coolers, turbine-generator lube oil coolers, steam generator blowdown non-regenerative heat exchanger, CVCS chiller unit, waterbox venting pump seal water coolers, motor driven feed pump, and air compressors. The system also supplies cooling water to the essential service water system during normal operation.

The system draws water from the cooling lake, pumps the coolant through the heat exchangers, and discharges it into the circulating water discharge, where it is directed back to the cooling lake. Water returning from the essential service water system is returned to the Ultimate Heat Sink and/or the cooling lake. Makeup water for the cooling lake is provided by pumps in the makeup water screenhouse. The SWS is described in detail in Section 9.2.1.

The system is similar in design and function to the service water system utilized at Midland Plant, Units 1 and 2.

1.2.9.4.2 Essential Service Water System

The essential service water system (ESWS) provides cooling water from the ultimate heat sink (cooling lake) 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, centrifugal charging pump room coolers, component cooling water pump room coolers, auxiliary feedwater pump room coolers, control room air-conditioning condensers, Class 1E switchgear air-conditioning condensers, penetration room coolers, fuel pool cooling pump room cooler, and air compressor and after cooler.

The ESW cooling water is discharged to the ultimate heat sink. The essential service water pumps, prelube storage tanks, self-cleaning strainers, and traveling water screens are located in a seismic Category I pumphouse. Other parts of the system located outside the power block are either buried underground or located in seismic Category I structures. The ESWS is described in detail in Section 9.2.1.

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

The essential service water system is similar in design and function to the essential service water system utilized at the Midland Plant, Units 1 and 2.

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.

The component cooling water system is similar in design and function to the component cooling water system that is utilized at the Midland Plant, Units 1 and 2 and is identical to the system utilized at Callaway.

1.2.9.6 Compressed Air Systems

Four 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.

The compressed air system is similar in design and function to the compressed air systems that are utilized at the Trojan and Midland Plant, Units 1 and 2 and is identical to the system utilized at Callaway.

1.2.9.7 Fire Protection Systems

The fire protection system was designed to provide water to the plant fire protection system and site fire protection facilities. An outside underground yard loop surrounds the power block and provides water to all buildings and hydrants spaced around the plant site. Water for the fire protection system is provided from the circulating water screenhouse intake bay. The system is described in detail in Section 9.5.1.

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; and

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 was 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 IE electrical buses. These systems satisfy the single failure criterion.

The HVAC systems serving the control room areas are similar in design and function to the HVAC systems at the Midland Plant, Units 1 and 2 and Callaway. 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.

1.2.9.10 Service Gas System

The service gas system provides for the handling and storage of commonly used service gases. 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 gases, e.g. carbon dioxide and oxygen.

1.2.9.11 Communications 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 and Callaway.

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 and Midland Plant, Units 1 and 2 and identical to the system at Callaway.

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 removes or concentrates radioactive constituents, and processes them until suitable 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.

1.2.11 SHARED FACILITIES AND COMPONENTS

WCGS utilized the SNUPPS standard plant or power block design which was developed to be acceptable for installation at any one of several sites. Wolf Creek is a single unit site and has no shared safety-related facilities and components.

1.2.12 REFERENCES SECTION 1.2

1. Dames & Moore, 1977, Penecontemporaneous Deformation Zones Wolf Creek Generating Station; for Kansas Gas and Electric Co. and Kansas City Power & Light Co., Dames & Moore, May 20, 1977.

TABLE 1.2-1

DESIGN ENVELOPE (Sheet 1) (1)

Parameter	SNUPPS	USAR Reference	
Hazards	There are no hazards which have an adverse effect on structures	Section 2.2	Explosions from accidents involving explosives or gases were postulated 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(2)		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	Cooling pond		
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 is precluded by the elevation of the plant and by the site drainage system.	Section 2.4 and Table 3.4-2	No special flood protection measures (such as external flood doors) are incorporated.
Maximum rainfall	7.4 in/hr, excess allowed to run off roofs	Section 2.4	Site drainage designed to preclude local flooding.

TABLE 1.2-1 (Sheet 2)

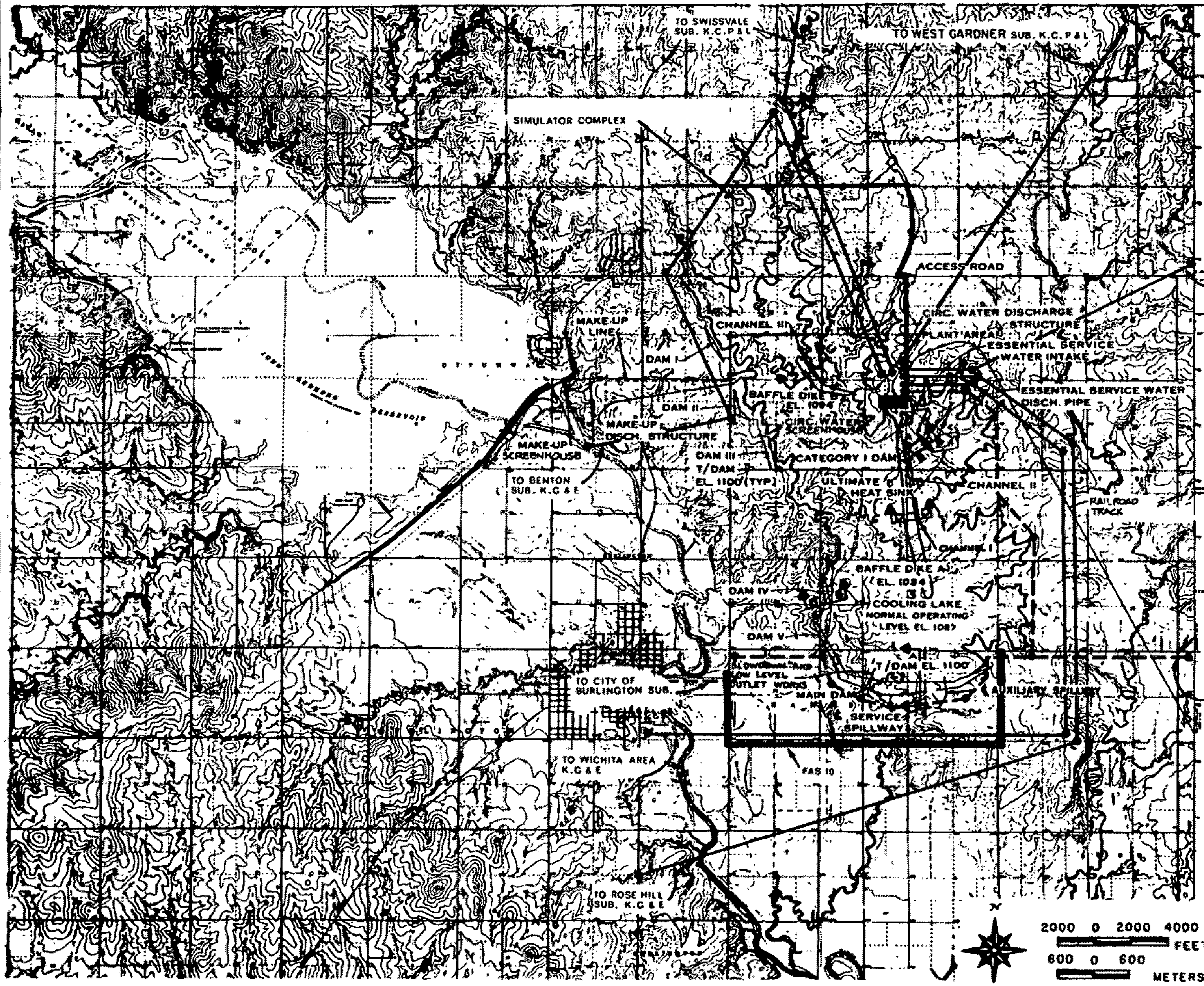
Parameter	SNUPPS	USAR Reference	Remarks
Ice and snow loading		Section 2.4	Basic snow load (100-year recurrence snow-pack) adjusted for geometry and drifting for roof design using Section 7.2 of ANSI A58.1-1972. "Extreme
1. Basic snow load, normal and severe environmental	91 psf		
2. Basic snow load, extreme environ-mental	153 psf environmental" includes		the effects of PMWP.
Ground water elevation	Maximum hydrostatic level is at Grade	Section 2.4	Conservative assumption for buoyancy calcula-tions and computation of uplift pressure on foundation base mats.
Seismology (OBE and	OBE - 0.12g, SSE - 0.20g	Section 2.5	The standard plant SSE) OBE and SSE were used with each site's soil properties to generate seismic structural loads. These loads were enveloped for design. Floor response spectra were generated in the same manner. All items were designed either to the envelope or all of the individual floor response spectra so that these items could be interchangeable at all sites, thus the Wolf Creek site design was limited by the other SNUPPS plants floor response spectra.
Foundation character-istics			
1. Settlement	Design settlements are within the following criteria: a. Total - 3 in. b. Post construction - 1 in. c. Post construction differen-tial (between buildings and/or columns) - 1/2 in.	Section 2.5	

TABLE 1.2-1 (Sheet 3)

Parameter	SNUPPS	USAR Reference	Remarks
2. Static and dynamic lateral earth pressures	The equations for the lateral earth pressures, shown in Figure 2.5-152, 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
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 2.4	WCGS is 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(2)	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) The design envelope was conservatively developed using data from all SNUPPS sites.

(2) The winter temperature conditions have been re-evaluated for Wolf Creek. The acceptable design minimum temperature for exposed outside structures or components is -30°F. The acceptable design winter temperature for HVAC design is 7°F.



NOTE:
LOCATION OF ESSENTIAL SERVICE WATER DISCHARGE PIPE IS HISTORIC. REFERENCE FIGURE 2.5-30 FOR THE LOCATION OF THE NEW ESWS DISCHARGE POINT.

345 KV LINE
69 KV LINE

WOLF CREEK
UPDATED SAFETY ANALYSIS REPORT

Figure 1.2-43 Rev. 28
SITE AREA LAYOUT
Sheet 1 of 1

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 WCGS 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). Wolf Creek and Callaway were both designed using the SNUPPS powerblock design. These comparisons were made at the time of application for the Operating License and are considered historic data. Table 1.3-1 will not be updated to reflect changes made at these facilities.

For a general design comparison of the major BOP systems utilized by WCGS 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 WCGS 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 were made to the power block since submittal of the SNUPPS PSAR but prior to receipt of the operating license. 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, 26, 50, 51, 55, 70, 71, 73, and 100 is considered.

WOLF CREEK

TABLE 1.3-1

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS[b]	Comanche Peak	W. B. McGuire	Trojan
Reactor core heat output, MWt	4.0, 5.0, 15.0	3,411	3,411	3,411	3,411
Minimum DNBR for design transients	4.1, 4.4, 15.0	>1.30	>1.30	>1.30	>1.30
Total thermal flow rate, 106 lb/hr	4.1, 4.4, 5.1	142.1	142.1	140.3	132.7
Reactor coolant temperatures, F	4.1, 4.4				
Core outlet		621.4	621.4	620.8	619.5
Vessel outlet		618.2	618.2	618.2	616.8
Core average		591.8	591.8	589.4	585.9
Vessel average		588.5	588.5	588.2	584.7
Core inlet		558.8	558.8	558.1	552.5
Vessel inlet		558.8	558.8	558.1	552.5
Average linear power, kW/ft	4.1, 4.4	5.44	5.44	5.44	5.44
Peak linear power for normal operation, kW/ft	4.1, 4.4	12.6	12.6	12.6	13.6
Heat flux hot channel factor, FQ	4.1, 4.4, 15.0	2.32	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

TABLE 1.3-1 (Sheet 2)

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS[(b)]	Comanche Peak	W. B. McGuire	Trojan
Number of fuel assemblies	4.1, 4.3	193	193	193	193
Uranium dioxide rods per assembly	4.1, 4.3	264	264	264	264
Fuel weight as uranium dioxide, lb	4.1, 4.3	222,739	222,739	222,739	222,739
Number of grids per assembly	4.1, 4.3	8-Type R	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 Hafnium	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd
Clad material		Stainless Steel	Stainless Steel	Stainless Steel	Stainless Steel
Clad thickness		0.0185	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	Core A	Unit 1		
Region 1		2.10	1.60	2.10	2.10
Region 2		2.60	2.40	2.60	2.60
Region 3		3.10	3.10	3.10	3.10

TABLE 1.3-1 (Sheet 3)

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS[b]	Comanche Peak	W. B. McGuire	Trojan
Number of coolant loops	5.0	4	4	4	4
Total steam flow, 10^6 lb/hr	5.1	15.14	15.14	15.14	5.07
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	54	54	54
Reactor coolant pumps	5.4.1				
Horsepower		7,000	7,000	7,000	6,000
Capacity, gpm		100,600	99,000	99,000	88,500
Steam generators	5.4.2				
Model		F	D	D	D
Heat transfer areas, ft.2		55,000	48,000	48,000	51,500
Number of U-tubes		5,626	4,578	4,674	3,388
Residual heat removal	5.4.7				
Initiation pressure, psig		~425	~425	~425	~400

TABLE 1.3-1 (Sheet 4)

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS(b)	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		~16	~16	~16	~16
Heater exchanger removal capacity, 106 Btu/hr		39.0	39.1	34.15	34.2
Pressurizer	5.4.10				
Heatup rate using heaters, F/hr		55	55	55	55
Internal volume, ft3		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(c)	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, ft3		819	950	950	870

TABLE 1.3-1 (Sheet 5)

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS[b]	Comanche Peak	W. B. McGuire	Trojan
Centrifugal charging pumps	6.3				
Number		2	2	2	2
Design flow, gpm		150	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		440	425	425	425
Design head, ft		2,780	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
Design head, ft		350	350	375	375
Instrumentation and controls	7.0	(a)	(a)	(a)	(a)
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

TABLE 1.3-1 (Sheet 6)

DESIGN COMPARISON

Parameter or Feature	USAR Chapter/Section	WCGS(b)	Comanche Peak	W. B. McGuire	Trojan
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	55/100	55/100	55/100

NOTES:

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

(b) The design conditions for WCGS listed in this Table have been changed by the Wolf Creek Power Rerate Program. However, since the comparisons were made at the time of application for the Operating License and are considered historic data, the Table will not be updated to reflect the new information.

(c) The capacity is reduced to 415,764 lb/hr due to the setpoint change from 2485 psig to 2460 psig.

TABLE 1.3-2

MAJOR ANALYSES NOT INCLUDED IN TOPICAL REPORTS

Analysis Description and Name	Applicable USAR 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

TABLE 1.3-2 (Sheet 2)

Analysis Description and Name	Applicable USAR Section	Previously Reviewed on Other Projects
Seismic		
a. Impedance Functions for Layered Soils [CE 970 (LUCON)]	3.7(B)	Palo Verde
b. Floor Response Spectra (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		

TABLE 1.3-2 (Sheet 3)

Analysis Description and Name	Applicable USAR Section	Previously Reviewed on Other Projects
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		
Essential Service Water Vertical Loop Chase		
a. Foundation and Substructure Walls Analysis (020544.14.01-C-002)	3.8	N/A
b. Vertical Loop Chase Structural Analysis (020544.14.01-C-005)	3.8	N/A
c. ESW Waterhammer Mitigation Analysis (EF-M-076)	9.2.1.2	N/A

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

TABLE 1.3-4

COMPLIANCE WITH NRC REGULATIONS, 10 CFR

Regulation (10 CFR)	Compliance
20.1001(a)	This regulation states the general purpose for which the Part 20 regulations are established and does not impose any independent obligations on licensees.
20.1001(b)	This regulation describes the overall purpose of the Part 20 regulations to control the possession, use, and transfer of licensed material by any licensee, so that the total dose to an individual will not exceed the standards prescribed therein. It does not impose any independent obligations on licensees.
20.1101(b)	Conformance with the ALARA principle stated in this regulation is ensured by the implementation of Company policies and appropriate Technical Specifications and health physics procedures. Chapters 11.0 and 12.0 of the USAR describe the specific equipment and design features utilized in this effort.
20.1002	This regulation states the general purpose for the Part 20 regulations and imposes no independent obligations on those licensees to which they apply.
20.1003	The definitions contained in this regulation are adhered to in appropriate Technical Specifications and procedures and in applicable sections of the USAR.
20.1004	The units of radiation dose specified in this regulation are accepted and conformed to in all applicable station procedures.
20.1005	The units of radioactivity specified in this regulation are accepted and conformed to in all applicable station procedures.
20.1006	This regulation governs the interpretation of regulations by the NRC and does not impose independent obligations on licensees.
20.1007	This regulation gives the address of the NRC and does not impose independent obligations on licensees.

TABLE 1.3-4 (Sheet 2)

Regulation (10 CFR)	Compliance
20.1201	The radiation dose limits specified in this regulation are complied with through the implementation of and adherence to administrative policies and controls and appropriate health physics procedures developed for this purpose. Conformance is documented by the use of appropriate personnel monitoring devices and the maintenance of all required records.
20.2104(d)	When required by this regulation, the accumulated dose for any individual permitted to exceed the exposure limits specified in 20.1201 is determined by the use of Form NRC-4. Appropriate health physics procedures and administrative policies control this process.
20.1204	Compliance with this regulation is ensured through the implementation of appropriate health physics procedures relating to air sampling for radioactive materials and bioassay of individuals for internal contamination. Administrative policies and controls provide adequate margins of safety for the protection of individuals against intake of radioactive materials. The systems and equipment described in Chapters 11.0 and 12.0 of the USAR provide the capability to minimize these hazards.
20.1701	Appropriate process and engineering controls and equipment, as described in Chapters 11.0 and 12.0 of the USAR, are installed and operated to maintain levels of airborne radioactivity as low as reasonably achievable.
20.1703	This regulation allows credit in estimating individual exposures for operators who are wearing respiratory protective equipment. Operating manuals contain procedures that ensure that approved respiratory protection equipment is being properly used and that plant practices are in compliance with Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."
20.1703(c)	This regulation prohibits the licensee from assigning protection factors higher than those specified in Appendix A and allows the Commission to authorize higher protection factors under certain conditions.

TABLE 1.3-4 (Sheet 3)

Regulation (10 CFR)	Compliance
20.1703(a)(2)	This regulation requires the licensee to obtain specific authorization from the NRC in order to make allowance for certain respiratory protection equipment and provides the requirements for the application for such authorization.
20.1703(c)	This regulation states the requirements for respiratory equipment which can be used as emergency devices.
20.1703(d)	The proper notification specified by this regulation will be made to the appropriate authority within the appropriate time limit.
20.1207	Conformance with this regulation is ensured by appropriate company policies regarding employment of individuals under the age of 18 and the station procedures restricting these individuals' access to the station restricted areas.
20.1301(c)	Chapter 11.0 of the USAR provides the information and related radiation dose assessments specified by this regulation.
20.1301(c)	The radiation dose rate limits specified in this regulation are complied with through the implementation of station procedures, Technical Specifications, and administrative policies which control the use and transfer of radioactive materials. Appropriate surveys and monitoring devices document this compliance.
20.1301(a)	Conformance with the limits specified in this regulation is ensured through the implementation of station procedures and applicable Technical Specifications which provide adequate sampling and analyses and monitoring of radioactive materials in effluents prior to and during their release. The level of radioactivity in station effluents is minimized to the extent practicable by the use of appropriate equipment designed for this purpose, as described in Chapter 11.0 of the USAR.
20.1301(c)	The Operating Agent has not and does not currently 20.106(c) intend to include in any license or amendment application proposed limits higher than those specified in 20.1301(a), as provided for in these regulations.

TABLE 1.3-4 (Sheet 4)

Regulation (10 CFR)	Compliance
20.1206	This regulation allows for Planned Special Exposures that are authorized by the licensee. These exposures are tracked separately with their own exposure limits. The lifetime whole body limit for Planned Special Exposure is 25 rem.
20.1302(b)(2)	Appropriate allowances for dilution and dispersion of radioactive effluents are made in conformance with this regulation, and are described in detail in Chapter 11.0 of the USAR and in appropriate reports required by the Technical Specifications.
20.1301	This regulation provides criteria by which the Commission may impose further limitations on releases of radioactive materials made by a licensee. It imposes no independent obligations on licensees.
20.1301(a)	This regulation states that the provisions of 20.2003 do not apply to disposal of radioactive material into sanitary sewage systems. It imposes no independent obligations on licensees.
20.1301(d)	This regulation pertains to licensees engaged in Uranium fuel cycle operations and does not apply to WCGS.
20.1002	This regulation clarifies that the Part 20 regulations are not intended to apply to the intentional exposure of patients to radiation for the purpose of medical diagnosis or therapy. It does not impose any independent obligations on licensees.
20.1204	Necessary bioassay equipment and procedures, including Whole Body Counting, are utilized at the station to determine exposure of individuals to concentrations of radioactive materials. Appropriate health physics procedures and administrative policies implement this requirement.
20.1501(a)	The surveys required by this regulation are performed at adequate frequencies and contain such detail as to be consistent with the radiation hazard being evaluated. Applicable health physics procedures require these surveys and provide for their documentation in such a manner as to ensure compliance with the regulations of 10 CFR 20.

TABLE 1.3-4 (Sheet 5)

Regulation (10 CFR)	Compliance
20.1502	Applicable health physics procedures set forth policies and practices which ensure that all individuals are supplied with and required to use appropriate personnel monitoring equipment. Work procedures are established to provide additional control of personnel working in radiation areas and to ensure that the level of protection afforded to these individuals is consistent with the radiological hazards in the work place.
20.1501	The terminology set forth in this regulation is accepted and conformed to in all applicable station procedures, Technical Specifications, and those portions of the station procedures in which its use is made.
20.1501(c)	This regulation pertains to personnel dosimeter processing and evaluation and is conformed to through appropriate Health Physics procedures.
20.1901(a)	All materials used for labeling, posting, or otherwise designating radiation hazards or radioactive materials, and using the radiation symbol, conform to the conventional design prescribed in this regulation.
20.1902(a)	This regulation is conformed to through the implementation of appropriate health physics procedures relating to posting of radiation areas, as defined in 10 CFR Part 20.1501.
20.1902(b)	The requirements of this regulation for "High Radiation Areas" are conformed to by the implementation of the Technical Specifications and appropriate health physics procedures. The controls and other protective measures set forth in the regulation are maintained under the surveillance of the station Health Physics group.
20.1902(d)	Each Airborne Radioactivity Area, as defined in this regulation, is required to be posted in accordance with appropriate health physics procedures. These procedures also provide for the surveillance requirements necessary to determine airborne radioactivity levels.

TABLE 1.3-4 (Sheet 6)

Regulation (10 CFR)	Compliance
20.1902	The area and room posting requirements set forth in this regulation pertaining to radioactive materials are complied with through the implementation of appropriate health physics procedures.
20.1902(e)	The container labeling requirements set forth in this regulation are complied with through the implementation of appropriate health physics procedures.
20.1904(a)	The posting requirement exceptions described in this regulation are used where appropriate and necessary at the station. Adequate controls are provided within the station health physics procedures to ensure safe and proper application of these exceptions.
20.1906	All of the requirements of this regulation pertaining to procedures for picking up, receiving, and opening packages of radioactive materials are implemented by the station procedures and appropriate health physics procedures. These procedures also provide for the necessary documentation to ensure an auditable record of compliance.
20.1801 20.1802	The storage and control requirements for licensed materials in unrestricted areas are conformed to and documented through the implementation of station health physics procedures.
20.2001	The general requirements for waste disposal set forth in this regulation are complied with through station health physics procedures, the Technical Specifications, and the provisions of the station license. Chapter 11.0 of the USAR describes the solid waste disposal system installed at the station.

TABLE 1.3-4 (Sheet 7)

Regulation (10 CFR)	Compliance
20.2002	No such application for proposed disposal procedures as described in this regulation is contemplated.
20.2003	No such plans for disposal of licensed material by release into sanitary sewage systems as provided for in this regulation are contemplated.
20.2004	No such incineration of licensed material as provided for in this regulation is contemplated.
20.2005	This regulation pertains to disposal of specific wastes and is not applicable to WCGS.
20.2006	This regulation provides requirements which are designed to control transfers of radioactive waste intended for a land disposal facility and establishes a tracking system, in addition to supplementing existing requirements concerning transfer and record keeping. These requirements are met via implementation of appropriate health physics procedures.
20.2101 20.2103	All of the requirements of this regulation are complied with through the implementation of appropriate Technical Specifications and health physics procedures pertaining to records of-surveys, radiation monitoring, and waste disposal. The retention periods specified for such records are also provided for in these specifications and procedures.
20.2201	The station has established an appropriate inventory and control program to ensure strict accountability for all licensed radioactive materials. Reports of theft or loss of licensed material are required by reference to the regulations of 10 CFR in the Technical Specifications.
20.2202	Notifications of accidents, as described in this regulation, are assured by the requirements of the Technical Specifications and appropriate health physics procedures, which also provide for the necessary assessments to determine the occurrence of such incidents.

TABLE 1.3-4 (Sheet 8)

Regulation (10 CFR)	Compliance
20.2203	Reports of overexposures to radiation and the occurrence of excessive levels and concentrations, as required by this regulation, are provided for by reference in the Technical Specifications and in appropriate health physics procedures.
20.2206	The personnel monitoring required by this regulation is provided for by the Technical Specifications. Appropriate health physics procedures establish the data base from which this report is generated.
20.2206	The report of radiation exposure required by this regulation upon termination of an individual's employment or work assignment is generated through the provisions of a station health physics procedure.
20.2301	This regulation provides for the granting of exemptions from 10 CFR Part 20 regulations, provided that such exemptions are authorized by law and will not result in undue hazard to life or property. It does not impose independent obligations on licensees.
20.2302	This regulation describes the means by which the Commission may impose upon any licensee requirements which are in addition to the regulations of Part 20. It does not impose independent obligations on licensees.
20.2401	This regulation describes the remedies which the Commission may obtain in order to enforce its regulations, and sets forth those penalties or punishments which may be imposed for violations of its rules. It does not impose any independent obligations on licensees.
26.1-26.11	Subpart A, This subpart prescibes requirements and standards for the establishment, implementation, and maintenance of fitness-for-duty (FFD) programs.
26.21-26.41	Subpart B, This subpart requires the establishment, implementation, and maintenance of FFD progrzms.
26.51-26.71	Subpart C, This subpart specifies the requirements to grant initial authorization, authorization update, authorization reinstatement, or authorization with potentially disqualifying FFD information.
26.73-26.77	Subpart D, This section defines the minimum sanctions that licensees and other entities shall impose when an individual has violated the drug and alcohol provisions of an FFD policy.

TABLE 1.3-4 (Sheet 9)

Regulation (10 CFR)	Compliance
26.81-26.119	Subpart E, This subpart contains requirements for collecting specimens for drug testing and conducting alcohol tests by or on behalf of the licensees and other entities.
26.151-26.169	Subpart G, This subpart contains requirements for the HHS-certified laboratories that licensees and other entities who are subject to this part use for testing urine specimens for validty and the presence of drugs and drug metabolites.
26.181-26.189	Subpart H, This subpart contains requirements for determining whether a donor has violated the FFD policy and for making a determination of fitness.
26.201-26.211	Subpart I, This subpart contains requirements for work hour controls and rest-break periods for select categories of workers.
26.709-26.719	Subpart N, This subpart contains the requirements for maintaining records and submitting certain reports to the NRC.
26.821-26.825	Subpart O, This subpart requires the allowance of duly authorized NRC inspectors.
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.
50.7	This regulation provides the requirements for employee protection and provides for remedy on the part of the employee who is discriminated against for engaging in certain protected activities as well as the penalty for violation.

TABLE 1.3-4 (Sheet 10)

Regulation (10 CFR)	Compliance
50.8	This regulation provides the NRC information collection requirements and specifies associated OMB approval.
50.9	This regulation provides the requirements for completeness and accuracy of information provided by the licensee to the NRC.
50.10 50.11	These regulations specify the types of activities that may not be undertaken without a license from the NRC. The Operating Agent does not propose to conduct any such activities at 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 described 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 organiza tion 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 11)

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)	An Updated Safety Analysis Report (USAR) 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 - Chapters 13.0 and 17.0. Chapter 17.0 discusses compliance with the quality assurance requirements of Appendix B. iii. Preoperational testing and initial operations - Chapter 14.0 iv. Plans for conduct of normal operations - Chapters 13.0 and 17.0. Surveillance and periodic testing is specified in the Technical Specifications. v. Plans for coping with emergencies - Emergency Plan. vi. Technical Specifications vii. Potential hazards analysis (Appendix 3B) 7. Technical qualifications - Chapter 13.0 8. Operator requalification program - Chapter 13.0

TABLE 1.3-4 (Sheet 12)

Regulation (10 CFR)	Compliance
50.34(c) 50.34(d)	The information required in these sections was submitted for Wolf Creek pursuant to Paragraph 2.790(d) 10 CFR 2, "Rules of Practice." This information includes both the Physical Security Plan and the Safeguards Contingency Plan.
50.34(e)	This regulation requires that the licensee who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan protect the plans and other Safeguards Information against unauthorized disclosure in accordance with 10 CFR 73.21.
50.34(f)	This regulation provides additional TMI-related requirements for applicants for a construction permit whose application was pending as of February 16, 1982. Wolf Creek is not impacted by this regulation.
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 prepared for implementation and 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) were prepared for implementation as required by this regulation. The RETS have taken the form prescribed by NUREG 0472, Revision 2, dated July 1979.

TABLE 1.3-4 (Sheet 13)

Regulation (10 CFR)	Compliance
50.36b	This regulation allows the NRC to attach to and incorporate in the license additional conditions to protect the environment.
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 licensees were 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:
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 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 was designed, constructed, and reviewed, including reviews by the architect-engineer, the NSSS vendor, the licensees 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 applicants was determined to be satisfactory during the hearing process at the construction permit stage. Additional information was provided in the operating license application.

TABLE 1.3-4 (Sheet 14)

Regulation (10 CFR)	Compliance
50.40(c)	The issuance of a license to the applicants was not 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 USAR and related correspondence on the record, plus the lengthy process of design, construction, and review by the 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) demonstrated that a license was not inimical to the common defense and security.
50.40(d)	The requirements set forth in this regulation were satisfied in that Environmental Reports were 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. The Operating Agent has not applied for a class 104 license, and therefore 50.41 is not applicable.
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 was provided with the operating license application, in accordance with Regulatory Guide 9.3.

TABLE 1.3-4 (Sheet 15)

Regulation (10 CFR)	Compliance
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	10 CFR 50.44 was revised in 2003. The revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release, and eliminates the requirements for hydrogen control systems to mitigate such a release. The installation of hydrogen recombiners and/or vent and purge systems required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The Commission has found that the hydrogen release is not risk-significant because the design-basis LOCA hydrogen release does not contribute to the conditional probability of a large relase up to approximately 24 hours after the onset of core damage. In addition, these systems were ineffective at mitigating hydrogen releases from risk-significant beyond design-basis accidents. License Amendment No. 157 was issued by the NRC on January 31, 2005 and deleted the Technical Specification requirements for the hydrogen recombiners and relocated the requirements for the hydrogen monitors. The WCGS power block combustible gas control system is described in USAR Section 6.2.5.2. The system is designed to maintain the hydrogen concentration in containment at a safe level following a LOCA, without purging the containment atmosphere, as specified in 10 CFR 50.44(c). The system consists of a hydrogen monitoring subsystem, and hydrogen recombiners. Sections 6.2.5.2 and 6.2.5.3 of the USAR describe the hydrogen mixing provisions and indicate that adequate mixing occurs following a LOCA without reliance on the hydrogen mixing fans. Section 6.2.5.2 of the USAR indicates that the recombiners or the hydrogen purge subsystem can be utilized in sufficient time to limit hydrogen concentration following a LOCA to less than 4 volume percent. In accordance with 50.44(d), the hydrogen contribution of the core metal-water reaction is assumed to be that resulting from reaction of 5 percent of the fuel cladding.
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	USAR 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 USAR Section 15.6.5 demonstrate conformance with 50.46.

TABLE 1.3-4 (Sheet 16)

Regulation (10 CFR)	Compliance
50.47	This regulation states that the NRC will not issue an operating license until adequate emergency plans have been assured based upon their evaluation of FEMA's assessments of state and local emergency plans and the NRC's assessment of the onsite emergency plans.
50.48	This regulation governs the fire protection plans required for operating nuclear power plants. USAR Section 9.5 describes the fire protection system designed to provide fire protection in accordance with 10 CFR 50 Appendix A, GDC3. USAR Appendix 9.5A provides a summary of the compliance with NRC Branch Technical Position APCSB 9.5-1, Appendix A. USAR Appendix 9.5B provides a summary of analyses performed to demonstrate that WCGS could meet the requirements of 10 CFR 50, Appendix R. Table 9.5E-1 of USAR Appendix 9.5E provides a design comparison to 10 CFR 50 Appendix R.
50.49	This regulation provides the requirements for environmental qualification of electrical equipment important to safety for nuclear power plants.
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.
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 WCGS 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 was effected by the inclusion of these conditions in the license when it was issued.
50.54(jj)	The regulation changes published in the Federal Register Vol. 79, No. 214, pages 65776 through 65814 on Nov. 5, 2014, relocated previous regulation 50.55a(a)(1) to 50.54(jj). As originally stated with regard to 50.55a(a)(1) and now with regard to 50.54(jj), Section 3.2 of the USAR describes compliance with this regulation.

TABLE 1.3-4 (Sheet 17)

Regulation (10 CFR)	Compliance
50.55	This regulation addresses conditions of construction permits, not operating licenses, and therefore it is not applicable to this application.
50.55a(a)	The regulation changes published in the Federal Register Vol. 79, No. 214, pages 65776 through 65814 on Nov. 5, 2014, relocated standards and other documents incorporated by reference from 50.55a(b) to a new 50.55a(a). Therefore, 50.55a(a) provides guidance concerning the approved edition and addenda of ASME Codes and IEEE Standards that are incorporated by reference in the regulations. Note: The previous 50.55a introductory text and 50.55a(a)(2), which specified the requirements for systems and components and protection systems for nuclear power reactors, were moved into 50.55a(b), 50.55a(c), 50.55a(d), 50.55a(e), 50.55a(f), 50.55a(g) and 50.55a(h).
50.55a(b)(1)	This regulation provides conditions on use of ASME BPV Code Section III.
50.55a(b)(2)	This regulation provides conditions on use of ASME BPV Code Section XI
50.55a(b)(3)	This regulation provides conditions on use of ASME OM Code.
50.55a(b)(4)	This regulation provides conditions on use of ASME BPV Code Section III Code Cases for design, fabrication and materials.
50.55a(b)(5)	This regulation provides conditions on use of ASME BPV Code Section XI Code Cases for inservce inspection and repair/replacement activities.
50.55a(b)(6)	This regulation provides conditions on use of ASME OM Code Code Cases.
50.55a(c)	This regulation provides the code requirements for components which are part of the reactor coolant pressure boundary and for components which are connected to the reactor coolant system, including inservice inspection requirements.

TABLE 1.3-4 (Sheet 18)

Regulation (10 CFR)	Compliance
	Design and fabrication of the reactor vessel, reactor coolant system piping, reactor coolant pumps, and reactor coolant system valves were carried out in accordance with ASME Section III as described in Section 5 of the USAR.
50.55a(d) 50.55a(e)	These regulations apply to nuclear power plants whose application for a construction permit was docketed after May 14, 1984.
50.55a(f)	Inservice testing (IST) requirements delineated in this part are specified in the Technical Specifications.
50.55a(g)	Inservice inspection (ISI) requirements delineated in this part are specified in the Technical Requirements Manual and Inservice Inspection Program.
50.55a(z)	The regulation changes published in the Federal Register Vol. 79, No. 214, pages 65776 through 65814 on Nov. 5, 2014, relocated te allowance for proposed alternatives contained in the previous 50.55a(a)(3) to a new paragraph 50.55a(z). 50.55a(z) allows for proposed alternatives to 50.55a paragraphs (b), (c), (d), (e), (f), (g), and (h).
50.55(h)	As discussed in Chapter 7.0, Section 7.1, the protection systems meet IEEE 279-1971.
50.55b	This regulation has been revoked. 43 Fed. Reg. 49775.
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 required the Commission to make certain findings prior to the issuance of the operating license.
50.57(b)	The license, as issued, contains appropriate conditions to ensure that items of construction or modification were completed on a schedule acceptable to the Commission.
50.57(c)	This regulation provided for a low-power testing license.
50.58	This regulation provided 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.

TABLE 1.3-4 (Sheet 19)

Regulation (10 CFR)	Compliance
50.60	This regulation provides the acceptance criteria for fracture prevention measures for nuclear power reactors during normal operation. Section 5.3 of the USAR details vessel material parameters in terms of the fracture toughness requirements set forth in Appendices G and H of 10 CFR 50.
50.61	This regulation provides the fracture toughness requirements for protection against pressurized thermal shock events. Fracture toughness for the reactor pressure vessel is addressed in Section 5.3 of the USAR. Compliance with Regulatory Guide 1.99 is addressed in Appendix 3A of the USAR.
50.62	This regulation specifies the requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water nuclear power plants.
50.63	This regulation pertains to the Station Blackout rule.
50.64	This regulation pertains to non-power reactors only and is not applicable to WCGS.
50.65	This regulation requires the implementation of a program to monitor the effectiveness of maintenance programs by monitoring performance of plant SSCs. Plant procedures implement and control this program.
50.68	This regulation provides the licensee with eight requirements that may be complied with in lieu of compliance with 10CFR70.24 for criticality monitoring. WCGS complies with this regulation.
50.70	The Commission has assigned resident inspectors to WCGS and space was provided in conformance with 50.70(b)(1) through (3).
50.71	Records are and will be maintained and reports will be made in accordance with the requirements of sections (a) through (e) of this regulation and the license.
50.72	This regulation provides the immediate notification requirements for operating nuclear power reactors.
50.73	This regulation requires the licensee to submit Licensee Event Reports for certain specific events.
50.74	This regulation requires the licensee to notify the NRC pertaining to a change in Reactor Operator or Senior Reactor Operator status.
50.78	This regulation pertains to holders of construction permits and does not apply to WCGS.
50.80	This regulation provides that licenses may not be transferred without NRC consent. No application for transfer has been made by the WCGS Licensees.

TABLE 1.3-4 (Sheet 20)

Regulation (10 CFR)	Compliance
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 WCGS 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 regarding no significant hazards considerations, notices for public comment and state consultation.
50.92	This regulation provides guidance to the NRC in issuing license amendments including no significant hazards consideration determinations.
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.
50.120	This regulation provides guidance for the training and qualifications of Nuclear Power Plant Personnel. This regulation establishes the requirements for a training program. Appropriate procedures control this program.
Appendix A	USAR Section 3.1 discusses the extent to which the design criteria for the WCGS's 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 USAR sections, and are located by reference.

TABLE 1.3-4 (Sheet 21)

Regulation (10 CFR)	Compliance
Appendix B	Chapter 17.0 describes in detail the provisions of the operating Quality 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.
Appendix E	This appendix specifies requirements for emergency plans. Emergency plans were 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 were factored into the emergency plans for the WCGS utilities. The Emergency Plan and associated facilities (EOF and Alternate TSC) were updated in 2011 to comply with revised regulations pertaining to “Enhancements to Emergency Preparedness Regulations” (Reference Section 18).
Appendix F	This appendix applies to fuel reprocessing plants and related waste management facilities, not to power reactors such as those found in WCGS plants and is, therefore, not applicable.
Appendix G	Fracture toughness compliance can be found in USAR 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 USAR Chapter 5.0.

TABLE 1.3-4 (Sheet 22)

Regulation (10 CFR)	Compliance
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. USAR 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 USAR 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 USAR 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 USAR Section 15.6.5) is in conformance with Appendix K requirements.
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 WCGS 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, WCGS used a common Safety Analysis Report prior to issuance of the first update after receipt of the Operating License; however, where site specific needs were addressed Addenda were included for reference. The two reports have been merged into one Wolf Creek Specific Safety Analysis Report.

TABLE 1.3-4 (Sheet 23)

Regulation (10 CFR)	Compliance
Appendix O	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.
Appendix R	Appendix R delineates the fire protection program for nuclear power facilities operating prior to January 1, 1979. Appendix 9B of the USAR provides a summary of analyses performed to demonstrate that WCGS could meet the requirements of Appendix R. Table 9.5E-1 of the USAR appendix 9.5E provides a design comparison of WCGS to Appendix R.
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	This regulation specifies that Subpart A of Part 51 implements section 102(2) of the NEPA act of 1969, as amended.
51.3	This regulation states that in any conflict between the general rule and Subpart A of Part 51 (or other applicable part of this chapter) the special rule governs.
51.4	The definitions contained in this regulation are adhered to in all appropriate documents.
51.5	This regulation governs the interpretation of regulations and does not impose independent obligations on licensees. This regulation specifies that interpretations of the regulations in Part 51 are not authorized other than a written interpretation by the General Counsel.
51.6	This regulation specifies the authority of the NRC in granting exemptions and does not impose independent obligations on licensees.
Subpart A	
51.10	This regulation provides the purpose and scope of Subpart A which is to implement Section 102(2) of the National Environmental Policy Act (NEPA) in a manner consistent with NRC's domestic licensing and regulatory authority.

TABLE 1.3-4 (Sheet 24)

Regulation (10 CFR)	Compliance
51.11	[Reserved]
51.12	This regulation states that subpart A applies to the NRC's ongoing environmental work and does not require retroactive measures for environmental reports or supplements filed prior to June 7, 1984. This regulation does not impose independent obligations or licensees.
51.13	This regulation permits the NRC to take immediate action in emergencies where the health and safety of the public may be adversely affected without observing the NEPA regulations. This regulation does not impose independent obligations on licensees.
51.14	This regulation provides a pertinent definitions related to NEPA which are adhered to in all appropriate documents.
51.15	This regulation provides the requirements for establishing time schedules for NRC-NEPA processes.
51.16	This regulation provides the requirements for the submittal of proprietary information.
51.17	This regulation indicates that the NRC has submitted the information requirements to OMB related to this part of the Code of Federal Regulations.
51.20	This regulation sets forth the requirements for an applicant for filing an Environmental Impact Report. These requirements were satisfied during the review of the Environmental Reports that were submitted with the application for a construction permit and the application for an operating license.
51.21	This regulation specifies that all licensing and regulatory actions subject to Part 51 require an environmental assessment except those identified in 51.20(b) as requiring an environmental impact statement and those identified in 51.22(c) as categorical exclusions.
51.22	This regulation sets forth the criterion for an identification of licensing and regulatory actions eligible for categorical exclusion.

TABLE 1.3-4 (Sheet 25)

Regulation (10 CFR)	Compliance
51.23	This regulation indicates that the NRC will require no environmental reports, impact statements, assessments or other analyses in connection with issuance of license amendment for storage of spent fuel up to 30 years beyond the expiration of reactor operating license.51.25 This regulation states than an appropriate NRC staff director will determine when a categorical exclusion environmental impact statement or environment assessment should be prepared.
51.26	This regulation states that when an NRC staff director determines that an environmental impact statement will be prepared a notice of intent will be published in the Federal Register and a scoping process will be conducted.
51.27	This regulation describes the requirements for the Notice of Intent as required by 10 CFR 51.26.
51.28	This regulation specifies who the NRC staff director shall invite to participate in the scoping process for an environmental impact statement.
51.29	This regulation provides the requirements for the scoping process for an environmental impact statement.
51.30	This regulation sets fourth the requirements for an environmental assessment by the NRC.
51.31	This regulation states that the NRC staff director will make the determination (based upon environmental assessments) whether to prepare an environmental impact statement or a finding of no significant impact.
51.32-51.35	These regulations provide the requirements for a finding of no significant impact by the NRC.
51.40	This regulation provides guidance to prospective applicants or petitioners for rulemaking for consultation with the NRC staff.5
51.41	This regulation gives the NRC authority to require permit or license applicants amendment applicants or petitioners to submit information useful in aiding NRC compliance with Section 102(2) of NEPA.

TABLE 1.3-4 (Sheet 26)

Regulation (10 CFR)	Compliance
51.45-51.69	These regulations set fourth the requirements for environmental reports.
51.70-51.125	These regulations set fourth the requirements for environmental impact statements.
Appendix A to Subpart A	This Appendix to Part 51 Subpart A provides the format for presentation of material in environmental impact statements.
55.1-55.71	These regulations set forth the requirements for nuclear power plant operator's licenses.
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 sets forth the requirements for communications with the NRC regarding special nuclear materials and includes with addresses for the Director, Office of Nuclear material Safety and Safeguards.
70.6	This regulation governs the interpretation of regulations and does not impose any independent obligations on licensees.
70.7	This regulation prohibits discrimination against and otherwise protects employees of all licensee engaged in certain protected activities.

TABLE 1.3-4 (Sheet 27)

Regulation (10 CFR)	Compliance
70.8	This regulation sets forth the information collection requirements and specifies OMB approval.
70.9	This regulation addresses the completeness and accuracy of information provided to the NRC.
70.11 - 70.14	These regulations specify those persons exempted from complying with Part 70. The licensees are not exempt from complying with the applicable requirements of Part 70.
70.15	Reserved.
70.18 - 70.20b	These regulations list types of licenses issued for special nuclear material. WCGS adhered to all applicable requirements.
70.21	This regulation sets forth the requirements concerning the filing of special nuclear material license applications. The requirements of this regulation were satisfied.
70.22	This regulation sets forth the requirements concerning the contents of special nuclear material license applications. The requirements of this regulation were 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. On June 24, 1997, the NRC issued to WCNOC an exemption from the requirements of 10CFR70.24. On November 12, 1998 the NRC issued 10CFR50.68, which provides eight criteria that may be followed in lieu of criticality monitoring per 10CFR70.24 and revised 10CFR70.24 to make any exemption ineffective so long as the licensee elects to comply to 10CFR50.68.
70.31	This regulation lists guidelines for the Commission to follow in issuing a license.
70.32	This regulation defines the conditions by which the licensee must abide in order to keep the special nuclear materials license. The health physics program found in Chapter 12.0, Section 12.5 provides information relating to the compliance of this regulation.

TABLE 1.3-4 (Sheet 28)

Regulation (10 CFR)	Compliance
70.33 - 70.35	These regulations dictate procedural requirements for renewing or amending a license. The Operating Agent 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 WCGS.
70.37	This is a disclaimer of warranty and does not affect the Licensees.
70.38	This regulation sets forth the requirements for expiration and termination of licenses.
70.39	This regulation sets guidelines for the manufacture of source material and does not apply to power units such as WCGS.
70.41	This regulation provides the requirements for authorized use of special nuclear material.
70.42	This regulation provides guidance on the transfer of special nuclear material. WCGS follows 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.

TABLE 1.3-4 (Sheet 29)

Regulation (10 CFR)	Compliance
70.53	This regulation sets forth the requirements for submitting Material Status Reports. Where applicable, proper procedures were 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.
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 have been 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 Operating Agent 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.

TABLE 1.3-4 (Sheet 30)

Regulation (10 CFR)	Compliance
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.
Subpart A 71.0	This regulation establishes the purpose, scope and applicability for the Part 71 regulations and does not impose any independent obligations on licensees .
71.1	This regulation provides the address for communications with the NRC.
71.2	This regulation states that only written interpretations of Part 71 by the NRC's General Counsel are binding.
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	This regulation states the information collection requirements submitted for OMB approval.
71.6a	This regulation governs the completeness and accuracy of information provided to the NRC.
Subpart B 71.7-71.10	These regulations delineate the exemptions from Part 71.

TABLE 1.3-4 (Sheet 31)

Regulation (10 CFR)	Compliance
Subpart C 71.12	This regulation issues a general license for shipment in certain NRC approved containers and packages provided the licensee has an approved QA program. QA programs for WCGS are filed with the NRC during the operations phase as part of the license application.
71.13-71.24	These regulations set forth the general license requirements for shipments in specific packages or contains under a general license.
Subpart D 71.31-71.39	These regulations provide the requirements for an application for a proposed packaging design including the contents of the application, package description, package evaluation, and quality assurance requirements.
Subpart E 71.41-71.65	These regulations provide the requirements for packaging radioactive material for transport. Compliance with each of the individual parts and paragraphs was demonstrated in the license application proceedings. These requirements of these regulations are referenced as the standards.
Subpart F 71.71-71.77	These regulations address the testing requirements for packages, containers and special form radioactive materials.
Subpart G 71.81-71.99	These regulations provide various operating controls and procedures pertinent to the transport and packaging of radioactive materials.
Subpart H 71.101-71.137	These regulations set forth the quality assurance requirements applying to design, purchases, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and modification of components of packaging (for radioactive materials) which are important to safety.
Appendix A	This regulation establishes the procedure for obtaining activity values A and A2 to be used in packaging and shipping processes.

TABLE 1.3-4 (Sheet 32)

Regulation (10 CFR)	Compliance
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 this is the only special nuclear material for which WCGS is currently licensed, it is exempt from the requirements of 73.20, 73.25, 73.26, 73.27, 73.45, 73.46, 73.70, and 73.72. This regulation sets forth the information requirements established by the Commission and specifies OMB approval.
73.8	This regulation specifies the information collection requirements and submitted for OMB approval.
73.20	The licensee is exempt from the requirements of this regulation. See 73.6.
73.21	This regulation sets forth the requirements for the protection of safeguards information. These requirements are addressed by the WCGS Site Security Plan.
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 use the requirements of this regulation for reference when such requirements are applicable.

TABLE 1.3-4 (Sheet 33)

Regulation (10 CFR)	Compliance
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 physical protection during the transport of irradiated reactor fuel.
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 at fixed sites. Physical security systems are provided and maintained to provide adequate physical protection against sabotage and theft of special nuclear material. In addition, a safeguards contingency plan was prepared in accordance with the criteria in Appendix C of this part, submitted for Commission approval and implemented.
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 were implemented to provide for physical barriers, access control, detection aids, and communications along with a physical security organization that ensures physical protection. The requirements, as prescribed by this regulation, has been satisfied to the extent practicable.
73.57	This regulation establishes the requirements for Criminal History Checks of individuals granted unescorted access to a nuclear power facility or access to Safeguards Information by power reactor licensees. These requirements are addressed in the WCGS Site Security Plan.
73.60	This regulation applies to non-power reactors and thus does not apply to WCGS.

TABLE 1.3-4 (Sheet 34)

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. At this time the licensee is exempt from the requirements of this regulation (See 73.6).
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. At this time the Licensee is exempt from the requirements of this regulation (See 73.6).
73.73	This regulation establishes the requirements for advance notice and protection of export shipments of special nuclear material of low strategy significance and does not apply to WCNOC since it is not licensed to export special nuclear material.
73.74	This regulation establishes the requirements for advance notice and protection of import shipments of nuclear materials from countries that are not party to the Convention of Physical Protection of Nuclear Material. It does not apply to WCNOC since it is not licensed to import special nuclear material.
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.

TABLE 1.3-4 (Sheet 35)

Regulation (10 CFR)	Compliance
Appendix B	The general criteria for security personnel are outlined in this appendix. The principles in this regulation were factored into the WCGS security plans.
Appendix C	This regulation sets forth the requirements for licensee safeguards contingency plans. This plan has been developed and implemented.
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.
Appendix E	This regulation specifies the levels of physical protection to be applied in international transport of Nuclear material and does not apply to WCGS, since WCNOC is not involved in such transport.
Appendix F	This regulation merely lists the nations that are parties to the convention on the physical protection of nuclear material.
Appendix G	This regulation sets forth the requirements for reportable safeguards events and is addressed by the WCGS Site Emergency Plan. 3
100.1	This regulation is explanatory and does not impose independent obligations on licensees.
100.2	This regulation is explanatory. WCGS is not novel in design and is not unproven as a prototype or pilot plant.
100.3	The definitions contained in this regulation are adhered to in all appropriate documents.
100.8	This regulation sets forth the information requirements established by the NRC and specifies OMB approval.

TABLE 1.3-4 (Sheet 36)

Regulation (10 CFR)	Compliance
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 USAR. The exclusion area, low population zone, and population center distance are provided and described. The USAR also describes the characteristics of reactor design and operation.
100.11	Exclusion areas have been established, as described in Section 2.1. The low population zone has been established in accordance with this requirement. The USAR 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

Kansas City Power & Light Company (KCPL), Kansas Electric Power Cooperative, Incorporated (KEPCo), and Kansas Gas and Electric Company (KG&E) are co-owners of WCGS, having 47, 6 and 47 percent participation, respectively. For the purposes of the operating license application KCPL and KG&E were considered co-applicants for Wolf Creek. KG&E was the lead applicant and was initially responsible for the design, construction and operation of WCGS.In Amendment No. 4 to the Operating License, Wolf Creek Nuclear Operating Corporation (WCNOC) was authorized to act as agent for KCPL, KG&E, and KEPCo and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility. WCNOC will be referred to as the Operating Agent in matters where the corporation acts in the interests of all three licensees.

KCPL is an independent investor-owned utility with headquarters in Kansas City, Missouri that provides electric service in a 5,700 square mile area of western Missouri and eastern Kansas. KCPL serves 331,000 customers and also provides electricity at wholesale to eight communities, three electric cooperatives and two utilities.

KEPCo is a rural electric cooperative association of 25 member cooperatives which provide electric service to the rural areas of Kansas. KEPCo is headquartered in Topeka, Kansas and was incorporated in 1975. KEPCo serves approximately 90,000 meters to provide electricity to nearly 325,000 consumers located throughout Kansas.

KG&E is an independent investor-owned utility that provides electric service in an 8,100 square mile area of south central and southeast Kansas. General offices of the company are in Wichita, Kansas. KG&E serves 212,000 retail customers and also provides, at wholesale, part or all of the electricity sold by 24 municipal electric systems and by 8 rural electric cooperatives.

The owners have over 97 years of experience in the operation of electric generating plants. The owners do not maintain engineering and construction staffs for power plants but do engage reputable engineering and construction firms for these purposes. As of January 1, 1980, the owners had in operation 11 power stations in which they are full or partial owners with a total system generator nameplate capacity of 4420 MW.

Great Plains Energy Incorporated (Great Plains) and Westar Energy, Inc. (Westar) through subsidiaries Kansas City Power and Light (KCP&L) and Kansas Gas and Electric Company (KG&E), respectively owns 47% of WCNOC and WCGS. The Great Plains and Westar merger was finalized June 4, 2018, and Westar became a wholly-owned subsidiary of Great Plains. As a result of this merger, Great Plains owns a combined 94% of WCNOC and WCGS. The remaining 6% ownership interest is held by Kansas Electric Power Cooperative, Inc. (KEPCO).

1.4.2 SNUPPS

KCPL, KEPCO, KG&E and Union Electric Company 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 (Wolf Creek and Callaway).

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 established 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 to provide architect-engineering services for the power block; 2) Westinghouse to supply two identical NSSSs and, under a separate contract, and to supply the first fuel loading; 3) General Electric to supply two identical turbine generators and directly related auxiliaries; and 4) Nuclear Projects, Inc., for project management and for furnishing the technical and administrative staff to represent the utility owners and to engage consulting services and contractors, as required. These contracts were administered as one project.

Nuclear Projects, Inc., established in May 1974, furnished 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 were 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 had responsibility for the power block 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 retained a site architect-engineer (Sargent & Lundy for WCGS) to design non-standardized facilities. Construction management at the Wolf Creek site 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 located 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 supplying the NSSS and first fuel load for WCGS.

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 Gaithersburg 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 covered by the application, except for certain environmental studies, Sargent & Lundy Engineers (S&L) was retained as the architect-engineer and design consultant. In general, the responsibilities 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. They were responsible for the design of site-related systems and facilities which are nonseismic Category I and for seismic Category I dams, canals, ponds and earthwork.

Sargent & Lundy is an independent consulting engineering organization founded in Chicago in 1891. The firm has specialized in the design of generation, transmission and distribution systems for steam utilization, electric power, and related facilities. The firm has provided the complete engineering services for more than 600 turbine-generator units with a total installed capacity of 53,000 MW. Of this, some 9,800 MW is nuclear generating capacity. Table 1.4-1 lists the nuclear plants S&L itself has completed or is currently designing. Table 1.4-2 lists other nuclear plants Sargent & Lundy has had partial responsibility for.

1.4.7 CONSULTANT FIRMS

1.4.7.1 SNUPPS Consultants

Principal consultants for the SNUPPS portions (powerblock) of the WCGS and their related responsibilities are:

a. Quadrex Corporation (formerly Nuclear Services Corporation

   This consultant assisted the SNUPPS staff to coordinate the owners' preparation of power block operating procedures and review and approval action by the owners 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, is 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 SNUPPS unit 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 the SNUPPS plants.

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 included drafting specifications for a loose parts monitor, carrying out an independent review of Bechtel's calculations for shielding the reactor cavity, and reviewing the SNUPPS units' 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 SNUPPS project to assist in bid evaluations and selection of the Standard Plant A/E. This consultant remained available on an as-needed basis, and provided 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 WCGS 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, including advising WCGS 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 SNUPPS utilities to develop 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 SNUPPS control room, emphasizing human factors considerations.

1.4.7.2 WCGS Specific Consultants

a. Dames & Moore

The independent consulting firm of Dames & Moore was retained to perform site investigations relating to demography, geography and land use, meteorology, hydrology, geology and seismology. Having performed such safety-realted and environmental impact related investigations for over 75 nuclear power plant sites, Dames & Moore is an acknowledged leader in the field of site investigations related to nuclear plant construction.

Listed below are some of the nuclear power plants for which Dames & Moore has performed geotechnical and/or environmental investigations:

U.S. NORTHEAST

Atlantic	James A. Fitzpatrick	Salem
Burlington	Limerick	Seabrook
Calvert Cliffs	Newbold Island	Shoreham
Douglas Point	Nine-Mile Point	Somerset
Forked River	Oyster Creek	Sterling
Hope Creek	Peach Bottom	Summit
Indian Point	Perryman	Susquehanna
Jamesport	Robert E. Ginna	Yankee

U.S. MIDWEST

Bailly N 1	Dresden	Midland
Braidwood	Duane Arnold	Monticello
Byron	Fermi	Palisades
Carroll County	Fort Calhoun	Point Beach
Central Iowa	Greenwood	Prairie Island
Clinton	Haven	Quad-Cities
Cooper	Kewaunee	Zimmer
Davis-Besse	LaSalle	Zion
Donald C. Cook	Marble Hill	

U.S. SOUTH

Brunswick	Joseph M. Farley	Shearon Harris
Catawba	McGuire	South Dade
Cherokee	North Anna	St. Lucie
Crystal River	Nuclear One	Surry
DeSoto	Oconee	Turkey Point
Edwin I. Hatch	Perkins	Virgil C. Summer
Isolte	Robinson	

U.S. SOUTHWEST

Allens Creek	Comanche Peak	South Texas Project

U.S. WEST AND NORTHWEST

Humboldt Bay	Skagit	Trojan
San Onofre	Sundesert	

b. Ecological Analysts, Inc.

The private research and service organization of Ecological Analysts, Inc. was retained to collect, analyze and report environmental data related to the environmental impact of the WCGS. These studies included biological, chemical, and radiological investigations. The organization was founded in 1978 and was formerly part of Industrial BIO-TEST Laboratories, Inc. (1968-75), NALCO Chemical Company (1975-78) and Hazleton Environmental Sciences Corporation (1978-80). The present full-time staff includes more than 160 scientists and technicians. Listed below are some of the nuclear power plants for which Ecological Analysts, Inc. has performed environmental studies and investigations:

Bailly	Fort Calhoun
Catawba	Kewaunee
Clinton	LaSalle
Cooper	Quad-Cities
Dresden	Wm. H. Zimmer
Duane Arnold	Zion

c. Hoad Engineers, Incorporated

Hoad Engineers, Incorporated (HEI), a wholly-owned subsidiary of Blount, Incorporated, was retained to provide the design, plans and specifications along with equipment procurement and construction management services for the WCGS security system. The Operating Agent took over responsibility for these activities in late 1982. HEI prepared and provided documents for the Security Plan, Safeguard Contingency Plan and Security Training and Qualifications Plan. Since its inception in 1953, HEI has primarily served the investor-owned utilities in all of the usual engineering and architectural disciplines.

HEI has been engaged in providing the design plans, specifications and equipment procurement services for security systems at two nuclear power plants for the Consumers Power Company in Jackson, Michigan.

d. Phoenix Power Services, Incorporated

Phoenix Power Services, Incorporated was retained to assist in preparing the early drafts of the Emergency Plan for WCGS. Phoenix has performed extensive work in the area of Emergency Planning.

e. Professional Loss Control, Inc.

Professional Loss Control, Inc., (PLC), was retained to review the Fire Plan and implementing procedures and advise The Operating Agent of their adequacy. Founded in 1976, PLC provides services in the fields of fire protection, safety and environmental engineering. Listed below are some of the utilities for which PLC has performed loss control services

Carolina Power and Light Company
Florida Power Corporation
Jersey Central Power and Light Company
Maine Yankee Atomic Power Company
Niagara Mohawk Power Corporation
Portland General Electric Company
Power Authority of the State of New York
Rochester Gas and Electric Company
Tennessee Valley Authority
Washington Public Power Supply System
Wisconsin Electric Power Company
Yankee Atomic Electric Company

1.4.8 CONSTRUCTOR

Daniel International Corporation, herein referred to as Daniel, was assigned construction and construction management responsibilities for WCGS.

Daniel's scope of work consisted of receiving design information as prepared by Bechtel, Westinghouse and Sargent & Lundy; receiving manufactured items and materials as procured by Bechtel and Westinghouse; procuring additional bulk materials and consumable items; procuring the services of various subcontractors; planning and scheduling the activities of the construction forces and directly supervising the construction forces to assemble the power plant in accordance with the design.

Daniel Construction Company was awarded the ASME Certificate of Authorization to perform nuclear code construction (N stamp) on September 11, 1973, following an ASME implementation and enforcement audit of Daniel's Quality Assurance Program.

Daniel's experience, past and present, includes construction of nuclear and fossil fueled power plants. The first project of this nature was construction of the nuclear power Carolinas-Virginia Tube Reactor at Parr, South Carolina.

This facility operated several years as a prototype plant. Other nuclear power plants under construction, in operation or on which Daniel is performing maintenance are specified below:

Callaway	Oconee
H. B. Robinson	Shearon Harris
Joseph M. Farley	Virgil Summer

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 WCGS rests with The Operating Agent.

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 included 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, the individual utilities arranged to procure this equipment.

The NSSS portion of the facility was procured by individual contract between The Operating Agent and the NSSS supplier. Similarly, the turbine generator is obtained by direct contract between the turbine generator supplier and The Operating Agent. 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.

Bechtel Power Corporation was also responsible for the design, engineering and procurement of the portions of the WCGS seismic Category I essential service water systems (ESWS) which lie outside the power block.

The design and engineering of all SSCs associated with the ESW vertical loops and chase were not part of the original SNUPPS standard plant design. Loops and chase were added to both trains of the ESW. Design and specification of these SSCs were obtained by direct contract between WCNOC and several contributing contractors. Interfaces were established and monitored by WCNOC to ensure compatibility in design between power block SSCs and the ESW vertical loops and chase. WCNOC is ultimately responsible for the the design, engineering, and procurement of the SSCs associated with the ESW vertical loops and chase.

1.4.9.3 SNUPPS Staff

The SNUPPS Staff functioned as an extension of the management, engineering, and operations organizations of the SNUPPS Utilities. During 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 the Standard Plant Architect/Engineer. This required a close relationship between the SNUPPS Staff and SNUPPS Utilities, which was achieved by frequent communications and regularly scheduled meetings of the various committees and groups.

1 4.9.4 Site Architect/Engineer

All systems, equipment and structures outside the power block except for the ESW components and station security-related systems were designed or specified by Sargent & Lundy. The ultimate heat sink was the only seismic Category I structure designed by Sargent & Lundy.

Interfaces were established and monitored by Bechtel Power Corporation to ensure compatibility in design between power block and site-related systems and equipment.

1.4.9.5 Security Consultant

The station security-related systems were initially designed and specified by Hoad Engineers, Incorporated, who was retained to design a security system for WCGS to meet the requirements of 10 CFR 73.55.

TABLE 1.4-1

NUCLEAR POWER PLANTS COMPLETED OR

CURRENTLY UNDER DESIGN BY SARGENT & LUNDY

UNIT	NOMINAL GROSS* RATING (MWe)	POWER OPERATION
EBWR	5	1956
Elk River	22	1962
La Crosse	60	1969
SEFOR	20 (MWt)	1969
Dresden 2	850	1969
Dresden 3	850	1971
Quad-Cities 1	850	1971
Quad-Cities 2	850	1972
Zion 1	1085	1973
Zion 2	1085	1973
Fort St. Vrain, Unit 1	330	1973
La Salle County Station, Unit 1	1122	1982
La Salle County Station, Unit 2	1122	1983
Byron Station, Unit 1	1175	1984
Byron Station, Unit 2	1175	1985
Braidwood Station, Unit 1	1175	1985
Clinton Power Station, Unit 1	992	1985
Braidwood Station, Unit 2	1175	1986
Clinton Power Station, Unit 2	992	Cancelled
Bailly Nuclear 1	685	Cancelled
Marble Hill, Unit 1	1190	Cancelled
Marble Hill, Unit 2	1190	Cancelled
Wm. H Zimmer, Unit 1	840	Cancelled
Carroll County Station, Unit 1	1175	Future
Carroll County Station, Unit 2		Future

*Note that this is a gross rating, not a net rating.

TABLE 1.4-2

OTHER NUCLEAR POWER PLANTS

WITH PARTIAL SARGENT & LUNDY DESIGN RESPONSIBILITY

UNIT	NOMINAL GROSS* RATING (MWe)	SARGENT & LUNDY RESPONSIBILITY
Bellefonte, Unit 1 Bellefonte, Unit 2	1235 1235	Consulting Services on Containment Design (Sructural Analysis of post-tension containment structures).
D. C. Cook, Unit 1 D. C. Cook, Unit 2	1083 1118	Design of water intake structures, crib house, turbine room foundations and miscellaneous electrical consulting services.
Enrico Fermi, Unit 2	1223	Design of Residual Heat Removal complex, shielding design calculations and class 1 piping analysis.
Kaiseraugst	992	All design inside the containment including the containment itself.
Point Beach, Unit 1 Point Beach, Unit 2	519 519	Consulting services on water intake structures.

*Note that this is a gross rating, not a net rating.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

One of the design bases for WCGS 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 summary 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 WCGS design were successfully tested, using silicone foam sealant. Details of the tests 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 WCGS and may lead to changes in safety analyses or modes of operation. Further progress on these programs is not required for safe operation of WCGS.

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.

Two general types of model boiler tests were conducted by Westinghouse (1) to confirm the thermal-hydraulic analyses used for the Model-F steam generator and (2) to explore the potential for corrosion and other water-chemistry induced effects in the Model-F steam generator. The initial series of each of these tests were completed prior to startup of WCGS. Further information on the steam generator test programs of Westinghouse is given in Section 5.4.2.

1.5.3.2 Generic Programs of Bechtel

Wolf Creek through 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 at WCGS 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 was equipped with special pressure and temperature instrumentation that enabled thermal-hydraulic performance characteristics to be measured during the early stages of power operation. The objective of these tests was primarily to confirm Westinghouse's design analyses. The test results are proprietary but are 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 Wolf Creek USAR incorporates, by reference, various topical reports as part of the application. 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.

Amendment No. 89 relocated various Technical Specifications to the USAR. The relocated Technical Specifications have subsequently been incorporated into the Technical Requirements Manual (TRM) with the same format and content they possessed in the Technical Specifications. The TRM is a physically separate document from the USAR, but by this specific reference, it is considered part of the USAR and is thereby incorporated by reference. Implementation of, and revision to, the TRM is controlled through administrative procedures.

Controlled drawings were removed from the USAR at Revision 17. The drawings are considered incorporated by reference. Table 1.6-3 identifies the controlled drawings that are incorporated by reference and also provides a cross-reference of the controlled drawings to the respective USAR figure number. The contents of the drawings are controlled by WCGS procedures.

Appendix 9.5B, “Fire Hazard Analyses”, was removed from the USAR at Revision 19, and is considered incorporated by reference. Table 1.6-4 identifies the section and provides a cross-reference of the controlled document, E-1F9905, “Fire Hazard Analysis”, which supersedes the information originally provided in Appendix 9.5B of the USAR. The contents of the Fire Hazard Analysis is controlled by WCGS procedures.

Chapter 17.2 Quality Assurance, was removed from the USAR at Revision 21. Chapter 17.2 is considered incorporated by reference. The Quality Program Manual supercedes the information originally provided in Chapter 17.2 of the USAR. The contents of the Quality Program Manual are controlled by WCGS procedures.

Table 3.11(B)-1, Plant Environmental Normal Conditions; Table 3.11(B)-2, Environmental Qualification Parameters for SNUPPS NUREG-0588 (LOCA, MSLB and HELB); Table 3.11(B)-3, Identification of Safety-Related Equipment and Components: Equipment Qualification; Table 3.11(B)-4, Containment Worst Case Radiation Levels (MRADs); Table3.11(B)-5, Containment Spray Requirements; Table 3.11(B)-8, Exemptions from NUREG-0588 Qualification; Table 3.11(B)-10, Equipment Added for NUREG-0737; Figures 3.11(B)-1 through 3.11(B)-49, were removed from the USAR at Revision 28. The listed Tables and Figures are considered incorporated by reference. EQSD-I, EQ Summary Document Section I Program Description, and EQSD-II, EQ Master List Section II, supercedes the information provided by listed Tables and Figures. The contents of EQSD-I and EQSD-II are controlled by WCGS procedures.

TABLE 1.6-1

BECHTEL TOPICAL REPORTS INCORPORATED BY REFERENCE

Bechtel Topical Report No.	Title	Revision Number	USAR 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 3A	3.8	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 REPORTS INCORPORATED BY REFERENCE

Bechtel Topical Report No.	Title	Revision Number	USAR 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 3.8	3.6	5/74	A
BN-TOP-3	Performance and Siz-ing 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) See Notes on Table 1.6-2

TABLE 1.6-2

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report 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	0
WCAP-2850-L(P) WCAP-7916	"Single Phase Local Boiling And Bulk Boiling Pressure Drop Correlations"	Rev. 0	4.4	5/66	0
WCAP-2923	"In-Pile Measurement of UO2 Thermal Conductivity"	Rev. 0	4.4	3/66	0
WCAP-3269-8	"Hydraulic Tests of the San Onofre Reactor Model"	Rev. 0	4.4	6/64	0
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	0
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	0

TABLE 1.6-2 (Sheet 2)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-3680-20	"Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors" (EURAEC-1974)	Rev. 0	4.3	3/68	0
WCAP-3680-21	"Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors" (EURAEC-2111)	Rev. 0	4.3	2/69	0
WCAP-3680-22	"Xenon-Induced Spatial Instabilities in Three-Dimensions" (EURAEC-2116)	Rev. 0	4.3	9/69	0
WCAP-3696-8	"Pressurized Water Reactor pH-Reactivity Effect Final Report" (EURAEC-2074)	Rev. 0	4.3	10/68	0
WCAP-3726-1	"Pu02 -U02 Fueled Critical Experiments"	Rev. 0	4.3	7/67	0
WCAP-6065	"Melting Point of Irradiated U02"	Rev. 0	4.4	2/65	0
WCAP-6069	"Burnup Physics of Heterogeneous Reactor Lattices"	Rev. 0	4.4	6/65	0

TABLE 1.6-2 (Sheet 3)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-6073	"LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS"	Rev. 0	4.3	4/66	0
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
WCAP-7015	"Subchannel Thermal Analysis of Rod Bundle Cores	Rev. 1	4.4	2/14/69	0
WCAP-7048 A(P) WCAP-7757-A	"The PANDA Code"	Rev. 0	4.3	1/9/75	A P-
WCAP-7198-L(P)	"Evaluation of Protective	Rev. 0	6.1	4/23/69	0
WCAP-7825	Coatings for use in Reactor Containment"			12/16/71	
WCAP-7213- P-A(P)	"The TURTLE 24.0 Diffusion Depletion Code"	Rev. 0	4.3, 15.0, 15.4	1/9/75	A
WCAP-7758-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

TABLE 1.6-2 (Sheet 4)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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 1/17/72	4.4	9/8/69	0
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)"	Supple- ment 1	3.10(N)	1/27/71 12/16/71	U
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

TABLE 1.6-2 (Sheet 5)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-7595-A	(See WCAP-7941-P-A(P))				
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
WCAP-7672	(See WCAP 7488-L(P))				
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

TABLE 1.6-2 (Sheet 6)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments – Final Development Report"	Supple-ment 1	6.2.5	5/23/72 5/31/72	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments - Equip-ment Qualification Report"	Supple-ment 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"	Supple-ment 3	6.2.5	1/23/74 3/22/74	A
WCAP-7709-L(P WCAP-7820	"Electric Hydrogen Recombiner for PWR Containments"	Supple-ment 4	6.2.5	4/21/74	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner Special Tests"	Supple-ment 5	6.2.5	1/7/76	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner IEEE 323-1974 Qualification"	Supple-ment 6	6.2.5	11/5/76	A
WCAP-7709-L(P) WCAP-7820	"Electric Hydrogen Recombiner LWR Containments - Supple-mental Test Number 2"	Supple-ment 7	6.2.5	9/21/77	A
WCAP-7735	(See WCAP-7477-L(P))				

TABLE 1.6-2 (Sheet 7)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Report Topical Report No.	Title	Revision Number	USAR Section Reference	Submitted to the NRC	Review(1) Status
WCAP-7750	"A Comprehensive Space Time Dependent Analysis of Loss-of-Coolant (SATAN-IV Digital (Code)"	Rev. 0	3.6.3	8/31/71	0
WCAP-7755-A	(See WCAP-7667-P-A(P))				
WCAP-7757-A	(See WCAP-7048-P-A(P))				
WCAP-7758-A	(See WCAP-7213-P-A(P))				
WCAP-7769	"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	0
WCAP-7800	"Nuclear Fuel Division Quality Assurance Program Plan"	Rev. 4A	3A 4.2	4/28/75	A
WCAP-7803	(See WCAP-7798-L(P))				
WCAP-7806	"Nuclear Design of Westing-house Pressurized Water Reactors with Burnable Poison Rods"	Rev. 0	4.3	21/16/71	B

TABLE 1.6-2 (Sheet 8)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-7810	(See WCAP-7308-L(P))				
WCAP-7811	"Power Distribution Control of Westinghouse Pressurized Water Reactors"	Rev. 0	4.3	12/16/71	0
WCAP-7817	(See WCAP-7397-L(P))	Rev. 0			
WCAP-7817	(See WCAP-7477-L(P))	Supple- ment 1			
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Low Seismic Plants)"	Supple- ment 2	3.10(N)	1/17/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Westing house Solid State Protection System) (Low Seismic Plants)"	Supple- ment 3	3.10(N)	1/17/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (WCID NUCANA 7300 Series) (Low Seismic Plants)"	Supple- ment 4	3.10(N)	12/14/72	U
WCAP-7817	"Seismic Testing of Electrical and Control Equipment (Instru- ment Bus Distribution Panel) (Low Seismic Plants)"	Supple- ment 5	3.10(N)	12/14/72	U
WCAP-7817	"Seismic Testing of Electrical and control Equipment (Type DB Reactor Trip Switchgear)	Supple- ment 6	3.10(N)	8/00/74	U

TABLE 1.6-2 (Sheet 9)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-7820	(See WCAP 7709-L[P])				
WCAP-7825	(See WCAP 7198-L[P])				
WCAP-7832	"Evaluation of Steam Generator Tube, Tube Sheet and Divider Plate Under Combined LOCA Plus SSE Conditions"	Rev. 0	5.4	21/26/73	A
WCAP-7836	"Inlet Orificing of Open PWR Cores"	Rev. 0	4.4	1/17/72	B
WCAP-7838	(See WCAP 7359-L[P])				
WCAP-7870	"Neutron Shielding Pads"	Rev. 0	3.9 (N)	7/17/72	A
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 U02 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 Multiploop PWR System"	Rev. 0		10/11/72	U

TABLE 1.6-2 (Sheet 10)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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 System (4-Loop Plant Using WCID-7300 Series Process Instrumentation)"	Rev. 0	7.2, 7.3	3/9/73	B
WCAP-7916	(See WCAP 2850-L[P])				
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
WCAP-7941-P-A (P) WCAP-7595-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-7958	(See WCAP-7695-P-A(P))				
WCAP-7964	"Axial Zenon Transient Tests at the Rochester Gas and Electric Reactor"	Rev. 0	4.3	6/15/71	O

TABLE 1.6-2 (Sheet 11)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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-7985	(See WCAP-7695, Addendum 1-P-A[P]) Addendum 1A				
WCAP-8028-A	(See WCAP-7979-P-A(P))				
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
WCAP-8172-A	(See WCAP 8082-P-A[P])				

TABLE 1.6-2 (Sheet 12)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-8183	"Operational Experience with Westinghouse Cores (up to December 31, 1977)"	Rev. 7	4.2	4/20/78	B
WCAP-8195	(See WCAP-8054(P))				
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	"Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations."	Rev. 0	3A, 4.2, 4.4	June 1985	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

TABLE 1.6-2 (Sheet 13)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-8252	"Documentation of Selected Westinghouse Structural Analysis Computer Codes"	Rev. 1	3.6.3, 3.9(N)	7/19/77	U
WCAP-8253	"Source Term Data for Westing-house 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-8261	(See WCAP-8200(P))				
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-8288	(See WCAP-8236(P))				
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 Fuel 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	0A

TABLE 1.6-2 (Sheet 14)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-8301 (P) WCAP-8305	"LOCTA-IV Program: Loss-of Coolant Transient Analysis"	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 Test"	Rev. 0	3.9(N)	7/18/75	A
WCAP-8305	(See WCAP-8301(P))				
WCAP-8306	(See WCAP-8302(P))				
WCAP-8317-A	(See WCAP-8303-P-A(P))				
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

TABLE 1.6-2 (Sheet 15)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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
WCAP-8370	"Westinghouse Water Reactor Divisions Quality Assurance Plan	Rev. 9A	3A		U
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

TABLE 1.6-2 (Sheet 16)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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
WCAP-8485	"Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Fall 1974"	Rev. 0		4/2/75	B

TABLE 1.6-2 (Sheet 17)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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	Rev. 0	5.3	12/00/75	U
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
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	11/00/83	U

TABLE 1.6-2 (Sheet 18)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
WCAP-8587	"Equipment Qualification Data Packages"	Rev. 1 Supple-ment 1	3.11(N) 3.10(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"	Rev. 0	3.10(N)	0/00/00	U
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	11/83	U
WCAP-8693	"Delta Ferrite in Production Austenitic Stainless Steel Weldments"	Rev. 0	5.2	3/16/76	B
WCAP-8708-P-A (P), Vol. I & II WCAP-8709-A, Volumes I & 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"	Rev. 0	4.2	11/2/76	A

TABLE 1.6-2 (Sheet 19)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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 THOT"	Rev. 0		5/6/77	A
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

TABLE 1.6-2 (Sheet 20)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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"	Rev. 0	15.6	4/77	U
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
WCAP-9166	"Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation With One Loop Out of Service for Plants Without Loop Isolation Values"	Rev. 0	15.6	2/00/78	U
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)	"Properties of Fuel and Core Component Materials"	Rev. 1	4.2	8/2/78	U

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WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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	Rev. 0	15.6	2/00/78	U
WCAP-9224, Ap- pendix A	"Hafnium"	Rev. 0	10/00/80		U
WCAP-9226(P) WCAP-9227	Reactor Core Response to Excessive Secondary Steam Releases		15.1	7/00/78	U
WCAP-9230 (P) WCAP-9231	"Report on the Consequences of a Postulated Main Feedline Rupture"	Rev. 0	15.2	1/27/78	U
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

TABLE 1.6-2 (Sheet 22)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	USAR Section Reference	Report Submitted to the NRC	Review(1) Status
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	Rev. 0	6.2.5	7/00/78	U
WCAP-9714-PA(P) WCAP-9750-A	Methodology for the Seismic Qualification of Westinghouse WRD Supplied Equip.		3.10.(N)	00/00/00	A
WCAP-9944(P) WCAP-9945	"Verification of Upper Head Injection Reactor Vessel Internals by Preoperational Tests on Sequoyah 1 Power Plant	Rev. 0	3.9(N)	7/00/81	U
WCAP-10297-P-A	Dropped Rod Methodology for Negative Flux Rate Trip Plants	Rev. 0	15.4	6/00/83	A
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-10858P-A & Addendum 1	"AMSAC Generic Design Package"	Rev. 1	7.7.1.11	07/25/85 02/26/87	A
WCAP-13589-A	"Assessment of Clad Flattening and Densificaiton Power Spike Factor Elimination in Westinghouse Nuclear Fuel"	Rev. 0	4.3	01/18/93	A
WCAP-16009-P-A	"Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)"	Rev. 0	15.0, 15.6	06/02/03	A

TABLE 1.6-2 (Sheet 23)

WESTINGHOUSE TOPICAL REPORTS INCORPORATED BY REFERENCE

(P) - Proprietary

(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 generations report with current validity; not actively under formal NRC review.

U - Actively under formal NRC review.

P - Pending approval by the NRC.

Table 1.6-3
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
1.1-1	1	Symbols and Legend for System Flow and Piping and Instrumentation Diagrams	M-120101
1.1-1	2	Symbols and Legend for System Flow and Piping and Instrumentation Diagrams	M-120102
1.1-1	3	Symbols and Legend for System Flow and Piping and Instrumentation Diagrams	M-020103
1.1-1	4	Symbols and Legend for System Flow and Piping and Instrumentation Diagrams	M-020104
1.2-1	0	Peninsular Plant Arrangement Standard Power Systems & Structure Interface	M-1G001
1.2-2	0	Equipment Location Radwaste Building Plan El. 1976'-0"	M-1G010
1.2-3	0	Equipment Location Radwaste Building Plan El. 2000'-0"	M-1G011
1.2-4	0	Equipment Location Radwaste Building Plan El. 2022'-0"	M-0G012
1.2-5	0	Equipment Location Radwaste Building El. 2031'-6" & Roof Plan	M-1G013
1.2-6	0	Equipment Location Radwaste Building Sections A & B	M-1G014
1.2-7	0	Equipment Location Radwaste Building Sections C & E	M-1G015
1.2-8	0	Equipment Location Radwaste Building Sections D & F	M-1G016
1.2-9	0	Equipment Location Reactor and Auxiliary Bldgs Plan - Basement El. 1974'-0"	M-1G020
1.2-10	0	Equipment Location Auxiliary Building Partial Plan El. 1988'-0" & El. 2013'-6"	M-1G021
1.2-11	0	Equipment Location Reactor and Auxiliary Building Plan Ground Floor Elevation 2000'-0"	M-1G022
1.2-12	0	Equipment Location Reactor and Auxiliary Building Plan El. 2026'-0"	M-1G023
1.2-13	0	Equipment Location Reactor and Auxiliary Buildings Plan Operating Floor El. 2047'-6"	M-1G024
1.2-14	0	Equipment Locations Reactor and Auxiliary Buildings Plan El. 2068'-8"	M-1G025
1.2-15	0	Equipment Location Reactor and Auxiliary Building Section A	M-1G026
1.2-16	0	Equipment Locations Reactor and Auxiliary Buildings Section B	M-1G027

Table 1.6-3 (Sheet 2)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
1.2-17	0	Equipment Location Reactor and Auxiliary Building Section C	M-1G028
1.2-18	0	Equipment Location Reactor and Auxiliary Building Section D	M-1G029
1.2-19	0	Equipment Location Auxiliary Building Sections E, F, & G	M-1G030
1.2-20	0	Equipment Location Fuel Building Plan Elevation 2000'- 0", 2026'-0" and 2047'-6"	M-1G040
1.2-21	0	Equipment Location Fuel Building Sections A, B, & C	M-1G041
1.2-22	0	Equipment Location Fuel Building Sections D, E, & F	M-1G042
1.2-23	0	Equipment Location Control Building & Communication Corridor Plan Elevation 1974'- 0" & 1984'-0"	M-1G050
1.2-24	0	Equipment Location Control & Diesel Generator Buildings & Communication Corridor Plan Elevation 2000'-0" & 2016'-0"	M-1G051
1.2-25	0	Equipment Location Control & Diesel Generator Buildings & Communication Corridor Plan Elevation 2032'-0" & 2047'-6"	M-1G052
1.2-26	0	Equipment Location Control & Diesel Generator Buildings & Corridor Plan Elevation 2061'- 6", 2066'-0" & 2073'-6" & Section D.	M-1G053
1.2-27	0	Equipment Location Control & Diesel Generator Buildings & Communication Corridor Section A	M-1G054
1.2-28	0	Equipment Location Control & Diesel Generator Buildings Sections B & C	M-1G055
1.2-29	0	Equipment Location Turbine Building Condenser Pit Plan Elevation 1983'-0"	M-1G060
1.2-30	0	Equipment Location Turbine Building Ground Floor Plan Elevation 2000'-0"	M-1G061
1.2-31	0	Equipment Location Turbine Building Partial Plan Elevation 2015'-4"	M-1G062
1.2-32	0	Equipment Location Turbine Building Mezzanine Floor Plan Elevation 2033'-0"	M-1G063
1.2-33	0	Equipment Location Turbine Building Operating Floor Plan Elevation 2065'-0"	M-1G064

Table 1.6-3 (Sheet 3)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
1.2-34	0	Equipment Location Turbine Building Section A	M-1G065
1.2-35	0	Equipment Location Turbine Building Section B	M-1G066
1.2-36	0	Equipment Location Turbine Building Section C	M-1G067
1.2-37	0	Equipment Location Turbine Building Section D	M-1G068
1.2-38	0	Equipment Location Turbine Building Section E	M-1G069
1.2-39	0	Equipment Location Turbine Building Section F	M-1G070
1.2-40	0	Equipment Location Turbine Building Section G	M-0G071
1.2-41	0	Equipment Location Turbine Building Section H	M-1G072
1.2-42	0	Turbine Component Laydown Area, Elevation 2065'-0"	M-1G073
1.2-44	0	Site Plan	8025-C-KG1202
2.4-3	2	Grading Plan Switchyard Area	S-0172
2.4-3	3	Drainage Plan Plant Area	S-0186
2.4-3	4A	Manhole, Pipe & Culvert Schedule	S-0189 Sheet 1
2.4-3	4B	Manhole, Pipe & Culvert Schedule	S-0189 Sheet 2
2.4-3	4C	Manhole, Pipe & Culvert Schedule	S-0189 Sheet 3
2.4-3	4D	Manhole, Pipe & Culvert Schedule	S-0189 Sheet 4
2.4-3	5	Manhole & Pipe Details	S-0191
2.4-3	6A	Manhole & Pipe Details	S-0296 Sheet 1
2.4-3	6B	Manhole & Pipe Details	S-0296 Sheet 2
2.4-3	7	Plant Area Roadway Grading & Drainage	S-0297
5.1-1	1	Reactor Coolant System	M-12BB01
5.1-1	2	Reactor Coolant System	M-12BB02
5.1-1	3	Reactor Coolant System	M-12BB03
5.1-1	4	Reactor Coolant System	M-12BB04
5.4-7	0	Residual Heat Removal System	M-12EJ01
5.4-21	0	Hot and Cold Leg Lateral Restraints	C-03BB53
6.2.2-1	0	Containment Spray System	M-12EN01
6.2.2-2	1	Containment Spray System Reactor Building A & B Trains	M-13EN03
6.2.2-2	2	Containment Spray System Reactor Building A & B Trains	M-13EN04
6.2.2-2	3	Containment Spray System Reactor Building A & B Trains	M-13EN05
6.2.5-1	0	Containment Hydrogen Control System	M-12GS01
6.2.6-1	0	Containment Integrated Leak Rate Test	M-12GP01
6.3-1	1	Borated Refueling Water Storage System	M-12BN01
6.3-1	2	High Pressure Coolant Injection System	M-12EM01
6.3-1	3	High Pressure Coolant Injection System	M-12EM02
6.3-1	4	Accumulator Safety Injection	M-12EP01

Table 1.6-3 (Sheet 4)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
7.2-1	1	Functional Diagrams (Index and Symbols)	M-744-00018
7.2-1	2	Functional Diagrams (Reactor Trip Signals)	M-744-00019
7.2-1	3	Functional Diagrams (Nuclear Instrumentation and Manual Trip Signals)	M-744-00020
7.2-1	4	Functional Diagrams (Nuclear Instrumentation Permissives and Blocks)	M-744-00021
7.2-1	5	Functional Diagrams (Primary Coolant System Trip Signals	M-744-00022
7.2-1	6	Functional Diagrams (Pressurizer Trip Signals)	M-744-00023
7.2-1	7	Functional Diagrams (Steam Generator Trip Signals)	M-744-00024
7.2-1	8	Functional Diagrams (Safeguards Actuation Signals)	M-744-00025
7.2-1	9	Functional Diagrams (Rod Controls and Rod Blocks)	M-744-00026
7.2-1	10	Functional Diagrams (Steam Dump Control)	M-744-00027
7.2-1	11	Functional Diagrams (Pressurizer Pressure and Level Control)	M-744-00028
7.2-1	12	Functional Diagrams (Pressurizer Heater Control)	M-744-00029
7.2-1	13	Functional Diagrams (Feedwater Control and Isolation)	M-744-00030
7.2-1	14	Functional Diagrams (Feedwater Control and Isolation)	M-744-00031
7.2-1	15	Functional Diagrams (Auxiliary Feedwater Pumps Start-up)	M-744-00032
7.2-1	16	Functional Diagrams (Turbine Trips, Runbacks and Other Signals)	M-744-00033
7.2-1	17	Functional Diagram (Pressurizer Pressure Relief System Train A)	M-744-00039
7.2-1	18	Functional Diagram (Pressurizer Pressure Relief System Train B)	M-744-00040
7.3-1	2	Logic Diagram Engineered Safety Features Actuation System (BOP)	J-104-00390
7.6-4	1	Train B Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System	M-744-00039
7.6-4	2	Train A Functional Diagram Showing Logic Requirements for Pressurizer Pressure Relief System	M-744-00040
8.2-3	0	Wolf Creek Substation General Plan	KD-7750
8.2-4	0	One-Line Diagram	KD-7496

Table 1.6-3 (Sheet 5)
USAR Figure/Controlled Drawing Cross-Reference

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8.3-1	1	Main Single Line Diagram	E-11001
8.3-1	2	Single Line Diagram, Essential Service Water System	E-K1001
8.3-1	3	Single Line Diagram Site Area Loads	E-1001
8.3-2	0	List of Loads Supplied by the Emergency Diesel Generator	E-11005
8.3-3	0	Logic Diagram Standby Generation Excitation Control	E-12NE01
8.3-4	0	Logic Diagram Standby Generator System Protection	E-12NE02
8.3-5	0	Logic Diagram Standby Generator Engine and Governor Control	E-12KJ01
8.3-6	1	DC Main Single Line Diagram	E-11010
8.3-7	0	DC Main Single Line Diagram (PK03 and PK04 Bus)	E-11010A
9.1-3	1	Fuel Pool Cooling and Cleanup System	M-12EC01
9.1-3	2	Fuel Pool Cooling and Cleanup System	M-12EC02
9.2-1	1	Service Water System	M-12EA01
9.2-1	2	Service Water System	M-12EA02
9.2-1	3	Service Water System	M-0022 Sheet 1
9.2-2	1	Essential Service Water System	M-12EF01
9.2-2	2	Essential Service Water System	M-12EF02
9.2-2	3	Essential Service Water System	M-K2EF01
9.2-2	4	Essential Service Water System	M-K2EF03
9.2-3	0	ESW Pumphouse Equipment Location - Plan	M-KG080
9.2-4	0	ESWS Pumphouse Equipment Location - Sections	M-KG081
9.2-5	1	Makeup Demineralizer System	M-0025 Sheet 1
9.2-5	2	Makeup Demineralizer System	M-0025 Sheet 2
9.2-5	3	Makeup Demineralizer System	M-0025 Sheet 3
9.2-5	4	Makeup Demineralizer System	M-0025 Sheet 4
9.2-5	4A	Makeup Demineralizer System	M-0025 Sheet 4A
9.2-5a	0	Potable Water System	A-0503 Sheet 1
9.2-13	0	Reactor Make-up Water System	M-12BL01
9.2-14	0	Closed Cooling Water System	M-12EB01
9.2-15	1	Component Cooling Water System	M-12EG01
9.2-15	2	Component Cooling Water System	M-12EG02
9.2-15	3	Component Cooling Water System	M-12EG03
9.2-16	0	Demineralized Water Storage and Transfer System	M-12AN01
9.2-17	1	Domestic Water System	M-12KD01
9.2-17	2	Domestic Water System	M-12KD02
9.2-23	0	Condensate Storage and Transfer System	M-12AP01

Table 1.6-3 (Sheet 6)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
9.2-24	1	Waste Water Treatment Facility	M-12WT01
9.2-25	1	Waste Water Treatment Facility	M-12WT03
9.3-1	1	Compressed Air System	M-12KA01
9.3-1	2	Compressed Air System (Service Air)	M-12KA02
9.3-1	3	Instrument Air System	M-12KA03
9.3-1	4	Instrument Air System	M-12KA04
9.3-1	5	Compressed Air System	M-12KA05
9.3-1	6	Compressed Air System	M-12KA06
9.3-1	7	Compressed Air System	M-12KA07
9.3-2	1	Nuclear Sampling System	M-12SJ01
9.3-2	2	Nuclear Sampling System	M-12SJ03
9.3-3	0	Nuclear Sampling System	M-12SJ02
9.3-4	1	Process Sampling System	M-12RM01
9.3-4	2	Process Sampling System	M-12RM02
9.3-4	3	Process Sampling System	M-12RM03
9.3-5	1	Sanitary Lift Station & Turb. Bldg. Sanitary Drainage System	M-12LA01
9.3-5	2	Comm. Corridor & Control Bldg. Sanitary Drainage System	M-12LA02
9.3-5	3	Chemical and Detergent Waste	M-12LD01
9.3-5	4	Turbine Bldg. and Aux. Feedwater Pump Rooms Oily Waste System	M-12LE01
9.3-5	5	Control and Diesel Generator Bldg. Oily Waste System	M-12LE02
9.3-5	6	Turbine Bldg. and Aux. Boiler Room Oily Waste System	M-12LE03
9.3-5	7	Tendon Access Gallery and Turbine Bldg. Oily Waste System	M-12LE04
9.3-5	8	Auxiliary Building Floor and Equipment Drain (FED) System	M-12LF01
9.3-5	9	Auxiliary Building Floor and Equipment Drain System	M-12LF02
9.3-5	10	Auxiliary Building Floor and Equipment Drain System	M-12LF03
9.3-5	11	Auxiliary Building Floor and Equipment Drain System	M-12LF04
9.3-5	12	Auxiliary Building Floor and Equipment Drain System	M-12LF05
9.3-5	13	Radwaste and Fuel Bldgs. FED System	M-12LF06
9.3-5	14	Radwaste Bldg. FED System	M-12LF07
9.3-5	15	Control and Fuel Bldgs. FED System	M-12LF08
9.3-5	16	Reactor Bldg. and Hot Machine Shop FED System	M-12LF09
9.3-5	17	Radwaste Bldg. and Tunnel FED System	M-12LF10

Table 1.6-3 (Sheet 7)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
9.3-7	1	Reactor Building, Stainless Steel Liner Plate, Reactor Refueling Canal	C-0L2931
9.3-7	2	Fuel Building-Area 1, Stainless Steel Liner Plate Plan, Spent Fuel Pool	C-1L6111
9.3-8	1	Chemical and Volume Control System	M-12BG01
9.3-8	2	Chemical and Volume Control System	M-12BG02
9.3-8	3	Chemical and Volume Control System	M-12BG03
9.3-8	4	Chemical and Volume Control System	M-12BG04
9.3-8	5	Chemical and Volume Control System	M-12BG05
9.3-9	1	Service Gas System	M-12KH01
9.3-9	2	Service Gas System	M-12KH02
9.3-11	1	Boron Recycle System	M-12HE01
9.3-11	2	Boron Recycle System	M-12HE02
9.3-11	3	Boron Recycle System	M-12HE03
9.4-1	1	Control Building HVAC	M-12GK01
9.4-1	2	Control Building HVAC	M-12GK02
9.4-1	3	Control Building HVAC	M-12GK03
9.4-1	4	Control Building HVAC	M-12GK04
9.4-2	1	Fuel Building HVAC	M-12GG01
9.4-2	2	Fuel Building HVAC	M-12GG02
9.4-3	1	Miscellaneous Buildings HVAC	M-12GF01
9.4-3	2	Miscellaneous Buildings HVAC	M-12GF02
9.4-3	3	Auxiliary Building HVAC	M-12GL03
9.4-3	4	Auxiliary Building HVAC	M-12GL02
9.4-3	5	Auxiliary Building HVAC	M-12GL01
9.4-4	1	Turbine Building HVAC	M-12GE01
9.4-4	2	Turbine Building HVAC	M-12GE02
9.4-4	3	Turbine Building HVAC	M-12GE03
9.4-4	4	Turbine Building HVAC	M-12GE04
9.4-5	1	Radwaste Building HVAC	M-12GH01
9.4-5	2	Radwaste Building HVAC	M-12GH02
9.4-6	1	Containment Cooling System	M-12GN01
9.4-6	2	Containment Cooling System	M-12GN02
9.4-6	3	Containment Atmospheric Control System	M-12GR01
9.4-6	4	Containment Purge Systems HVAC	M-12GT01
9.4-7	0	Diesel Generators Building HVAC	M-12GM01
9.4-8	0	Essential Service Water Pump House HVAC	M-K2GD01
9.4-9	1	Plant Heating System	M-12GA01
9.4-9	2	Plant Heating System	M-12GA02
9.4-10	0	Central Chilled Water System	M-12GB01
9.4-11	0	Waste Water Treatment Facility HVAC	M-12VW01

Table 1.6-3 (Sheet 8)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
9.5-1	1	Fire Protection System (site)	M-0023 Sheet 1
9.5-1	2	Fire Protection System (site)	M-0023 Sheet 2
9.5-1	3	Fire Protection System (site)	M-0023 Sheet 3
9.5-1	4	Fire Protection System (site)	M-0023 Sheet 4
9.5-2	0	Outdoor Piping, Key Plan and General Notes	M-0051
9.5.1-1	1	Fire Protection Turbine Building	M-12KC01
9.5.1-1	2	Fire Protection System (power block)	M-12KC02
9.5.1-1	3	Fire Protection System (power block)	M-12KC03
9.5.1-1	4	Fire Protection (Halon) System	M-12KC04
9.5.1-1	5	Fire Protection System (power block)	M-12KC05
9.5.1-1	6	Fire Protection (Halon) System	M-12KC06
9.5.1-1	7	Fire Protection (Halon) System	M-12KC07
9.5.1-2	1	Fire Area Delineation el. 1974’	10466-A-1801
9.5.1-2	2	Fire Area Delineation el. 2000’	10466-A-1802
9.5.1-2	3	Fire Area Delineation el. 2026’	10466-A-1803
9.5.1-2	4	Fire Area Delineation el. 2047’-6”	10466-A-1804
9.5.2-1	0	Telephone System Riser Diagram	E-14QE01
9.5.2-2	0	Public Address System Riser Diagram	E-1L9903
9.5.3-1	0	Lighting Distribution Riser Diagram	E-1L9901
9.5.4-1	0	Emergency Fuel Oil System	M-12JE01
9.5.5-1	1	Standby Diesel Generator "A" Cooling Water System	M-12KJ01
9.5.5-1	2	Standby Diesel Generator "B" Cooling Water System	M-12KJ04
9.5.6-1	1	Standby Diesel Generator "A" Intake, Exh., F.0. and Starting Air System	M-12KJ02
9.5.6-1	2	Standby Diesel Generator "B" Intake, Exh., F.0. and Starting Air System	M-12KJ05
9.5.7-1	1	Standby Diesel Generator "A" Lube Oil System	M-12KJ03
9.5.7-1	2	Standby Diesel Generator "B" Lube Oil System	M-12KJ06
9.5.9-1	1	Auxiliary Boiler System	M-12FA01
9.5.9-1	2	Auxiliary Steam System	M-12FB01
9.5.9-1	3	Auxiliary Steam System	M-12FB02
9.5.9-1	4	Auxiliary Steam Chemical Addition System	M-12FE01
9.5.10-1	1	Breathing Air System	M-12KB01
9.5.10-1	2	Breathing Air System	M-12KB02
9.5.10-1	3	Breathing Air System	M-12KB03

Table 1.6-3 (Sheet 9)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
10.2-1	1	Main Turbine	M-12AC01
10.2-1	2	Main Turbine	M-12AC02
10.2-1	3	Main Turbine	M-12AC03
10.2-1	4	Main Turbine	M-12AC04
10.2-1	5	Lube Oil Storage, Transfer and Purification System	M-12CF01
10.2-1	6	Lube Oil Storage, Transfer and Purification System	M-12CF02
10.2-1	7	Main Turbine Control Oil System	M-12CH01
10.2-1	8	Main Turbine Control Oil System	M-12CH02
10.3-1	1	Main Steam System	M-12AB01
10.3-1	2	Main Steam System	M-12AB02
10.3-1	3	Main Steam System	M-12AB03
10.4-1	1	Circulating Water & Waterbox Drains System	M-12DA01
10.4-1	2	Circulating Water System	M-0021
10.4-1	3	Circulating Water Waterbox Venting System	M-12DA02
10.4-1	4	Circulating Water Screenhouse Plans	M-0004
10.4-1	5	Circulating Water Screenhouse – Sections	M-0005
10.4-2	1	Condensate System	M-12AD01
10.4-2	2	Condensate System	M-12AD02
10.4-2	3	Condensate System	M-12AD03
10.4-2	4	Condensate System	M-12AD04
10.4-2	5	Condensate System	M-12AD05
10.4-2	6	Condensate System	M-12AD06
10.4-3	0	Condenser Air Removal	M-12CG01
10.4-4	0	Steam Seal System	M-12CA01
10.4-5	1	Condensate Demineralizer System	M-12AK01
10.4-5	2	Condensate Demineralizer System	M-12AK02
10.4-5	3	Condensate Demineralizer System	M-12AK03
10.4-6	1	Feedwater System	M-12AE01
10.4-6	2	Feedwater System	M-12AE02
10.4-6	3	Feedwater Heater Extraction Drains & Vents	M-12AF01
10.4-6	4	Feedwater Heater Extraction Drains & Vents	M-12AF02
10.4-6	5	Feedwater Heater Extraction Drains & Vents	M-12AF03
10.4-6	6	Feedwater Heater Extraction Drains & Vents	M-12AF04
10.4-6	7	Auxiliary Turbines S.G.F.P. Turbine “A”	M-12FC03
10.4-6	8	Auxiliary Turbines S.G.F.P. Turbine "B"	M-12FC04
10.4-7	1	Condensate Chemical Addition System	M-12AQ01
10.4-7	2	Feedwater Chemical Addition System	M-12AQ02
10.4-8	1	Steam Generator Blowdown System	M-12BM01
10.4-8	2	Steam Generator Blowdown System	M-12BM02
10.4-8	3	Steam Generator Blowdown System	M-12BM03
10.4-8	4	Steam Generator Blowdown System	M-12BM04
10.4-8	5	Steam Generator Blowdown System	M-12BM05

Table 1.6-3 (Sheet 10)
USAR Figure/Controlled Drawing Cross-Reference

Figure #	Sheet	Title	Drawing #
10.4-9	0	Auxiliary Feedwater System	M-12AL01
10.4-10	0	Auxiliary Turbines Auxiliary Feedwater Pump Turbine	M-12FC02
10.4-12	1	Secondary Liquid Waste System	M-12HF01
10.4-12	2	Secondary Liquid Waste System	M-12HF02
10.4-12	3	Secondary Liquid Waste System	M-12HF03
10.4-12	4	Secondary Liquid Waste System	M-12HF04
11.2-1	1	Liquid Radwaste System	M-12HB01
11.2-1	2	Liquid Radwaste System	M-12HB02
11.2-1	3	Liquid Radwaste System	M-12HB03
11.2-1	4	Liquid Radwaste System	M-12HB04
11.3-1	1	Gaseous Radwaste System	M-12HA01
11.3-1	2	Gaseous Radwaste System	M-12HA02
11.3-1	3	Gaseous Radwaste System	M-12HA03
11.4-1	1	Solid Radwaste System	M-12HC01
11.4-1	2	Solid Radwaste System	M-12HC02
11.4-1	3	Solid Radwaste System	M-12HC03
11.4-1	4	Solid Radwaste System	M-12HC04
12.3-2	1	Radiation Zones for Normal Operation El. 1974’	10466-A-1701
12.3-2	2	Radiation Zones for Normal Operation El. 2000’	10466-A-1702
12.3-2	3	Radiation Zones for Normal Operation El. 2026’	10466-A-1703
12.3-2	4	Radiation Zones for Normal Operation El. 2047’-6”	10466-A-1704
12.3-2	5	Radiation Zones for Normal Operation Turbine Bldg El. 1983’ & 2000’	10466-A-1705
12.3-2	6	Radiation Zones for Normal Operation Turbine Bldg El. 2033’ & 2065’	10466-A-1706
12.3-4	0	Decontamination System	M-12HD01
18.2-15	0	Nuclear Sampling System	M-12SJ04

Table 1.6-4 (sheet 1)
Incorporated by Reference
USAR Section/Controlled Document Cross-Reference

Section	Title	Document #
Table 3.11(B)-1	Plant Environmental Normal Conditions	EQSD-I, Attachment A and B
Table 3.11(B)-2	Environmental Qualification Parameters for SNUPPS NUREG-0588 (LOCA, MSLB and HELB)	EQSD-I, Attachment A and B
Table 3.11(B)-3	Identification of Safety-Related Equipment and Components: Equipment Qualification	EQSD-I, Attachment A and B; EQSD-II, Tables 1 and 2
Table 3.11(B)-4	Containment Worst Case Radiation Levels (MRADs)	EQSD-I, Attachment A
Table3.11(B)-5	Containment Spray Requirements	EQSD-I, Attachment A
Table 3.11(B)-8	Exemptions from NUREG-0588 Qualification	EQSD-I, Attachment C
Table 3.11(B)-10	Equipment Added for NUREG-0737	EQSD-II, Tables 1 and 2
Figures 3.11(B)-1 through 3.11(B)-49	Figures	EQSD-I, Attachment A
Section 9.5.1.2.2.3*	Fire Barriers *Note: Only portions of this section have been relocated and Incorporated by Reference	M-663-00017A
Appendix 9.5B.1	Fire Hazard Analyses – Introduction	E-1F9905
Table 9.5B-1	Minimum Equipment Required for Safe Shutdown	XX-E-013
Appendix 9.5B.2	Fire Hazard Analyses – Assumption on Plant Conditions	E-1F9905
Table 9.5B-2	Equipment Required for Shutdown following a Fire	XX-E-013
Appendix 9.5B.3	Fire Hazard Analyses – Fire Effects on Electrical Equipment and Safe Shutdown Information	E-1F9905

Table 1.6-4 (sheet 2)
Incorporated by Reference

USAR Section/Controlled Document Cross-Reference

Section	Title	Document #
Table 9.5B-3	Safety-Related Fire Areas Containing Rooms Without Detection Provisions	E-1F9905, Attachment A
Appendix 9.5B.4	Fire Hazard Analyses – General Information on Design Features	E-1F9905, or XX-E-013, or E-1F9900
Table 9.5B-4	Safety-Related Fire Areas Outside Containment With Area Suppression Coverage	E-1F9905 Attachment A
Appendix 9.5B.5	Fire Hazard Analyses – Combustible Loadings and Flame Spread	E-1F9905
Table 9.5B-5	Non-Safety Related Site Structures	E-1F9905, Attachment C
Appendix 9.5B.6	Fire Hazard Analyses – Fire Hazard Review Methodology	E-1F9905, or XX-E-013
Appendix 9.5B.7	Fire Hazard Analyses – Power Block Fire Hazards Analysis	E-1F9905
Appendix 9.5B.8	Fire Hazard Analyses – Site Specific Fire Hazards Analysis	E-1F9905
Appendix 9.5B	Design Basis Document for OFN AP-017, Control Room Evacuation	E-1F9915
Chapter 16.0	Technical Requirements Manual	Technical Requirements Manual
Chapter 17.2	Quality Assurance During the Operation Phase	Quality Program Manual

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 USAR text and tables. The controlled drawings that were removed from the USAR in Revision 17 and incorporated by reference are listed in Table 1.6-3

1.7.1 Electrical, Instrumentation and Control Drawings

Table 1.7-1 contains a list of electrical, instrumentation, and control drawings that are considered to be necessary to evaluate the safety-related features pertaining to the power block.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of each piping and instrumentation diagram and the corresponding USAR 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 symbols and characters used in these USAR figures.

1.7.3 Miscellaneous Controlled Drawings

Table 1.7-3 contains a list of other controlled drawings utilized as figures in the USAR and the corresponding USAR figure number as it appears at the end of the respective text section.

TABLE 1.7-1

ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

DRAWING NUMBER	TITLE
E-K1001	SINGLE LINE DIAGRAM ESSENTIAL SERVICE WATER SYSTEM
E-11NB01	LOWER MED VOLTAGE SYS CLASS 1E 4.16 KV SINGLE LN
E-11NB02	LOWER MED VOLTAGE SYS CLASS 1E 4.16 KV SINGLE LN
E-11NE01	STBY GENERATOR CLASS 1E 4.16 KV S/L
E-11NG01	CLASS 1E LOW VOLTAGE 480V SYS S/L M & R
E-K1NG01	LOW VOLTAGE SYSTEM CLASS 1E 480V SINGLE LINE METER & RELAY DIAGRAM
E-11NG02	CLASS 1E LOW VOLTAGE 480V SYS S/L M & R
E-11NG20	LOW VOLTAGE SYSTEM CLASS 1E MOTOR CONTROL CENTERS SUMMARY
E-11NK01	CLASS 1E 125 DC SYS S/L M & R
E-11NK02	CLASS 1E 125 DC SYS S/L M & R
E-11PA01	HIGH MED VOLT SYS 13.8 KV M & R
E-11PA02	HIGH MED VOLT SYS 13.8 KV M & K
E-11PG06	NON-CLASS 1E LOW VOLTAGE SYS 480V S/L M & R
E-11PN01	NON-CLASS 1E INSTRUMENT AC POWER
E-12KJ01	STANDBY GENERATOR SYSTEM
E-K2NG01	LOGIC DIAGRAM - 4.16 KV MOTOR CONTROL CENTER TRANSFORMER FEEDER BREAKERS (VOID)
E-K2NG02	LOGIC DIAGRAM - 480V MOTOR CONTROL CENTER TRANSFORMER FEEDER BREAKERS (VOID)
E-K2NG03	LOGIC DIAGRAM - 480V SYSTEM NOTES & REFERENCES
E-13AB01	MAINSTEAM SUPPLY VLV TO TURB DR AUX FEEDW PUMP
E-13AB03A	MAIN STEAM LINE DRAIN VLV
E-13AB03B	MAIN STEAM LINE DRAIN VLV
E-13AB06A	MAIN STM ATMOS VENT VLV POS INT LIGHT

TABLE 1.7-1 (SHEET 2)

DRAWING NUMBER	TITLE
E-13AB06B	MAIN STM ATMOS VENT VLV POS INT LIGHT
E-13AB08	MN STM COOLDOWN VLV
E-13AB11A	SCHEMAT DIAG MAIN DUMP VLVS
E-13AB11B	SCHEMAT DIAG MAIN STM DUMP VLV
E-13AB11C	SCHEMAT DIAG MAIN STM DUMP VLV
E-13AB17	MN STM BYPASS VAL TO AUX FEEDWTR PUMP TURB
E-13AB23A	SCHEMAT DIAG MAIN STM BYPASS VLVS
E-13AB23B	SCHEMAT DIAG MAIN STM BYPASS VLVS
E-13AB26	SCHEMATIC DIAGRAM MAIN STEAM ISO VALVES ALL CLOSE – SEPARATION GROUP 1
E-13AB27	SCHEMATIC DIAGRAM MAIN STEAM ISO VALVES ALL CLOSE – SEPARATION GROUP 4
E-13AB28	SCHEMATIC DIAGRAM MAIN STEAM ISO VALVES CONTROL PART 1
E-13AB29	SCHEMATIC DIAGRAM MAIN STEAM ISO VALVES CONTROL PART 2
E-13AB30	SCHEMATIC DIAGRAM MAIN STEAM AND FEEDWATER ISO VLV MISCELLANEOUS CIRCUITS
E-13AB31	STEAM DUMP CONTROL & BYPASS INDICATION
E-13AC38	MAIN TURB SYS WITH NSSS INTERFACE
E-13AE05	SCH DIAG STEAM GENER CHEMICAL INJECT
E-13AE06	SCH DIAG MAIN FEEDWTR CONTR VALVES
E-13AE07	SCH DIAG MAIN FEEDWTR CONTR VALVES
E-13AE14	SCHEMATIC DIAGRAM FEEDWATER ISOLATION VALVES ALL CLOSE – SEPARATION GROUP 1
E-13AE15	SCHEMATIC DIAGRAM FEEDWATER ISOLATION VALVES ALL CLOSE – SEPARATION GROUP 4
E-13AE16	SCHEMATIC DIAGRAM FEEDWATER ISO VALVES CONTROL PART 1
E-13AE17	SCHEMATIC DIAGRAM FEEDWATER ISO VALVES CONTROL PART 2
E-13AE18	BYPASS FEEDWATER CONTROL VALVES
E-13AL00	AUX FEEDWATER SCHEMATIC INDEX SHEET
E-13AL01A	AUX FEEDWATER SYSTEM MOTOR DRIVEN PUMP
E-13AL01B	AUX FEEDWATER SYSTEM MOTOR DRIVEN PUMP
E-13AL02A	AUX FEEDWATER SYSTEM MOTOR OPERATED VALVES

TABLE 1.7-1 (SHEET 3)

DRAWING NUMBER	TITLE
E-13AL02B	AUX FEEDWATER SYSTEM MOTOR OPERATED VALVES
E-13AL02C	SCHEMATIC DIAGRAM MOTOR OPERATED VALVE ALHV0036
E-13AL03A	AUX FEEDWATER PUMPS DISCH CONTR MOVS
E-13AL03B	AUX FEEDWATER PUMPS DISCH CONTROL MOVS
E-13AL04A	SUPPLY FROM ESS SERV WTR SYS
E-13AL04B	SUPPLY FROM ESS SERV WTR SYS
E-13AL05A	AUX FDWR PUMPS DISCH CONTR AIR OP VALVES
E-13AL05B	AUX FDWR PUMPS DISCH CONTR AIR OP VALVES
E-13AL06	SCHEMATIC DIAGRAM INSTRUMENTATION & ALARMS
E-13AL07A	SCHEMATIC DIAGRAM INSTRUMENTATION & ALARMS
E-13AL07B	SCHEMATIC DIAGRAM INSTRUMENTATION & ALARMS
E-13AL08	SCHEMATIC DIAGRAM INSTRUMENTATION & ALARMS
E-13AL09	SCHEMATIC DIAGRAM MISCELLANEOUS CIRCUITS
E-13AL10	SCHEMATIC DIAGRAM AUX FDWTR SYS
E-13AP04	SCHEMATIC DIAGRAM CONDENSATE SYSTEM
E-13BB03	S.D. RCP THERM BARRIER CCW ISO VLVS
E-13BB04	S.D. SEAL WTR INJECT ISO VALVE
E-13BB11	S.D. PRZR RELIEF TANK VENT TO WPS ISO VLV
E-13BB12A	S.D. RHR LOOP INLET ISO VALVE
E-13BB12B	S.D. RHR LOOP INLET ISO VALVE
E-13BB13	S.D. PRZR RELIEF TK VENT TO WPRS ISO VLV
E-13BB27	SCH DIAG REACTOR COOLANT PUMP MOTORS
E-13BB28	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB30	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB31	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB33	SCHEMATIC DIAGRAM
E-13BB34	SCHEMATIC DIAGRAM
E-13BB35	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB36	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP

TABLE 1.7-1 (SHEET 4)

DRAWING NUMBER	TITLE
E-13BB37	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB38	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB39	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BB40	SCHEMATIC DIAGRAM REACTOR COOLANT PUMP
E-13BG01	CENTRIFUGAL CHARGING PUMPS SCHEMATIC DIAGRAM
E-13BG11A	CHARGING PUMP TO REACTOR COOLANT & MINIFLOW ISO
E-13BG11B	CHARGING PUMP TO REACTOR COOLANT & MINIFLOW ISO
E-13BG11C	SCHEMATIC DIAGRAM CHARGING PUMP TO REACTOR COOLANT AND MINIFLOW ISOLATION VALVE BGHV8111
E-13BG12	VOLUME CONT TANK OUTLET ISO VLVS SCH DIAG
E-13BG12A	VOLUME CONTROL TANK OUTLET ISO VALVE
E-13BG13	BORIC ACID FILTER TO CHG PUMP VLV
E-13BG17	LETDOWN LINE ISO VLV SCH DIAG
E-13BG24	REACTOR COOLANT PUMP SEAL WATER ISO VLV SCH DIAG
E-13BG27	BORIC ACID TRANSFER PUMPS
E-13BG36	LETDOWN CONTAINMENT ISO VLV SCH DIAG
E-13BG38	REACTOR COOLANT PUMP SEAL WATER ISO VLV SCH DIAG
E-13BG48	EXCESS LETDOWN LINE ISO VLV SCH DIAG
E-13BG52	RCP SEAL INJECTION FLOW THROTTLING VALVES
E-13BL01	PRZR RELIEF TANK REACTOR TO MAKEUP WTR SUPPLY
E-13BM01	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BM02	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BM03	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BM06A	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BM06B	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BM06C	STEAM GENERATOR BLOWDOWN SCHEMATIC
E-13BN01	REFUELG WTR STRG TK TO CHARGE PUMP MOV
E-13BN01A	SCHEMATIC DIAGRAM REFUELING WATER STORAGE TANK TO CHARGING PUMP MOV BNLCV0112E

TABLE 1.7-1 (SHEET 5)

DRAWING NUMBER	TITLE
E-13BN02	REFUELG WTR STRG TK TO SFPCS PUMP MOV
E-13BN03	REFUELG WTR STRG TK RHR PUMP MOV
E-13BN04	RWST TO CONTMET SPRAY PUMP MOV
E-13BN06	RWST TO SAFETY INJECT PUMP VLV
E-13BN08	SIS PUMP MINIFLW ISO VLV
E-13BN10	SCHEMATIC DIAGRAM
E-13EC01	FUEL POOL COOLING PUMPS SCHEMATIC DIAGRAM
E-13EC02	SCHEM DIAG CCW DISCHARGE VLVS FROM FUEL POOL CLG
E-13EF01	ESW TO AIR COMP ISOL VALVES SCHEMATIC DIAGRAM
E-K3EF01	SCHEMATIC DIAGRAM – ESW PUMPS
E-13EF02	SCHEMATIC DIAG ESW TO SW SYS ISOL VLVS
E-K3EF02	SCHEMATIC DIAGRAM – TRAVELING WATER SCREENS
E-13EF03	SCHEMATIC DIAG ESW TO SW SYS ISOL VLVS
E-K3EF03	SCHEMATIC DIAGRAM – SCREEN WASH WATER VALVE
E-13EF04	SCHEM DIAG ESW FROM COMPONENT COOLG WTR HEAT EXC
E-K3EF04	SCHEMATIC DIAGRAM – ESW SELF-CLEANING STRAINER
E-13EF05	SCHEMATIC DIAGRAM ESW TO COMPONENT COOLG WTR HEA
E-13EF05A	SCHEMATIC DIAGRAM ESW TO COMPONENT COOLING WATER HEAT EXCHANGER ISOLATION VALVE EFHV0052
E-K3EF05	SCHEMATIC DIAGRAM - SELF-CLEANING STRAINER TRASH VALVE
E-13EF06	SCHEMATIC DIAG ESW TO ULTIM HEAT SINK ISOL VALVE
E-K3EF06	SCHEMATIC DIAGRAM - ESW PUMP DISCHARGE AIR RELEASE VALVE
E-13EF07	ESWS CONTAIN COOLER VALVE
E-13EF07A	SCHEMATIC DIAGRAM ESW TO CONTAINMENT AIR COOLERS ISOLATION VALVE EFHV0032
E-K3EF07	SCHEMATIC DIAGRAM - SYSTEM BACKPRESSURE CONTROL VALVE
E-13EF08	SCHEM DIAG ESW FROM CONTAIN-MENT AIR COOLERS ISOL
E-K3EF08	SCHEMATIC DIAGRAM - MISCELLANEOUS CIRCUITS
E-13EF08A	SCHEMATIC DIAGRAM ESW TO CONTAINMENT AIR COOLERS ISOLATION VALVE EFHV0050
E-13EF09	SCHEM DIAG ESW TO/FROM CONTAIN AIR COOL ISOL VLV
E-13EF09A	ESW TO/FROM CONTAIN AIR COOLERS ISOLATION VALVES
E-K3EF09	SCHEMATIC DIAGRAM - STATUS PANEL CIRCUITS
E-13EF10	SCHEM DIAG ESW FROM CONTAIN AIR COOLERS ISOL VLV

TABLE 1.7-1 (SHEET 6)

DRAWING NUMBER	TITLE
E-K3EF10	SCHEMATIC DIAGRAM - MISCELLANEOUS CIRCUITS
E-13EG01A	CCW PUMPS A & B
E-13EG01B	CCW PUMPS C & D
E-13EG01C	SCHEMATIC DIAGRAM COMPONENT COOLING WATER
E-13EG01D	SCHEMATIC DIAGRAM
E-13EG03	CCW SURGE TANK VENT
E-13EG04	ESW MAKEUP TO CCW SYS
E-13EG05A	CCWS SUPPLY RETURN FROM NUCLEAR AUX COMPONENT
E-13EG05B	CCWS SUPPLY TO NUCLEAR AUX COMPONENT
E-13EG05C	SCHEMATIC DIAGRAM COMPONENT COOLING WATER RETURN FROM NUCLEAR AUX. COMPONENT EGHV0015
E-13EG05D	SCHEMATIC DIAGRAM COMPONENT COOLING WATER SUPPLY FROM NUCLEAR AUX. COMPONENT EGHV0054
E-13EG06	CCW TO CONTAINMENT ISO VALVES
E-13EG07	CCW SUPPLY TO RHR HEAT EXCHANGER
E-13EG07A	COMPONENT COOLING WATER SUPPLY TO RHR HEAT EXCHANGER EGHV0102
E-13EG08	CCW SUPPLY RETURN FROM RADWASTE BLDG
E-13EG09	CCW CONTAINMENT ISO VALVES
E-13EG10	CONTAINMENT ISO VALVE RETURN FROM THERM BAR COOL
E-13EG16	CCW HEAT EXCHANGER OUTLET TEMP CONT VALVES
E-13EG17	SCHEMATIC DIAGRAM ISOMETRIC VALVES
E-13EG18	SCHEMATIC DIAGRAM ISOMETRIC VALVES
E-13EG19	SCHEMATIC DIAGRAM ISOMETRIC VALVES
E-13EG20	SCHEMATIC DIAGRAM ISOMETRIC VALVES
E-13EJ01	RESIDUAL HEAT REMOVAL PUMPS 1 & 2 SCHEM DIAGRAM
E-13EJ02	NUCLEAR SAMPLE LINE VALVES SCHEMATIC DIAGRAM
E-13EJ03	RHR CROSS CONNECT VALVES SCHEMATIC DIAGRAM
E-13EJ04A	RHR PUMP TO CHG PUMP VLV SCH DIA
E-13EJ04B	RHR PUMP TO CHARGING PUMP VALVE
E-13EJ05A	RHR PUMP TO CHG PUMP VLV SCH DIAG
E-13EJ05B	RHR LOOP INLET ISO VALVE
E-13EJ06A	RHR PUMP TO CHG PUMP VLV SCH DIAG
E-13EJ06B	SUMP TO RESIDUAL HEAT REMOVAL PUMP

TABLE 1.7-1 (SHEET 7)

DRAWING NUMBER	TITLE
E-13EJ07	RESIDUAL HEAT REMOV HOT & COLD LEG TEST LINE
E-13EJ08	RESIDUAL HEAT REMOV MINI FLOW VALVES
E-13EJ08A	SCHEMATIC DIAGRAM RESIDUAL HEAT REMOVAL PUMP MINIFLOW VALVE EJFCV0611
E-13EJ09A	RHR COLD & HOT LEG VALVES
E-13EJ09B	RHR COLD & HOT LEG VALVES
E-13EJ09C	SCHEMATIC DIAGRAM RHR TO COLD AND HOT LEG VALVES EJHV8809B AND EJHV8840
E-13EJ13	SCHEMATIC DIAGRAM
E-13EJ14	SCHEMATIC DIAGRAM
E-13EM01	HIGH HEAD SAFETY INJECTION SCHEMATIC
E-13EM02	BORON INJ TANK DISCH & INLET ISOLA VALVES
E-13EM02A	SCHEMATIC DIAGRAM BORON INJECTION TANK DISCHARGE & INLET ISOLATION VALVE EMHV8801B
E-13EM02B	SCHEMATIC DIAGRAM BORON INJECTION TANK DISCHARGE ISOLATION VALVE EMHV8803B
E-13EM02C	SCHEMATIC DIAGRAM BORON INJECTION TANK DISCHARGE AND INLET ISOLATION VALVE EMHV8801A
E-13EM03	SIS MINI FLOW ISOLATION VALVES
E-13EM04	SAFETY INJ & CHARGING PUMPS HOT & COLD LEG TEST
E-13EM04A	SCHEMATIC DIAGRAM CHARGING PUMPS COLD LEG
E-13EM08	SUCTION HEADER & SAFETY INJ CROSS CONNECTION
E-13EM09	SAFETY INJECTION PUMP SUCTION VALVES
E-13EM11	SAFETY INJECTION PUMP TO HOT LEG MOV
E-13EM12	ACCUMULATOR FILL LINE & TEST HEADER LINE
E-13EM13A	SAFETY INJ PUMP TO HOT & COLD LEGS
E-13EM13B	SAFETY INJ PUMP TO HOT & COLD LEGS
E-13EM17	SCHEMATIC DIAGRAM
E-13EN01	CONTAINMENT SPRAY SYS SCHEMAT DIAG
E-13EN02	CONTAINMENT SPRAY PUMP SUCT SCHEMATIC DIAGRAM
E-13EN03	CONT SPRAY P.P ISOLATION SCH DIAG
E-13EN04	CONT SPRAY PP ISO DIAG
E-13EP01	ACCUMULATOR N2 SUPP ISO VLV
E-13EP02A	ACCUMULATOR ISO VLV
E-13EP02B	ACCUMULATOR ISO VLV
E-13EP07	SD ACCUMULATOR ISO VLV
E-13EP09	SAFETY DRY ACCUMULATOR VENT VLVS

TABLE 1.7-1 (SHEET 8)

DRAWING NUMBER	TITLE
E-13FC21	SCH DIAG SGFP
E-13FC23	SCH DIAG AUX WATER TURB TRIP & THROTTLE VALVE
E-13FC24	SCH DIAG AUX WATER TURB TRIP & THROTTLE VALVE
E-13FC25	SCH DIAG AUX WATER TURB TRIP & THROTTLE VALVE
E-13FC26	SCH DIAG AUX WATER TURB TRIP & THROTTLE VALVE
E-13FC27	SGFPT ISOLATION INPUT TO ESFAS
E-K3GD01	SCHEMATIC DIAGRAM - ESW PUMP ROOM SUPPLY FANS
E-K3GD02	SCHEMATIC DIAGRAM - ESW PUMP ROOM UNIT HEATERS
E-K3GD03	SCHEMATIC DIAGRAM - ESW PUMP ROOM EXHAUST VALVE
E-K3GD04	SCHEMATIC DIAGRAM - ESW PUMP ROOM MISCELLANEOUS CIRCUITS
E-K3GD05	SCHEMATIC DIAGRAM - ESW VALVE PIT UNIT HEATER
E-13GE18	COND AIR REMOVAL FILTER SYS DAMPERS
E-13GF01	AUX FDWTR PUMP RM COOLS FANS
E-13GF07	MAIN STREAM ENCL BLDG EXHAUST FANS & DAMPERS
E-13GF08	TENDON GALLERY SUPPLY/RETURN ISOL DAMPERS
E-13GF13	MAIN STEAM ENCLOSURE BLDG MISC DAMPERS
E-13GF14	MISC MOTOR SPACE HEATERS
E-13GG01	EMERGING EXHAUST FAN & DIS- CHARGE DAMPERS
E-13GG02	SPENT FUEL PUMP ROOM COOLERS
E-13GG03	EMERGENCY EXHAUST HTG COILS
E-13GG08	FUEL BLDG ISOL DAMPERS OUTSIDE AIR SUPPLY
E-13GG09	FUEL BLDG EXHAUST TO EMERG FILT ADSORB UNITS ISO
E-13GG10	EMERG FILTER ADSORB UNITS AUX BLDG INTAKE ISOL
E-13GG11	SPENT FUEL POOL DISCHARGE TO AUX BLDG DAMPER
E-13GG12	FUEL BLDG INSTRUMENTATION
E-13GG15	EMERG EXHAUST CROSS CONNECTION DAMPER
E-13GG17	SPENT FUEL POOL NORMAL/EMRGNCY EXHAUST RADIOACTIVITY SAMPLE

TABLE 1.7-1 (SHEET 9)

DRAWING NUMBER	TITLE
E-13GG18	MANUAL INITIATION FUEL BLDG ISOL SIGNAL
E-13GK01A	SCHEMATIC DIAG CONTROL ROOM FAN DAMPER
E-13GK01B	CONTROL ROOM HVAC SYSTEM
E-13GK02A	CONTROL ROOM A/C UNIT FAN
E-13GK02B	CONTROL ROOM A/C UNIT FAN
E-13GK02C	CONTROL ROOM A/C UNIT SUP
E-13GK03A	FIRE ISOLATION DAMPERS
E-13GK03B	MISCELLANEOUS DAMPERS
E-13GK07	MISCELLANEOUS DAMPERS
E-13GK09A	ISO DAMPERS
E-13GK09B	ISO DAMPERS
E-13GK10A	CONTROL ROOM PRESSURIZATION FAN & SUP DAMPERS
E-13GK10B	CONTROL ROOM PRESSURIZATION FAN & SUPPLY DAMPERS
E-13GK11	SCH DIAG CONTROL ROOM
E-13GK13	CLASS IE ELE A/C UNIT
E-13GK13A	SCHEMATIC DIAGRAM
E-13GK17	SCH DIAGRAM ISO DAMPERS
E-13GK19	SCH DIAGRAM AIR RETURN CONT RM ASB UNITS
E-13GK23	SCH DIAG FIRE ISO DAMPERS
E-13GK25	SCH DIAGRAM MISCELLANEOUS ALARMS
E-13GK28	CONTROL ROOM HVAC SYSTEM
E-13GK30A	ISOLATION DAMPERS
E-13GK30B	ISOLATION DAMPERS
E-13GK31	FIRE SIGN ISOMETRIC
E-13GL02	AUX BLDG SUPPLY AIR UNIT SUPPLY DAMPERS
E-13GL05	PUMP ROOM COOLERS
E-13GL06	PUMP ROOM COOLERS
E-13GL12	PENETRATION ROOM COOLERS
E-13GL12A	PENETRATION ROOM COOLER
E-13GL14	SCH DIAGRAM ISOL DAMPERS
E-13GL14A	SCHEMATIC DIAGRAM
E-13GL16	FUEL BLDG NORMAL EXHAUST ISOL DAMPERS
E-13GL27	CCW PUMP ROOM EXHAUST DAMP
E-13GL30	ISOLATION DAMPERS
E-13GM01	DIESEL GENERATOR VENTILATION SUPPLY FANS SCHEM
E-13GM02	DIESEL GENERATOR SUPPLY DAMPER CONTROL & MISC AL
E-13GM04	DIESEL GENERATOR BLDG EXHAUST DAMPERS SCHEMATIC
E-13GM04A	SCHEMATIC DIAGRAM

TABLE 1.7-1 (SHEET 10)

DRAWING NUMBER	TITLE
E-13GN01	HYDROGEN MIXING FANS
E-13GN02	CONTAINMENT COOLER FANS
E-13GN02A	SCHEMATIC DIAGRAM MISC FAN SPACE HEATER
E-13GN03	CRDM COOLING FANS & DISCHARGE DAMPERS
E-13GN09	MISC FAN SPACE HTRS
E-13GS01A	SCHEMATIC DIAGRAM
E-13GS01B	SCHEMATIC DIAGRAM
E-13GS02A	SCHEMATIC DIAGRAM
E-13GS02B	SCHEMATIC DIAGRAM
E-13GS03	SCHEMATIC DIAG
E-13GS04	SCHEMATIC DIAG
E-13GS05	SCHEMATIC DIAG
E-13GS06	SCHEMATIC DIAG
E-13GS07	SCHEMATIC DIAG
E-13GS10	SCHEMATIC DIAGRAM
E-13GS11	SCHEMATIC DIAGRAM
E-13GS12	SCHEMATIC DIAGRAM
E-13GS13	SCHEMATIC DIAGRAM
E-13GS14	SCHEMATIC DIAGRAM
E-13GT03	CONTAINMENT PURGE SYSTEM SCHEMATIC
E-13GT04	CONTAINMENT PURGE SYSTEM SCHEMATIC
E-13GT05	MINI PURGE FAN SCHEMATIC
E-13GT06	SCH DIAGRAM CON PUR SUP EX DM
E-13GT13	CTMT PURGE EXHAUST SMPL VLVS
E-13HB02	REACTOR COOLANT DRAIN TK PUMP DISC & VENT ISOL VALVE
E-13HB03	REACTOR COOLANT DRAIN TK VENT ISOL VLV
E-13HB19	REACTOR COOLANT DRAIN TK PUMP DISCH VLV
E-13JE01	EMERGENCY FUEL OIL TRANSFER PUMPS
E-13JE01A	SCHEMATIC DIAGRAM EMERGENCY FUEL OIL TRANSFER PUMP PJE01B
E-13JE02	MISC CIRCUITS
E-13JE03	FIRE SIGNAL ISOL RELAY
E-13KA01A	SCH DIA AIR COMPR ISOL CIRCUIT BREAKER
E-13KA02	SCH DIA COMPRESSED AIR CONTAIN-MENT ISO VLV
E-13KA04	SCH DIA HYDROGEN CONTR SYST MAKE-UP AIR VALVE
E-13KC08	FIRE PROTECTION SYSTEM ISO VLV
E-13KJ01A	DIESEL GEN KKJ01A ENG CON SCH
E-13KJ01B	DIESEL GEN KKJ01A ENG CON SCH
E-13KJ02	S.D. DIESEL GEN CONTROLS
E-13KJ03A	DG KKJ01B ENG CON SCH

TABLE 1.7-1 (SHEET 11)

DRAWING NUMBER	TITLE
E-13KJ03B	DG KKJ01B ENG CON SCH
E-13KJ04	DG KKJ01B ANNUN MISC CKTS SCH DIAG
E-13KJ05	JKT COOL & LUBE OIL LEVEL CON
E-13KJ06	DIESEL GEN KKJ01B GOV CON
E-13KJ07	DIESEL GEN KKJ01B GOV CON
E-13KJ08	STANDBY JKT COOL HTR
E-13KJ09	STANDBY JKT COOL CIRC PMP
E-13KJ10	STANDBY LUBE OIL HTR
E-13KJ11	SCH DIAGRAM LUBE OIL KEEP WARM PUMP
E-13KJ12	SCH DIAG GEN SPACE HEATER
E-13KJ13	ROCKER ARM PRE LUBE PUMP
E-13KJ16	DIESEL GEN RTD'S THERMOCOUPLES
E-13KJ20	STANDBY D/G STARTING AIR SYSTEM
E-13LF07	SUMP DISCHARGE VALVES
E-13LF08	SUMP PUMP DISCHARGE ISOLATION VALVE
E-13LF09A	REACTOR BLDG SUMP PUMP DISCHARGE ISO VLVS
E-13LF09B	REACTOR BLDG SUMP PUMP DISCHARGE ISO VALVES
E-13NB01	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB02	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB03	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB04	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB05	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB06	LOWER MED VOLTAGE SYS CLASS IE 4.16 KV 3/L METER
E-13NB10	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NB11	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NB12	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NB13	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NB14	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NB15	SCHEMAT DIAGRAM FEEDER BRKR 13.8 KV
E-13NE01	STANDBY GENER. SYS. 3/L M&R DIAG
E-13NE02	STANDBY GENER. SYS. 3/L M&R DIAG

TABLE 1.7-1 (SHEET 12)

DRAWING NUMBER	TITLE
E-13NE10	SCHEMATIC DIAGRAM
E-13NE11	SCHEMATIC DIAGRAM
E-13NE12	STANDBY GEN SYS
E-13NE13	STANDBY GEN SYS
E-13NF01	LOAD SHEDDER AND LOAD SEQUENCER
E-03NG01	CLASS IE LOW VOLTAGE SYSTEM 3/L M&R
E-K3NG01	LOW VOLTAGE SYSTEM CLASS IE 480 V THREE LINE METER & RELAY DIAGRAM
E-03NG10	CLASS IE 4.16 KV LC XFMR FDR BKRS
E-K3NG10	SCHEMATIC DIAGRAM 4.16 KV TRANSFORMER FEEDER BREAKERS
E-03NG10A	SCHEMATIC DIAGRAM
E-03NG11	BREAKER CLASS IE LOW VO
E-03NG11A	SCHEMATIC DIAGRAM
E-03NG11B	SCHEMATIC DIAGRAM 480 V LC MAIN FEEDER
E-03NG12	SCHEMATIC DIAG 480 V LC TIE BREAKER MISC COMP INPU
E-03NK10	125 VDC & 250 VDC POW SYS SCHEMATICS
E-03NK11	125 VDC CLASS IE POWER SYSTEM
E-13NN01	CLASS IE INSTRUMENT AC SCHEMATIC
E-13PA02	HIGH MED VOLT SYS 13.8 KV 3/L M&R
E-13PA05	HIGH MED VOLT SYS MED 3/L M&R
E-03PG05	NON-CLASS IE L.V. SYSTEM 3/L M&R
E-03PG12	NON-CLASS IE L.V. SYSTEM SCHEMATIC
E-03PG12A	SCHEMATIC DIAGRAM
E-13PG15B	480 V & LC TIE BKR SCHEMATIC
E-03PJ11	250 VDC POW SYS SCHEMAT
E-13PN01	NON CLASS IE INST AC LINE DIAG
E-03QB01	STANDBY LIGHTING SYSTEM PWR FDRS
E-K3QB01	SCHEMATIC DIAGRAM STANDBY LIGHTING SYSTEM POWER FEEDER
E-03QB02	STANDBY LIGHTING SYSTEM PWR FDRS
E-13SB12A	REACTOR TRIP & SAFETY INJEC SWITCHES
E-13SB12B	REACTOR TRIP AND SAFETY INJECT SWITCHES
E-13SB13	SCH DIAG SPRAY ACTUATION & CTMT ISO SWITCHES
E-03SB14	SCH DIAG SWITCH DEVEL
E-13SB15	CONTROL BOARD SWITCHES

TABLE 1.7-1 (SHEET 13)

DRAWING NUMBER	TITLE
E-13SB17	SAF INJ (RWST) RESET SWITCH/ SWITCHOVER STATUS INDICATOR
E-13SB19	SCHEMATIC DIAGRAM
E-13SJ01	NUC SAMP SYS CTMT ISO VLVS
E-13SJ02	NUC SAMP SYS SCHEMAT CTMT ISO VLVS
E-13SJ05	NUCLEAR SAMPLE SYSTEM
E-13SJ06	NUCLEAR SAMPLE SYSTEM
E-13SJ07	NUCLEAR SAMPLE SYSTEM ISO VLVS
E-13SJ09	SCHEMATIC DIAGRAM
E-13SJ11	SCHEMATIC DIAGRAM
E-1R0223	RACEWAY PLAN - STATION SERVICE AND STARTUP XFMR AREA
E-0R0224	RACEWAY PLAN - ESF TRANSFORMER AREA
E-1R3211	RACEWAY PLAN CONTROL BLDG. AREA 1
E-1R3221	RACEWAY PLAN – COMMUNICATION CORRIDOR AREA - 2 EL. 1974'-0" AND EL. 1984'-0"
E-1R3321	RACEWAY PLAN – COMMUNICATION CORRIDOR AREA - 2 EL. 2000'-0"
E-1R4321	RACEWAY PLAN - TURBINE BUILDING AREA - 2 EL. 2000'-0"
E-1R4331	RACEWAY PLAN - TURBINE BUILDING AREA - 3 EL. 2000'-0"
J-020101	SYMBOLS & LEGEND FOR LOGIC DIAGRAMS
J-02AB01	CONTROL LOGIC DIAGRAM
J-02AB02	MN STM ATMOS VENT VLVS INDICATING LIGHTS
J-02AB02A	CONTROL LOGIC DIAGRAM
J-02AB02B	CONTROL LOGIC DIAGRAM
J-02AB03	MAIN STEAM SUPPLY TD AUX FEED-WATER PUMP
J-12AB04	MAIN STEAM LINE DRAIN VALVES
J-02AB10	MAIN STEAM SYSTEM BYPASS VALVE AUX FDWTR PUMP
J-02AB12	MAIN STEAM STOP VALVE BYPASS VALVE
J-02AC06	MAIN TURBINE TRIP LOGIC LEGENDS & NOTES
J-02AC07	MAIN TURBINE TRIP BLOCK DIAGRAM
J-02AC08	MAIN TURBINE TRIP LOGIC DIAGRAM
J-12AC09	CONTROL LOGIC DIAGRAM
J-12AE08	STEAM GENERATOR CHEMICAL INJEC VALVE
J-02AL01	MOTOR DRIVEN AUX FEEDWATER PUMPS

TABLE 1.7-1 (SHEET 14)

DRAWING NUMBER	TITLE
J-02AL01A	CONTROL LOGIC DIAGRAM AUX FW SYSTEM
J-02AL01B	CONTROL LOGIC DIAGRAM AUX FW SYSTEM
J-02AL02	AUX FDWTR SYS SUPPLY FROM COND STORAGE TANK
J-02AL02A	CONTROL LOGIC DIAGRAM
J-02AL03	AUX FEEDWATER PUMPS DISC CONTROL VALVES
J-12AL04	AUX FEEDWATER PUMPS SUPPLY ESWS SYSTEM
J-12AL04A	CONTROL LOGIC DIAGRAM
J-12AL05	AUX FEEDWATER SYS PUMPS SUCTION DISC PRESS ALAR
J-02AL06	CONTROL LOGIC DIAGRAM
J-02AL07	CONTROL LOGIC DIAGRAM
J-02BB01	REACTOR COOLANT SYSTEM RCP THERM BAR ISO VALVE
J-02BM01	STEAM GENERATOR BLOWDOWN LOGIC DIAG CTMNT VLVS
J-02BM04	STU BEN BLOWDOWN ISOLATION VALVES
J-02BN01	BORON REFUELING WATER STORAGE RWST HEATER VALVE
J-02BN02	STEAM CONTROL VALVE
J-12EC01	FUEL POOL COOLING & CLEANUP SYSTEM
J-02EC02	FUEL POOL COOL CLEANUP SYS DIXC VLV HEAT EXCHAN
J-12EC05	SYSTEM ALARMS
J-12EF01A	ESWS MOTOR OPERATED ISOLATION VALVES
J-K2EF01A	ESWS LOGIC DIAGRAM
J-02EF01B	ESWS MOTOR OPERATED ISOLATION VALVES
J-K2EF01B	ESWS LOGIC DIAGRAM
J-02EF02	ESWS AIR COMPRESSORS ISOLATION VALVES
J-K2EF02A	ESWS ESW PUMPS LOGIC DIAGRAM
J-K2EF02B	ESWS ESW PUMP LUBE VALVE LOGIC DIAGRAM
J-12EF03	ESWS CTMT AIR COOLERS ISOLATION VALVES
J-K2EF03A	ESWS SELF-CLEANING STRAINER LOGIC DIAGRAM
J-K2EF03B	ESWS SELF-CLEANING STRAINER LOGIC DIAGRAM
J-12EF04	ESWS CTMT AIR COOLERS ISOL VALVE BYPASS VALVE

TABLE 1.7-1 (SHEET 15)

DRAWING NUMBER	TITLE
J-12EF05	ESWS MOTOR OPERATED ISOLATION VALVES
J-K2EF06	ESWS ESW PUMP DISCHARGE AIR RELEASE VALVE LOGIC DIAGRAM
J-02EF07	CONTROL LOGIC DIAGRAM STR SYS ESWS
J-K2EF07	ESWS BACKPRESSURE CONTROL VALVE LOGIC DIAGRAM (VOID)
J-02EG01A	COMPONENT COOLING WATER SYSTEM PUMPS
J-02EG01B	COMPONENT COOLING WATER SYSTEM PUMPS
J-02EG01C	CONTROL LOGIC DIAG COOL WTR SYSTEM
J-12EG02	CCWS DEMIN WATER MAKEUP CCW SURGE TANK
J-02EG04	CCW ESWS MAKE-UP SURGE TANK VENT
J-02EG05A	CCWS SUPPLY RETURN NUCLEAR AUX CMPNT SHEET A
J-02EG05B	CCWS SUPPLY RETURN NUCLEAR AUX CMPNT
J-02EG05C	CCWS SUP RTRN NUC AUX CMPNT VLV POSITION ALARM
J-02EG06	CCWS HX'S DISC TEMP ALARM RHR HX'S FLOW ALARM
J-02EG07	CCWS SUPPLY RHR HX'S
J-12EG08A	CCWS SUPPLY RETURN RADWASTE BLDG
J-12EG08B	CCWS SUPPLY RETURN RADWASTE BLDG
J-12EG09	CCWS CONTAINMENT ISOLATION VALVE
J-12EG10	CCWS INSIDE CTMT ISO VLV RTRN THRM BAR COOL CO
J-02EG11	CCWS HEAT EXCHANGER OUTLET TEMP CONT
J-12EG13A	CONTROL LOGIC DIAGRAM
J-02EG13B	CONTROL LOGIC DIAGRAM
J-12EG13C	CONTROL LOGIC DIAGRAM
J-12EG14	CONTROL LOGIC DIAGRAM
J-02EJ01	RESIDUAL HEAT RMVL SYS NUC SAMP SYS VALVE
J-02EJ03	RESIDUAL HEAT REMOVAL CONTROL LOGIC DIAGRAM
J-02EJ04	CONTROL LOGIC DIAGRAM
J-02EN01	CONTAINMENT SPRAY SYS CONTAIN SPRAY PUMPS
J-02EN02	CONT SPRAY SYSTEM CTMT RECIRC SUMP ISOL VALVE

TABLE 1.7-1 (SHEET 16)

DRAWING NUMBER	TITLE
J-12EN03	CONT SPRAY SYSTEM SPRAY ADD TANK ISOL VLVS
J-12EN04	CONT SPRAY SYSTEM SPRAY ADD TANK ALARMS
J-02EN05	CONT SPRAY SYSTEM CTMT SPRAY NOZZLES ISOL VLV
J-02FC18	AUX TURB AFP STEAMLINE WATER TRAP DRAIN
J-02FC19	CONTROL LOGIC DIAGRAM
J-K2GD01	ESSENTIAL SERVICE WATER PUMPHOUSE HVAC SUPPLY FANS LOGIC DIAGRAM
J-K2GD02	ESSENTIAL SERVICE WATER PUMPHOUSE HVAC UNIT HEATER LOGIC DIAGRAM
J-02GE08	TURB BLDG HVAC LOGIC DIAG COND AIR REMVL FILTER
J-02GF03	MISC BLDG HVAC MAIN STM BLDG SUP EXH DAMPER
J-12GF06	MISC BLDG HVAC TENDON GALL SUP RETURN DAMPER
J-12GF06A	CONTROL LOGIC DIAGRAM MISC. BUILDING HVAC
J-12GF07	MISC BLDG HVAC FDWTR PUMP ROOM COOLER FANS
J-12GF09A	CONTROL LOGIC DIAGRAM MISC. BUILDING HVAC
J-12GF09B	CONTROL LOGIC DIAGRAM MISC. BUILDING HVAC
J-02GG04	FB HVAC ISOLATION DAMPERS
J-02GG05	FUEL BLDG SPENT FUEL POOL ROOM COOLERS
J-12GG06	FB HVAC FILTER UNITS INTAKE ISOL DAMPERS
J-02GG07	FB HVAC FILTER UNITS AUX BLDG ISOLATION DAMPERS
J-02GG08	FB HVAC MANUAL INITIATION OF FBIS
J-02GG09	FB HVAC SPENT FUEL POOL DISC
J-02GG10A	FB HVAC EMERGENCY EXHAUST FAN
J-02GG13	FB HVAC EMERGENCY EXHAUST CROSS CONNENC DAMPERS
J-02GG14A	FUEL BLDG HVAC SPENT FUEL POOL
J-02GG14B	FUEL BLDG HVAC SPENT FUEL POOL
J-12GK01A	LOGIC DIAG CONTROL ROOM FILTRATION FAN
J-12GK01B	FILTER ABSORBER UNIT SUPPLY
J-12GK01C	FILTER ABSORBER UNIT DISK DAMPER
J-12GK01D	CONTROL ROOM RECIRC DAMPERS CONTROL BLDG HVAC

TABLE 1.7-1 (SHEET 17)

DRAWING NUMBER	TITLE
J-12GK01E	CONTROL BLDG HVAC CONTROL A/C UNIT DISCHARG DMP
J-02GK02A	CONTROL ROOM A/C UNIT FAN & TEMP
J-02GK02B	CONTROL ROOM A/C UNIT SUPPLY DAMPER
J-02GK02C	CONTROL BLDG HVAC CONT ROOM A/C UNIT DISC DAMPE
J-02GK03A	CONTROL BLDG HVAC FIRE ISO-LATION DAMPERS
J-02GK07	MISC DAMPERS
J-02GK09	ISO DAMPERS
J-12GK10A	CONTROL ROOM PRESS FAN
J-12GK10B	CONTROL ROOM PRESS SUPPLY DAMPER
J-12GK10C	CONT BLDG HVAC CONT ROOM PRESS SYS UNIT SUP DAM
J-12GK11	CONTROL ROOM PRESS SYSTEM FIL UNIT
J-12GK13	CLASS IE ELEC EQUIP A/C UNIT
J-12GK15	CHLORINE ALAR
J-12GK17A	CONTROL LOGIC DIAGRAM CONTROL BLDG HVAC
J-12GK17B	CONTROL LOGIC DIAGRAM CONTROL BLDG HVAC
J-02GK19	CONT BLDG HVAC RETRN DAMPERS RM FLTR ABSORB UNI
J-02GK23	CONTROL BLDG HVAC FIRE ISO DAMPERS
J-02GK25	CONTROL LOGIC DIAGRAM FOR CONTROL BLDG HVAC
J-02GK26	CONTROL LOGIC DIAGRAM FOR CONTROL BLDG HVAC
J-02GK27	CONTROL LOGIC DIAGRAM FOR CONTROL BLDG HVAC
J-02GL01A	AUX BLDG HVAC AIR SUPPLY UNIT
J-02GL01B	AUX BLDG HVAC AIR SUPPLY
J-12GL03A	AUX BLDG HVAC MISC ROOM COOLERS
J-12GL03B	AUX BLDG HVAC MISC ROOM COOLER
J-12GL03C	AUX BLDG HVAC MISC ROOM COOLER
J-12GL11	AUX BLDG HVAC PENE RM COOLER FANS
J-02GL13	AUX BLDG HVAC ISO DAMPERS
J-02GL15	AUX BLDG HVAC FUEL BLDG HVAC DISCH ISO DAMPERS
J-02GL21	AUX BLDG HVAC CCW PUMP ROOM EXH DAMPERS
J-12GL23	AUX BLDG HVAC AUX/FUEL BLDG EXH FANS DISC

TABLE 1.7-1 (SHEET 18)

DRAWING NUMBER	TITLE
J-12GM01A	DIESEL GENERATOR BLDG HVAC FAN & DAMPER
J-02GM01B	DIESEL GENERATOR BLDG EXH DAMPERS
J-12GN01	CONTAINMENT COOLING
J-12GN02A	CONTROL LOGIC DIAGRAM CONTAINMENT COOLING SYS
J-12GN02B	CONTROL LOGIC DIAGRAM CONTAINMENT COOLING SYS
J-02GN03A	CONTAINMENT COOLING FANS
J-02GS02	CTMT HYDROGEN CONTROL THERMAL HYDROGEN RECOM
J-02GS03	CTMT HYDROGEN CONTROL SOLENOID ISO VLV
J-02GS06	CTMT HYDROGEN CONTROL PURGE SUBSYS ISO VLV
J-02GS08	CONTROL LOGIC DIAGRAM
J-02GS09	CONTROL LOGIC DIAGRAM
J-02GS10	CONTROL LOGIC DIAGRAM
J-02GT03	CONTAINMENT PURGE SYSTEM VALVES
J-02GT06	CONTAINMENT PURGE SYS ISO DAMPERS
J-02GT10	CONTROL LOGIC DIAGRAM
J-12JE01	EMERGENCY FUEL OIL TRANSFER PUMPS
J-12JE02	CONTROL LOGIC DIAGRAM
J-02KA02	COMPRESSED AIR SYS COMP AIR CONTAIN ISO VALVE
J-02KA03	COMPRESSED AIR SYS HYDROGEN CS M/U AIR VALVE
J-02KA08	CONTROL LOGIC DIAGRAM COMPRESSED AIR SYSTEM
J-02KC08	CONTROL LOGIC DIAG FIRE PROTECTION SYSTEM
J-12KJ02	CONTROL LOGIC DIAGRAM
J-02KJ03	CONTROL LOGIC DIAGRAM
J-12LF03	FLOOR & EQUIP DRAINS REACT BLDG SUMP PUMP ISO VL
J-02LF04	FLOOR & EQUIP DRAINS REACT BLDG SUMP PMP ISO VLV
J-02LF08	FLOOR & EQUIP DRAINS DISCHARGE VALVES
J-02RP01	CONTROL LOGIC DIAGRAM
J-02RP01A	CONTROL LOGIC DIAGRAM
J-12SA03	CONTROL LOGIC DIAGRAM
J-12SA04	CONTROL LOGIC DIAGRAM
J-12SA05	CONTROL LOGIC DIAGRAM
J-02SJ01	NUC SYSTEM CON ISOLATION VALVES
J-02SJ03	CONTROL LOGIC DIAGRAM

TABLE 1.7-1 (SHEET 19)

DRAWING NUMBER	TITLE
J-14001	CONTROL ROOM EQUIPMENT ARRANGEMENT
J-14002	REACT COOL SUPRT SYSTEM CONSOLE RL001 & RL002
J-14003	REACTOR OPERATORS CONSOLE RL003 & RL004
J-14004	TURB GEN & FEEDWATER CONSOLE RL005 & RL006
J-14005	SITE RELATED MAIN CONT BOARD RL013 & RL014
J-14006	STATION ELEC DIST MAIN CONT BOARD RL015 & RL016
J-14007	ENG SAFETY FEATURES MAIN CONT BOARD RL017 & RL018
J-14008	ENG SAFETY FEATURES MAIN CONT BOARD RL019 & RL020
J-14009	REACT AUX MAIN CONTROL BOARD RL021 & RL022
J-14010	TURBOGENERATORS & FDWTR MAIN CONT BD RL023 & RL024
J-14011	TURBOGENERATORS & FDWTR MAIN CONT BD RL025 & RL026
J-14013	END SECTION MAIN CONT BRD RL011 & RL012
J-04014	MAIN CONTROL BOARD DETAILS
J-04015	OPERATORS CONSOLE DETAILS
J-14016	BILL OF MAT MAIN CONTROL PANEL
J-05001	AUX CONTROL PANEL DWGS
J-05002	AUX SHUTDOWN PANEL DETAILS
J-05003	AUX SHUTDOWN PANEL RP 118 7 SHTS
J-05021	LOCAL CONT PANEL RPO 68 MISC BOP INST PANEL
J-15023	MISC BOP INST PANEL RP068 BILL OF MAT
J-K5041	ESSENTIAL SERVICE WATER PANEL (EF 155 & EF 156) DRAWING
J-K5042	ESSENTIAL SERVICE WATER PANEL (EF 155) BILL OF MATERIAL
J-K5043	ESSENTIAL SERVICE WATER PANEL (EF 156) BILL OF MATERIAL
J-16002	ANN WINDOW ARRGT RK016.018.020
J-16003	ANN WINDOW ARRGT RK022.024.026

TABLE 1.7-1 (SHEET 20)

DRAWING NUMBER	TITLE
M-761-0066 Through 0104 and 0459, 0460, 0494 (8756d37*)	PROCESS CONTROL BLOCK DIAGRAM (1 THROUGH 42)
M-762-0001 (5655D49*)	NIS SOURCE RANGE FUNCTIONAL BLOCK DIAGRAM
M-762-0002 (5655D50*)	NIS INTERMEDIATE RANGE FUNCTIONAL BLOCK DIAGRAM
M-762-0417 (9552D32*)	NIS POWER RANGE FUNCTIONAL BLOCK DIAGRAM
M-762-0032 (5655D52*)	NIS AUXILIARY CHANNELS FUNCTIONAL BLOCK DIAGRAM
M-767-221 THROUGH 240 (8761D17*)	SAFEGUARDS TEST CABINET (1 THROUGH 20)
J-104-00347	INSTRUCTION MANUAL ESFAS/LSELS
J-104-00437	SIGNAL FLOW BLOCK DIAGRAM FOR THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (SHEET 1)
J-104-00438	ISOLATION AND DISTRIBUTION OF ANALOG SIGNALS TO COMPUTER (ESFAS SHEET 1A)
J-104-00439	CHANNEL IV BLOCK DIAGRAM (ESFAS SHEET 2)
J-104-00440	CHANNEL II BLOCK DIAGRAM (ESFAS SHEET 3)
J-104-00441	ATI BLOCK DIAGRAM (ESFAS SHEET 4)
J-104-00442	BISTABLE RACK - CHANNEL I (ESFAS SHEET 5)
J-104-00443	ISOLATION RACK - CHANNEL I (ESFAS SHEET 6)
J-104-00444	ACTUATION INPUTS - CHANNEL I (ESFAS SHEET 7)
J-104-00445	ACTUATION OUTPUTS AND STATUS INDICATIONS - CHANNEL I (ESFAS SHEET 8)
J-104-00446	ANNUNCIATOR/COMPUTERS OUTPUTS - CHANNEL I (ESFAS SHEET 9)

* Drawings suplied by NSSS Vendor.

TABLE 1.7-1 (SHEET 21)

DRAWING NUMBER	TITLE
J-104-00447	ANALOG SIGNALS - CHANNEL II (ESFAS SHEET 10)
J-104-00448	ISOLATION RACK LOGIC SIGNALS - CHANNEL II (ESFAS SHEET 11)
J-104-00449	RELAY OUTPUTS - CHANNEL II (ESFAS SHEET 12)
J-104-00450	BISTABLE RACK - CHANNEL IV (ESFAS SHEET 13)
J-104-00451	ISOLATION RACK - CHANNEL IV (ESFAS SHEET 14)
J-104-00452	ACTUATION INPUTS - CHANNEL IV (ESFAS SHEET 15)
J-104-00453	ACTUATION OUTPUTS AND STATUS INDICATIONS - CHANNEL IV (ESFAS SHEET 16)
J-104-00454	ANNUNCIATOR/COMPUTER OUTPUTS - CHANNEL IV (ESFAS SHEET 17)
J-104-00455	ATI MODULE A (ESFAS SHEET 18)
J-104-00456	ATI MODULE B (ESFAS SHEET 19)
J-200-00029	TURB SUPERVISORY MAIN CONT BRD RL027 & RL028

TABLE 1.7-2

PIPING AND INSTRUMENTATION DIAGRAMS

Drawing Number	Figure Number	Sheet	Title
M-120101	1.1-1	1	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams
M-120102	1.1-1	2	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams
M-120103	1.1-1	3	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams
M-120104	1.1-1	4	Symbols and Legend for System Flow and Piping & Instrumentation Diagrams
M-12AB01	10.3-1	1	Main Steam System
M-12AB02	10.3-1	2	Main Steam System
M-12AB03	10.3-1	3	Main Steam System
M-12AC01	10.2-1	1	Main Turbine
M-12AC02	10.2-1	2	Main Turbine
M-12AC03	10.2-1	3	Main Turbine
M-12AC04	10.2-1	4	Main Turbine
M-12AD01	10.4-2	1	Condensate System
M-12AD02	10.4-2	2	Condensate System
M-12AD03	10.4-2	3	Condensate System
M-12AD04	10.4-2	4	Condensate System
M-12AD05	10.4-2	5	Condensate System
M-12AD06	10.4-2	6	Condensate System
M-12AE01	10.4-6	1	Feedwater System
M-12AE02	10.4-6	2	Feedwater System
M-12AF01	10.4-6	3	Feedwater Heater Extraction Drains & Vents
M-12AF02	10.4-6	4	Feedwater Heater Extraction Drains & Vents
M-12AF03	10.4-6	5	Feedwater Heater Extraction Drains & Vents
M-12AF04	10.4-6	6	Feedwater Heater Extraction Drains & Vents
M-12AK01	10.4-5	1	Condensate Demineralizer System
M-12AK02	10.4-5	2	Condensate Demineralizer System
M-12AK03	10.4-5	3	Condensate Demineralizer System

TABLE 1.7-2 (SHEET 2)

Drawing Number	Figure Number	Sheet	Title
M-12AL01	10.4-9		Auxiliary Feedwater System
M-12AN01	9.2-16		Demineralized Water Storage and Transfer System
M-12AP01	9.2-23		Condensate Storage and Transfer System
M-12AQ01	10.4-7	1	Condensate Chemical Addition System
M-12AQ02	10.4-7	2	Feedwater Chemical Addition System
M-12BB01	5.1-1	1	Reactor Coolant System
M-12BB02	5.1-1	2	Reactor Coolant System
M-12BB03	5.1-1	3	Reactor Coolant System
M-12BB04	5.1-1	4	Reactor Coolant System
M-12BG01	9.3-8	1	Chemical and Volume Control System
M-12BG02	9.3-8	2	Chemical and Volume Control System
M-12BG03	9.3-8	3	Chemical and Volume Control System
M-12BG04	9.3-8	4	Chemical and Volume Control System
M-12BG05	9.3-8	5	Chemical and Volume Control System
M-12BL01	9.2-13		Reactor Makeup Water System
M-12BM01	10.4-8	1	Steam Generator Blowdown System
M-12BM02	10.4-8	2	Steam Generator Blowdown System
M-12BM03	10.4-8	3	Steam Generator Blowdown System
M-12BM04	10.4-8	4	Steam Generator Blowdown System
M-12BM05	10.4-8	5	Steam Generator Blowdown System
M-12BN01	6.3-1	1	Borated Refueling Water Storage System
M-12CA01	10.4-4		Steam Seal System
M-12CF01	10.2-1	5	Lube Oil Storage, Transfer and Purification System
M-12CF02	10.2-1	6	Lube Oil Storage, Transfer and Purification System
M-12CG01	10.4-3		Condenser Air Removal
M-12CH01	10.2-1	7	Main Turbine Control Oil System
M-12CH02	10.2-1	8	Main Turbine Control Oil System

TABLE 1.7-2 (SHEET 3)

Drawing Number	Figure Number	Sheet	Title
M-12DA01	10.4-1	1	Circulating Water & Water Box Drains System
M-12DA02	10.4-1	3	Circulating Water & Water Box Venting System
M-12EA01	9.2-1	1	Service Water System
M-12EA02	9.2-1	2	Service Water System
M-12EB01	9.2-14		Closed Cooling Water System
M-12EC01	9.1-3	1	Fuel Pool Cooling and Cleanup System
M-12EC02	9.1-3	2	Fuel Pool Cooling and Cleanup System
M-12EF01	9.2-2	1	Essential Service Water System
M-12EF02	9.2-2	2	Essential Service Water System
M-K2EF01	9.2-2	3	Essential Service Water System
M-K2EF03	9.2-2	4	Essential Service Water System
M-12EG01	9.2-15	1	Component Cooling Water System
M-12EG02	9.2-15	2	Component Cooling Water System
M-12EG03	9.2-15	3	Component Cooling Water System
M-12EJ01	5.4-7		Residual Heat Removal System
M-12EM01	6.3-1	2	High Pressure Coolant Injection System
M-12EM02	6.3-1	3	High Pressure Coolant Injection System
M-12EN01	6.2.2-1		Containment Spray System
M-12EP01	6.3-1	4	Accumulator Safety Injection
M-12FA01	9.5.9-1	1	Auxiliary Boiler System
M-12FB01	9.5.9-1	2	Auxiliary Steam System
M-12FB02	9.5.9-1	3	Auxiliary Steam System
M-12FC02	10.4-10		Auxiliary Feedwater Pump Turbine
M-12FC03	10.4-6	7	S.G.F.P. Turbine "A"
M-12FC04	10.4-6	8	S.G.F.P. Turbine "B"
M-12FE01	9.5.9-1	4	Auxiliary Steam Chemical Addition System
M-12GA01	9.4-9	1	Plant Heating System
M-12GA02	9.4-9	2	Plant Heating System
M-12GB01	9.4-10		Central Chilled Water System
M-K2GD01	9.4-8	1	Essential Service Water Pump House HVAC

TABLE 1.7-2 (SHEET 4)

Drawing Number	Figure Number	Sheet	Title
M-12GE01	9.4-4	1	Turbine Building HVAC
M-12GE02	9.4-4	2	Turbine Building HVAC
M-12GE03	9.4-4	3	Turbine Building HVAC
M-12GE04	9.4-4	4	Turbine Building HVAC
M-12GF01	9.4-3	1	Miscellaneous Buildings HVAC
M-12GF02	9.4-3	2	Miscellaneous Buildings HVAC
M-12GG01	9.4-2	1	Fuel Building HVAC
M-12GG02	9.4-2	2	Fuel Building HVAC
M-12GH01	9.4-5	1	Radwaste Building HVAC
M-12GH02	9.4-5	2	Radwaste Building HVAC
M-12GK01	9.4-1	1	Control Building HVAC
M-12GK02	9.4-1	2	Control Building HVAC
M-12GK03	9.4-1	3	Control Building HVAC
M-12GK04	9.4-1	4	Control Building HVAC
M-12GL01	9.4-3	5	Auxiliary Building HVAC
M-12GL02	9.4-3	4	Auxiliary Building HVAC
M-12GL03	9.4-3	3	Auxiliary Building HVAC
M-12GM01	9.4-7		Diesel Generators Building HVAC
M-12GN01	9.4-6	1	Containment Cooling System
M-12GN02	9.4-6	2	Containment Cooling System
M-12GP01	6.2.6-1		Containment Integrated Leak Rate Test
M-12GR01	9.4-6	3	Containment Atmospheric Control System
M-12GS01	6.2.5-1		Containment Hydrogen Control System
M-12GT01	9.4-6	4	Containment Purge System HVAC
M-12HA01	11.3-1	1	Gaseous Radwaste System
M-12HA02	11.3-1	2	Gaseous Radwaste System
M-12HA03	11.3-1	3	Gaseous Radwaste System
M-12HB01	11.2-1	1	Liquid Radwaste System
M-12HB02	11.2-1	2	Liquid Radwaste System
M-12HB03	11.2-1	3	Liquid Radwaste System
M-12HB04	11.2-1	4	Liquid Radwaste System
M-12HC01	11.4-1	1	Solid Radwaste System
M-12HC02	11.4-1	2	Solid Radwaste System
M-12HC03	11.4-1	3	Solid Radwaste System
M-12HC04	11.4-1	4	Solid Radwaste System
M-12HD01	12.3-4		Decontamination System
M-12HE01	9.3-11	1	Boron Recycle System
M-12HE02	9.3-11	2	Boron Recycle System
M-12HE03	9.3-11	3	Boron Recycle System

TABLE 1.7-2 (SHEET 5)

Drawing Number	Figure Number	Sheet	Title
M-12HF01	10.4-12	1	Secondary Liquid Waste System
M-12HF02	10.4-12	2	Secondary Liquid Waste System
M-12HF03	10.4-12	3	Secondary Liquid Waste System
M-12HF04	10.4-12	4	Secondary Liquid Waste System
M-12JE01	9.5.4-1		Emergency Fuel Oil System
M-12KA01	9.3-1	1	Compressed Air System
M-12KA02	9.3-1	2	Compressed Air System (Service Air)
M-12KA03	9.3-1	3	Instrument Air System
M-12KA04	9.3-1	4	Instrument Air System
M-12KA05	9.3-1	5	Compressed Air System
M-12KA06	9.3-1	6	Compressed Air System
M-12KA07	9.3-1	7	Compressed Air System
M-12KB01	9.5.10-1	1	Breathing Air System
M-12KB02	9.5.10-1	2	Breathing Air System
M-12KB03	9.5.10-1	3	Breathing Air System
M-12KC01	9.5.1-1	1	Fire Protection System
M-12KC02	9.5.1-1	2	Fire Protection System
M-12KC03	9.5.1-1	3	Fire Protection System
M-12KC04	9.5.1-1	4	Fire Protection (Halon) System
M-12KC05	9.5.1-1	5	Fire Protection System
M-12KC06	9.5.1-1	6	Fire Protection (Halon) System
M-12KC07	9.5.1-1	7	Fire Protection (Halon) System
M-12KD01	9.2-17	1	Domestic Water System
M-12KD02	9.2-17	2	Domestic Water System
M-12KH01	9.3-9	1	Service Gas System
M-12KH02	9.3-9	2	Service Gas System
M-12KJ01	9.5.5-1	1	Standby Diesel Generator "A" Cooling Water System
M-12KJ04	9.5.5-1	2	Standby Diesel Generator "B" Cooling Water System
M-12KJ02	9.5.6-1	1	SDG "A" Intake, Exh., F.O. & Starting Air System
M-12KJ05	9.5.6-1	2	SDG "B" Intake, Exh., F.O. & Starting Air System
M-12KJ03	9.5.7-1	1	Standby Diesel Generator "A" Lube Oil System
M-12KJ06	9.5.7-1	2	Standby Diesel Generator "B" Lube Oil System

TABLE 1.7-2 (SHEET 6)

Drawing Number	Figure Number	Sheet	Title
M-12LA01	9.3-5	1	Sanitary Lift Station & Turb. Bldg. Sanitary Drainage System
M-12LA02	9.3-5	2	Comm. Corridor & Control Bldg. Sanitary Drainage System
M-12LD01	9.3-5	3	Chemical and Detergent Waste
M-12LE01	9.3-5	4	Turb. Bldg. & Aux. Feedwater Pump Rooms Oily Waste System
M-12LE02	9.3-5	5	Control & Diesel Gen. Bldg. Oily Waste System
M-12LE03	9.3-5	6	Turb. Bldg. & Aux. Boiler Room Oily Waste System
M-12LE04	9.3-5	7	Tendon Access Gallery & Turb. Bldg. Oily Waste System
M-12LF01	9.3-5	8	Aux. Bldg. Floor and Equip-ment Drain System
M-12LF02	9.3-5	9	Aux. Bldg. Floor and Equip-ment Drain System
M-12LF03	9.3-5	10	Aux. Bldg. Floor and Equip-ment Drain System
M-12LF04	9.3-5	11	Aux. Bldg. Floor and Equip-ment Drain System
M-12LF05	9.3-5	12	Aux. Bldg. Floor and Equip-ment Drain System
M-12LF06	9.3-5	13	Radwaste & Fuel Bldgs. FED System
M-12LF07	9.3-5	14	Radwaste Bldg. FED System
M-12LF08	9.3-5	15	Control and Fuel Bldgs. FED System
M-12LF09	9.3-5	16	Reactor Bldg. & Hot Machine Shop FED System
M-12LF10	9.3-5	17	Radwaste Bldg. and Tunnel FED System
M-12RM01	9.3-4	1	Process Sampling System
M-12RM02	9.3-4	2	Process Sampling System
M-12RM03	9.3-4	3	Process Sampling System
M-12SJ01	9.3-2	1	Nuclear Sampling System Primary Sampling System
M-12SJ02	9.3-3		Nuclear Sampling System Radwaste Sampling System
M-12SJ03	9.3-2	2	Nuclear Sampling System Primary Sampling System

TABLE 1.7-2 (SHEET 7)

Drawing Number	Figure Number	Sheet	Title
M-12SJ04	18.2-15		Nuclear Sampling System
M-12WT01	9.2-24	1	Wastewater Treatment Facility
M-12WT03	9.2-25	1	Wastewater Treatment Facility
M-12VW01	9.4-11		Wastewater Treatment Facility HVAC System
M-0021	10.4-1	2	Circulating Water System
M-0022	9.2-1	3	Service Water System
M-0023	9.5-1	1	Fire Protection System (Site)
M-0023	9.5-1	2	Fire Protection System (Site)
M-0023	9.5-1	3	Fire Protection System (Site)
M-0023	9.5-1	4	Fire Protection System (Site)
M-0025	9.2-5	1	Demineralized Water Makeup System
M-0025	9.2-5	2	Demineralized Water Makeup System
M-0025	9.2-5	3	Demineralized Water Makeup System
M-0025	9.2-5	4	Demineralized Water Makeup System
M-0025	9.2-5	4A	Demineralized Water Makeup System
M-0051	9.5-2		Outdoor Piping, Key Plans & General Notes

TABLE 1.7-3

ADDITIONAL CONTROLLED DRAWINGS USED IN THE USAR

Drawing Number	Figure Number	Sheet	Title
A-0503	9.2-5A		Potable Water System
10466-A-1701	12.3-2	1	Radiation Zones For Normal Operation
10466-A-1702	12.3-2	2	Radiation Zones For Normal Operation
10466-A-1703	12.3-2	3	Radiation Zones For Normal Operation
10466-A-1704	12.3-2	4	Radiation Zones For Normal Operation
10466-A-1705	12.3-2	5	Radiation Zones For Normal Operation
10466-A-1706	12.3-2	6	Radiation Zones for Normal Operation
10466-A-1801	9.5.1-2	1	Fire Delineation Floor Plan EL. 1974'-0"
10466-A-1802	9.5.1-2	2	Fire Delineation Floor Plan EL. 2000'-0"
10466-A-1803	9.5.1-2	3	Fire Delineation Floor Plan EL. 2026'-0"
10466-A-1804	9.5.1-2	4	Fire Delineation Floor Plan EL. 2047'-0"
8025-C-KG1202	1.2-44		Site Plan
C-0L2931	9.3-7	1	Reactor Building Stainless Steal Liner Plate Reactor Refueling Canal
C-1L6111	9.3-7	2	Reactor Building Stainless Steel Liner Plate Reactor Refueling Canal
C-03BB53	5.4-21		Hot and Cold Leg Lateral Restraints
E-1L9901	9.5.3-1		Lighting Distribution Riser Diagram
E-1L9903	9.5.2-2		Public Address System Riser Diagram
E-11005	8.3-2		List of Loads Supplied by the Emergency Diesel Generator
E-1001	8.3-1	3	Single Line Diagram Site Area Loads
E-K1001	8.3-1	2	Single Line Diagram Essential Service Water System
E-11001	8.3-1	1	Main Single Line Diagram
E-11010	8.3-6	1	DC Main Single Line Diagram
E-11010A	8.3-7		DC Main Single Line Diagram (PK03 & PK04 Bus)
E-12KJ01	8.3-5		Standby Generator Engine and Governor Control Logic Diagram

TABLE 1.7-3 (Sheet 2)

Drawing Number	Figure Number	Sheet	Title
E-12NE01	8.3-3		Logic Diagram Standby Generator Excitation Control
E-12NE02	8.3-4		Logic Diagram Standby Generator System Protection
E-14QE01	9.5.2-1		Telephone System Riser Diagram
J-104-00390	7.3-1	2	Logic Diagram Engineered Safety Features Actuation System (BOP)
KD-7496	8.2-4		WCGS Electrical One-Line Diagram
KD-7750	8.2-3		Wolf Creek Substation General Plan
M-0004	10.4-1	4	Circulating Water Screenhouse Planview
M-0005	10.4-1	5	Circulating Water Screenhouse Section View
M-1G001	1.2-1		Peninsular Plant Arrangement Standard Power System & Structure Interface
M-1G010	1.2-2		Equipment Location Radwaste Building Plan EL 1976'-0"
M-1G011	1.2-3		Equipment Location Radwaste Building Plan EL 2000'-0"
M-0G012	1.2-4		Equipment Location Radwaste Building Plan EL 2022'-0"
M-1G013	1.2-5		Equipment Location Radwaste Building Plan EL 2031'-6" and Roof Plan
M-1G014	1.2-6		Equipment Location Radwaste Building Sections A & B
M-1G015	1.2-7		Equipment Location Building Sections C & E
M-1G016	1.2-8		Equipment Location Building Sections D & F
M-1G020	1.2-9		Equipment Location Reactor and Auxiliary Building Plan Basement EL. 1974'-0"
M-1G021	1.2-10		Equipment Location Auxiliary Building Partial Plan EL. 1988'-0" and 2013'-6"
M-1G022	1.2-11		Equipment Location Reactor and Auxiliary Building Plan Ground Floor Elevation 2000'-0"
M-1G023	1.2-12		Equipment Location Reactor and Auxiliary Building Plan EL. 2026'-0"
M-1G024	1.2-13		Equipment Location Reactor and Auxiliary Building Plan Operating Floor Elevation 2047'-6"

TABLE 1.7-3 (Sheet 3)

Drawing Number	Figure Number	Sheet	Title
M-1G025	1.2-14		Equipment Location Reactor and Auxiliary Building Plan Elevation 2068'-8"
M-1G026	1.2-15		Equipment Location Reactor and Auxiliary Building Section A
M-1G027	1.2-16		Equipment Location Reactor and Auxiliary Building Section B
M-1G028	1.2-17		Equipment Location Reactor and Auxiliary Building Section C
M-1G029	1.2-18		Equipment Location Reactor and Auxiliary Building Section D
M-1G030	1.2-19		Equipment Location Auxiliary Building Sections E, F & G
M-1G040	1.2-20		Equipment Location Fuel Building Plan Elevation 2000'-0", 2026'-0" and 2047'-6"
M-1G041	1.2-21		Equipment Location Fuel Building Sections A, B & C
M-1G042	1.2-22		Equipment Location Fuel Building Sections D, E & F
M-1G050	1.2-23		Equipment Location Control Building & Communication Corridor Plan Elevation 1974'-0" & 1984'-0"
M-1G051	1.2-24		Equipment Location Control and Diesel Generator Buildings & Communication Corridor Plan Elevation 2000'-0" and 2016'-0"
M-1G052	1.2-25		Equipment Location Control and Diesel Generator Buildings & Communication Corridor Plan Elevation 2032'-0" & 2047'-6"
M-1G053	1.2-26		Equipment Location Control and Diesel Generator Buildings & Communication Corridor Plan Elevation 2061'-6", 2066'-0" & 2073'-6" & Section D
M-1G054	1.2-27		Equipment Location Control and Diesel Generator Building Communication Corridor Section A
M-1G055	1.2-28		Equipment Location Control and Diesel Generator Building Sections B & C
M-1G060	1.2-29		Equipment Location Turbine Building Condenser Pit Plan Elevation 1983'-0"
M-1G061	1.2-30		Equipment Location Turbine Building Ground Floor Plan Elevation 2000'-0"

TABLE 1.7-3 (Sheet 4)

Drawing Number	Figure Number	Sheet	Title
M-1G062	1.2-31		Equipment Location Partial Plan Elevation 2015'-4"
M-1G063	1.2-32		Equipment Location Turbine Building Mezzanine Floor Plan Elevation 2033'-0"
M-1G064	1.2-33		Equipment Location Turbine Building Operating Floor Plan Elevation 2065'-0"
M-1G065	1.2-34		Equipment Location Turbine Building Section "A"
M-1G066	1.2-35		Equipment Location Turbine Building Section "B"
M-1G067	1.2-36		Equipment Location Turbine Building Section "C"
M-0G068	1.2-37		Equipment Location Turbine Building Section "D"
M-1G069	1.2-38		Equipment Location Turbine Building Section "E"
M-0G070	1.2-39		Equipment Location Turbine Building Section "F"
M-0G071	1.2-40		Equipment Location Turbine Building Section "G"
M-1G072	1.2-41		Equipment Location Turbine Building Section "H"
M-0G073	1.2-42		Turbine Component Laydown Area Elevation 2065'-0"
M-13EN03	6.2.2-2	1	Containment Spray System Reactor Building A & B Trains
M-13EN04	6.2.2-2	2	Containment Spray System Reactor Building A & B Trains
M-13EN05	6.2.2-2	3	Containment Spray System Reactor Building A & B Trains
M-KG080	9.2-3		ESWS Pumphouse Equipment Location – Plan
M-KG081	9.2-4		ESWS Pumphouse Equipment Location – Sections
M-744-00018	7.2-1	1	Functional Diagrams Index and Symbols
M-744-00019	7.2-1	2	Functional Diagrams (Reactor Trip Signals)
M-744-00020	7.2-1	3	Functional Diagrams (Nuclear Instrumentation and Manual Trip Signals)
M-744-00021	7.2-1	4	Functional Diagrams (Nuclear Instrumentation Permissives and Blocks)
M-744-00022	7.2-1	5	Functional Diagrams (Primary Coolant System Trip Signals)
M-744-00023	7.2-1	6	Functional Diagrams (Pressurizer Trip Signals)
M-744-00024	7.2-1	7	Functional Diagrams (Steam Generator Trip Signals)

TABLE 1.7-3 (Sheet 5)

Drawing Number	Figure Number	Sheet	Title
M-744-00025	7.2-1	8	Functional Diagrams (Safe-guards Activation Signals)
M-744-00026	7.2-1	9	Functional Diagrams (Rod Controls and Rod Blocks)
M-744-00027	7.2-1	10	Functional Diagrams (Steam Dump Control)
M-744-00028	7.2-1	11	Functional Diagrams (Pressur-izer Pressure & Level Control)
M-744-00029	7.2-1	12	Functional Diagrams (Pressur-izer Heater Control)
M-744-00030	7.2-1	13	Functional Diagrams (Feedwater Control and Isolation)
M-744-00031	7.2-1	14	Functional Diagrams (Feedwater Control and Isolation)
M-744-00032	7.2-1	15	Functional Diagrams (Auxiliary Feedwater Pumps Startup)
M-744-00033	7.2-1	16	Functional Diagram (Turbine Trip Runbacks and Other Signals)
M-744-00039	7.2-1	17	Functional Diagram (Pressurizer Pressure Relief)
M-744-00040	7.2-1	18	Functional Diagram (Pressurizer Pressure Relief)
SK-C-250	3B-2		Plan and Elevation View of Main Steam/Main Feedwater Isolation Valve Compartment
S-0172	2.4-3	2	Grading Plan Switchyard Area
S-0186	2.4-3	3	Drainage Plan Plant Area
S-0189-1	2.4-3	4A	Manhole, Pipe & Culvert Schedule
S-0189-2	2.4-3	4B	Manhole, Pipe & Culvert Schedule
S-0189-3	2.4-3	4C	Manhole, Pipe & Culvert Schedule
S-0189-4	2.4-3	4D	Manhole, Pipe & Culvert Schedule
S-0191	2.4-3	5	Manhole & Pipe Details
S-0296	2.4-3	6	Manhole & Pipe Details
S-0297	2.4-3	7	Plant Area Roadway Grading & Drainage

TABLE 1.7-4

ERECTED SCAFFOLD EXPECTED TO BE IN PLACE THROUGHOUT NEXT REQUIRED USAR UPDATE CYCLE

This Table has been deleted

TABLE 1.7-4 (sheet 2)

ERECTED SCAFFOLD EXPECTED TO BE IN PLACE THROUGHOUT NEXT REQUIRED USAR UPDATE CYCLE

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TABLE 1.7-4 (sheet 3)

ERECTED SCAFFOLD EXPECTED TO BE IN PLACE THROUGHOUT NEXT REQUIRED USAR UPDATE CYCLE

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TABLE 1.7-4 (sheet 4)

ERECTED SCAFFOLD EXPECTED TO BE IN PLACE THROUGHOUT NEXT REQUIRED USAR UPDATE CYCLE

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TABLE 1.7-4 (sheet 5)

ERECTED SCAFFOLD EXPECTED TO BE IN PLACE THROUGHOUT NEXT REQUIRED USAR UPDATE CYCLE

This Table has been deleted

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

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

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

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

The RRRC developed a categorization nomenclature to aid in the uniform implementation of new and revised regulatory staff concerns. The system included 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 are 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 a list of Category 2, 3, and 4 matters for consideration in preparing the WCGS FSAR. A discussion of how Wolf Creek complies with each of the listed matters is contained in Tables 1.9-1 through 1.9-4. 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*
(Historical Information)

Regulatory Guide		RRRC Category		
Number	Revision	2	3	4
1.12	2			X
1.13	1			X
1.14	1			X
1.27	2	X		
1.52	2	X		
1.56	1		X	
1.59	2	X		
1.63	2	X		
1.68.2	1		X	
1.75	2			X
1.76	0			X
1.79	1			X
1.80	0			X
1.82	0			X
1.83	1			X
1.89	1			X
1.91	1	X		
1.93	0			X
1.97	2		X	
1.99	2		X	
1.101	1,2		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	1	X		
1.137	0	X		
1.141	0	X		
8.8	2	X		

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

TABLE 1.9-2

CATEGORY 2, 3, AND 4
BRANCH TECHNICAL POSITIONS

Branch Technical Position	Title	RRRC Cat.	Remarks
ASB 9.5-1, Rev. 1	Guidelines for Fire Protection for Nuclear Power Plants	2	The recommendations of this BTP are met to the extent described in Section 9.5.1.
MTEB 5-7	Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary	2	The recommendations of this BTP are not applicable to the WCGS (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 Sections 5.2.2 and 7.6.6.
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.

TABLE 1.9-2 (Sheet 2)

Branch Technical Position	Title	RRRC Cat.	Remarks
CSB 6-3	Determination of Bypass Leakage Paths in Dual Containment Plants	4.B.4	The recommendations of BTP are not applicable to the WCGS design, since there is no dual containment.
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	
3.5.3 (Par II.1.C)	4.B.8	Procedures for Composite Section Local Damage Prediction	The recommendations of this SRP are met to the extent described in Section 3.5.3.
3.7.1 (Par. II.2)	4.B.9	Development of Design Time History for Soil-Structure Interaction Analysis	The recommendations of this SRP are met to the extent described in Section 3.7(B).1.
3.7.2 (Par. II)	4.B.10	Procedures for Seismic System Analysis	The recommendations of this SRP are met. Refer to Sections 3.7(B).2 and 3.7(N).2.
3.7.3 (Par. II)	4.B.11	Procedures for Seismic Subsystem Analysis	The recommendations of this SRP are met. Refer to Sections 3.7(B).3 and 3.7(N).3
3.8.1 (Par. II)	4.B.12	Design and Construction of Concrete Containments	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 (Par. II)	4.B.13	Design and Construction of Steel Containments	The recommendations of this SRP are not applicable to WCGS.
3.8.3 (Par. II)	4.B.14	Structural Design Criteria for Category I Structures Inside Containment	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.
3.8.4 (Par. II)	4.B.15	Structural Design Criteria for Other Seismic Category I Structures	The design meets or exceeds the load combinations of ACI 359/SRP criteria. Refer to Section 3.8.4.

TABLE 1.9-3 (Sheet 2)

SRP Section	RRRC Category	Title	Remarks
3.8.5 (Par. II)	4.B.16	Structural Design Criteria for Foundations	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 deisgn capabilities of the radwaste building.
3.3.2 (Par. II.2.d)	4.B.18	Tornado Load Effect Combinations	The recommendations of this SRP are met. Refer to Section 3.3.2.
3.4.2 (Par II)	4.B.19	Dynamic Effects of Wave Action	The recommendations of this SRP are met. Refer to Section 3.4.2.
10.4.7 (Par. I.2.b)	4.B.20	Water Hammer for Steam Generators with Preheaters	The WCGS steam qenerators (Model F) have no preheaters. Refer to Sections 5.4.2 and 10.4.7.
4.4 (Par. II.5)	4.B.21	Thermal-Hydraulic Stability	The recommendations of this SRP are met as discussed in Section 4.4.4.6.
5.2.5 (Par II.4)	4.B.22	Intersystem Leakage Detection (See RG 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 WCGS (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 WCGS (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.
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. Operator action is not assumed for 10 minutes.

TABLE 1.9-4 (Sheet 2)

SRP Section	RRRC Category	Title	Remarks
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 Technical Specifications.
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.
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.

TABLE 1.9-4 (Sheet 3)

	Category	Title	Remarks
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 was 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).