CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

TABLE OF CONTENTS

CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

TABLE OF CONTENTS (continued)

CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

LIST OF TABLES

CHAPTER 1
INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

LIST OF FIGURES

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This updated Final Safety Analysis Report (FSAR) is for the H. B. Robinson Steam Electric Plant Unit 2 (HBR 2). It is submitted to fulfill the requirements of 10CFR 50.71(e), as published in the Federal Register on May 9, 1980. The Atomic Energy Commission (AEC) issued Facility Operating License No. DPR-23 for HBR 2, dated July 31, 1970, specifying maximum power level of 2200 MWt upon completion of preliminary testing and certain other prerequisites. These prerequisites were satisfactorily met in the following months. The Docket No. is 50-261, and the Construction Permit for the unit was No. CPPR-26. The Nuclear Regulatory Commission (NRC) issued a license amendment on June 29, 1979, which authorized operation at a maximum power level of 2300 MWt. A subsequent license amendment was issued on November 5, 2002, authorizing operation at a maximum power level of 2339 MWt. The FSAR for the unit was submitted in November, 1968, and was amended a number of times prior to the issuance of the licenses. This updated FSAR has been organized in accordance with the guidelines contained in Regulatory Guide 1.70, Revision 3.

Table 1.1.0-1 lists the acronyms and Table 1.1.0-2 gives the abbreviations used throughout this updated FSAR.

HBR 2 is situated on Lake Robinson, a man-made 2250 acre lake, about 4.5 miles from Hartsville, South Carolina.

The HBR 2 reactor is a pressurized light water moderated and cooled system. The unit is designed to produce 2339 MWt. All steam and power conversion equipment, including the turbine generator, was designed to permit generation of 787 MWe (gross).

The nuclear power plant incorporates a closed-cycle pressurized water Nuclear Steam Supply System (NSSS) and a Turbine-Generator System utilizing dry and saturated steam. Equipment includes systems for the processing of radioactive wastes, handling of fuel, electrical distribution, cooling, power generation structures, and all other onsite facilities required to provide a complete and operable nuclear power plant.

TABLE 1.1.0-1

ACRONYMS

AC	air conditioning; also, alternating current
ACI	American Concrete Institute
ACS	Auxiliary Coolant System
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as is reasonably achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
APCSB	Auxiliary and Power Conversion Systems Branch
API	American Petroleum Institute
ARC	Automatic Rod Control
ASA	American Standards Association
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AVT	All Volatile Treatment
AWS	American Welding Society
AWWA	American Water Works Association
BAT	boron acid tank
B&PV	boiler and pressure vessel
BIT	boron injection tank

TABLE 1.1.0-1 (Cont'd)

BOC	beginning of cycle
BOL	beginning of life
BTP	branch technical position
BWR	boiling water reactor
CCW	component cooling water
CES	Critical Experiment Station
CFPP	Containment Fire Protection Status Panel
CFR	Code of Federal Regulations
CIS	Containment Isolation System
CMAA	Crane Manufacturers Association of America, Inc.
CNS	Corporate Nuclear Safety
CP&L	Carolina Power & Light Company
CRD	control rod drive
CRDM	control rod drive mechanism
CRDS	Control Rod Drive System
CS	carbon steel
CSS	Containment Spray System
CV	containment vessel
CVCS	Chemical and Volume Control System
CVTR	Carolina-Virginia Tube Reactor
DBA	design basis accident
DBE	design basis earthquake
DC	direct current
DECLG	double-ended, cold leg guillotine
DF	decontamination factor

TABLE 1.1.0-1 (Cont'd)

DG	diesel generator
DMIMS	Digital Metal Impact Monitoring System
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOT	Department of Transportation
DS	dedicated shutdown
DTT	Design Transition Temperature
Ebasco	Electric Bond and Share Company
ECCS	Emergency Core Cooling System
EFPH	effective full-power hours
EFPY	effective full-power years
ENC	Exxon Nuclear Corporation
EOC	end of cycle
EOL	end of life
EPRI	Electric Power Research Institute
ESF	Engineered Safety Features
FAP	Fire Alarm Status Panel
FDAP	Fire Detection Actuation Panel
FDAS	Fire Detection Actuation System
FHB	Fuel Handling Building
FI	Fully Integral Ruggedized Rotor
FIRL	Franklin Institute Research Laboratory
FM	Factory Mutual Research; also, frequency modulated
FSAR	Final Safety Analysis Report
FW	feedwater
GDC	General Design Criteria

TABLE 1.1.0-1 (Cont'd)

GM	Geiger - Mueller
GWPS	Gaseous Waste Processing System
HBR	H. B. Robinson
HEPA	high-efficiency particulate air filters
HIS	Hydraulic Institute Standards
HP	high pressure
HPSI	high pressure safety injection
HVAC	heating, ventilating, and air conditioning
IAEA	International Atomic Energy Agency
I&C	instrumentation and control
IC	internal combustion
ID	inside diameter
IEEE	Institute of Electrical and Electronic Engineers
INPO	Institute of Nuclear Power Operations
IPBS	Integrated Planning, Budgeting, and Scheduling
IPCEA	Insulated Power Cable Engineers Association
ISI	in-service inspection
IVSW	isolation valve seal water
IVSWS	Isolation Valve Seal Water System
LED	light emitting device; also, light-emitting diode
LHGR	linear heat generation rate
LMFBR	liquid metal fast breeder reactor
LOCA	loss-of-coolant accident
LP	low pressure
LPSI	low pressure safety injection
LVDT	linear variable differential transducers

TABLE 1.1.0-1 (Cont'd)

LWR	light water reactor
MCA	maximum credible accident
MCC	motor control center
MDNBR	minimum departure from nucleate boiling ratio
MG	motor generator
MM	Modified Mercalli
MOV	motor operated valve
MPC	maximum permissible concentration
MSIV	main steam isolation valve
MSL	mean sea level
MTC	moderator temperature coefficient
MWT	makup water tank
NBS	National Bureau of Standards
NDE	nondestructive examination
NDTT	nil-ductility transition temperature
NEMA	National Electrical Manufacturer's Association
NFPA	National Fire Protection Association
NIS	Nuclear Instrumentation System
NML	Nuclear Mutual Limited
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center (EPRI)
NSSS	Nuclear Steam Supply System
NWRC	National Weather Records Center
OBE	operational basis earthquake
OD	outside diameter

TABLE 1.1.0-1 (Cont'd)

ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PA	Public Address
PACV	Post-Accident Containment Venting
PBX	private branch exchange
PCI	pellet-cladding interaction
PCT	peak clad temperature
PDC II	power distribution control II
PF	power factor
PFI	Pipe Fabrication Institute
pH	Concentration of Hydrogen ions
PLA	Pickard, Lowe & Associates
PLSA	Part Length Shield Assembly
PMF	probable maximum flood
PORV	power-operated relief valve(s)
PPS	Penetration Pressurization System
PRCF	Plutonium Recycle Facility
PRT	pressurizer relief tank
PSAR	Preliminary Safety Analysis Report
PTS	Pressurized Thermal Shock
PVC	polyvinyl chloride
PWR	pressurized water reactor
QA	quality assurance
QAC	Quality Assurance Committee
QC	Quality Control
QCS	Quality Control Systems

TABLE 1.1.0-1 (Cont'd)

RAB	Reactor Auxiliary Building
RC	radiation control
RCC	rod cluster control
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCGVS	Reactor Coolant Gas Vent System
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCPS	Reactor Coolant Pump System
RCS	Reactor Coolant System
RG	Regulatory Guide
RG&E	Rochester Gas and Electric
RH	relative humidity
RHR	residual heat removal
RHRS	Residual Heat Removal System
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RPV	reactor pressure vessel
RTD	resistance temperature detectors
RTG	reactor and turbine-generator
RTGB	reactor and turbine-generator board
RT_{NDT}	Reference Temperature, Nil-Ductility Transition
RTS	Reactor Trip System
RWP	radiation work permits
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit

TABLE 1.1.0-1 (Continued)

SAR	Safety Analysis Report
SBOStation Blackout	
SCR	Silicon Control Rectifier
SFP	Spent Fuel Pit
SHNPP	Shearon Harris Nuclear Power Plant
SG	Steam Generator
SI	Safety Injection
SIS	Safety Injection System
SOR	Senior Operator License
SPCSiemens Power Corporation	
SRO	Senior Reactor Operator
SRWP	Standing Radiation Work Permit
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
SSPC	Steel Structure Painting Council
STP	Standard Temperature and Pressure
SWP	Service Water Pump
SWPS	Solid Waste Processing System

TABLE 1.1.0-1 (Cont'd)

SWRI	Southwest Research Institute
SWS	Service Water System
TD	theoretical density
TDH	total developed head
TEMA	Tubular Exchanger Manufacturers' Association
TLD	thermoluminescent dosimeters
TSC	Technical Support Center
USAEC	United States Atomic Energy Commission
USAS	United States of America Standards
USGS	United States Geological Survey
UHF	ultra high frequency
UL	Underwriter's Laboratories, Inc.
UPS	uninterruptible power supply
UT	ultrasonic test
UTM	Universal Transverse Mercator
VCT	volume control tank
VHF	very high frequency
WAPD	Westinghouse Atomic Power Division
WDS	Waste Disposal System
Westinghouse	Westinghouse Electric Corporation
WOL	wedge opening loading
WREC	Westinghouse Reactor Evaluation Center

TABLE 1.1.0-2

UNITS

acre	ac
acre-foot	ac-ft
ampere	amp
atmosphere per cubic centimeter	atm/cc
average temperature	^{T}avg
brake horsepower	bhp
British thermal units per hour	Btu/hr
British thermal units per hour/foot	Btu/hr/ft
British thermal units per hour/square foot	Btu/hr-ft^2
calorie	cal
centimeter	cm
centipoise	cp
Chi/Q	χ/Q
cubic centimeter	cm^3 or cc
cubic centimeters per hour	cc/hr
cubic centimeters per minute	cc/min
cubic feet per minute	cfm or ft^3/min
cubic feet per second	ft^3/sec or cfs
cubic foot	ft^3
cubic meter	m^3
cubic yard	yd^3
curie	Ci
curie per cubic centimeter	Ci/cc
curie per second	Ci/sec

TABLE 1.1.0-2 (Cont'd)

curie per year	Ci/yr
cycles per second	cps or Hz
degrees centigrade	°C
degrees Farenheit	°F
degrees Kelvin	°K
direct current	DC
disintegrations per second per milligram	DPS/mg
electron volt	ev
feet (foot)	ft
feet per second	fps
foot-inch	ft-in.
foot-pound	ft-lb
gallon	gal
gallons per day	gal/day or gpd
gallons per hour	gal/hr
gallons per minute	gal/min or gpm
gram	g
gram mole per degree Rankine	g mole/°R
gram per cubic centimeter	g/cc or g/cm^3
Hertz	hz
horsepower	HP
hour	hr
inch (inches)	in.
inch water gage	in. wg or wg
inside diameter	ID
kiloelectron volt	Kev

TABLE 1.1.0-2 (Cont'd)

kilogram	kg
kilovolt	kV
kilovolt-ampere	kVa
kilowatt	kW
kilowatt per foot	kW/ft
megawatt	MW
megawatt days per metric ton of uranium	MWD/MTU
megawatt (electric)	Mwe
megawatt (thermal)	Mwt
mercury	Hg
meter	m
micro curie per cubic centimeter	μCi/cc
micro mho per centimeter (conductivity)	μmho/cm
micron, micro	μ
mile per hour	mph
mile per second	mps
millicurie	mCi
milligram per square decimeter	mg/dm^2
milliliter	ml
million electron volts	Mev
millirem per hour	mrem/hr
milliroentgen per hour	mR/hr
millivolt	mv
minute or minimum	min

TABLE 1.1.0-2 (Cont'd)

neutron multiplication factor, effective	K_{eff}
neutron multiplication factor, infinity	$K\infty$
neutrons per square centimeter-second (nv)	n/cm^2-sec
ohm-centimeter (Resistivity)	ohm-cm
parts per billion	ppb
parts per million	ppm
per second	sec^{-1}
pound	lb
pound mass	lbm
pounds per cubic foot	lb/ft^3 or pcf
pounds per hour	lb/hr
pounds per square foot	lb/ft^2 or psf
pounds per square inch	psi
pounds per square inch (absolute)	psia
pounds per square inch (differential)	psid
pounds per square inch (gage)	psig
radiation absorption dose	rad
radius	r
reactivity	$\Delta k/k$
reactivity change rate	Δk/sec
revolutions per minute	rpm
revolutions per second	rps
roentgen	R
roentgen equivalent man	rem
roentgens per hour	R/hr
root mean square	rms

TABLE 1.1.0-2 (Cont'd)

second	sec
specific gravity	sp gr
square	($)^2$ or sq
square centimeter	cm^2
square foot	ft^2 or sq ft
square inch	$in.^2$ or sq in.
square mile	mi^2 or sq mi
standard cubic feet	scf or stdft
standard cubic feet per minute	scfm or $stdft^3/min$
standard cubic feet per second	scfs
temperature of the cold leg	T_{cold}
temperature of the hot leg	T_{hot}
thickness	T
thousand pounds	kip
thousand pounds per linear foot	k/lf
thousand pounds per square inch	ksi
thousandth of an inch	mil
ton (short ton)	ton, st
tonne (metric ton, 2,204.62 lb)	te, mt
volt	V
volt alternating current	V AC
volt ampere	Va
volt direct current	V DC
volume percent	vol %
water column	wc
watt	W

TABLE 1.1.0-2 (Cont'd)

watt per cubic centimeter	W/cc
week	wk
weight percent	wt %
yard	yd
year	yr

1.2 GENERAL PLANT DESCRIPTION

1.2.1 SITE AND ENVIRONMENT

The site is in northeastern South Carolina, 56 miles ENE of Columbia, the state capital. The location is about 25 miles NW of Florence, and about 35 miles NNE of Sumter, S. C. Coordinates of the site are latitude 34° 24.2' N and longitude 80° 09.5' W. It is located on the southwestern corner of Lake Robinson which was impounded to furnish cooling water for power plants at the site. The exclusion distance and low population distances are 1400 ft and 4.5 miles respectively. Exclusion distance is the distance from the reactor to the closest point on the boundary of the exclusion area defined in 10CFR100. The low population distance is the distance from the reactor to the boundary of the low population zone defined in 10CFR100. The total site area including Lake Robinson is more than 5,000 acres. Farming is the predominant activity in the sparsely populated immediate environs of the plant site. The site surface soil is sandy and surface water drains to the lake. The region is gently rolling and is not subject to severe persistent inversions. Tornadoes occur in the region but have not affected the site. While many hurricanes affect southeastern United States, no hurricane storm tracks were reported in the near vicinity of the site during the period between 1900 and the beginning of commercial operation of the plant. However, hurricane Hugo passed within 40 miles of the plant site in September, 1989, and no weather stations within 50 miles of the plant reported sustained hurricane force winds associated with Hugo.

1.2.2 SUMMARY PLANT DESCRIPTION

The inherent design of the pressurized water, closed-cycle reactor significantly reduces the quantities of fission products which must be released to the atmosphere. Four barriers exist between the fission products accumulated and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Fission products, which escape through a fuel cladding defect are contained within the pressure vessel, loops and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to retain adequately these fission products under the most severe accident conditions, as analyzed in Chapter 15.0.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss-of-coolant accident (LOCA). These safety features include a Safety Injection System. This system automatically delivers borated water to the reactor vessel for cooling under high and low reactor coolant pressure conditions. The Safety Injection System also serves to insert negative reactivity into the core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a Containment Air Recirculation Cooling System which would effect a rapid depressurization of the containment following a loss of coolant, and a Containment Spray System which would depressurize the containment and remove elemental iodine from the atmosphere by washing action. The Containment Spray System provides backup cooling for the Containment Air Recirculation Cooling System.

The Radwaste Facility was designed as a separate and complete structure from the remainder of the H. B. Robinson facility, and as such reflects current design philosophy that is equal or superior to previous design practices. To facilitate design of the Radwaste Facility, a design basis document was developed.

1.2.2.1 Structures

The major structures are a Reactor Containment, Auxiliary Building, Turbine Building, Radwaste Facility and Fuel Handling Building. A general plan of the building arrangements is shown on Figure 1.2.2-1. Figures 1.2.2-2 through 1.2.2-12 show the general internal layout of the buildings.

The reactor containment is a vertical, reinforced concrete cylinder with prestressed tendons in the vertical wall, a reinforced concrete hemispherical domed roof and a substantial base slab of reinforced concrete supported by piles. The containment was designed to withstand environmental effects and the internal pressure accompanying a LOCA. It also provides adequate radiation shielding for both normal operation and accident conditions.

Particular structures and equipment are classified according to seismic design. The definition of the three seismic classifications is given in Section 3.2.

1.2.2.2 Nuclear Steam Supply System (NSSS)

The NSSS consists of a pressurized water reactor, Reactor Coolant System (RCS), and associated auxiliary fluid systems. The RCS is arranged as three closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The control rods consist of clusters of stainless steel clad absorber rods and guide tubes located within the fuel assembly.

The steam generators are vertical U-tube units containing Inconel tubes. Integral separating equipment reduces the moisture content of the steam at the turbine throttle to 1/4 percent or less.

The reactor coolant pumps are vertical, single stage, certrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to charge the RCS and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shutdown, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, and vent and drain the RCS.

1.2.2.3 Reactor and Plant Control

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The automatic rod control system is designed to restore programmed average temperature following a scheduled or transient step change in load of 10 percent or a ramp change of 5 percent per minute within the range of 15 to 100 percent power. However, under nominal operating conditions, the control system allows the plant to accept step load changes of 19 percent and ramp load changes of 14 percent per minute over the load range of 15 to 95 percent power.

Supervision of both the reactor and turbine generator is accomplished from the Control Room.

The waste disposal control board is located in the Auxiliary Building, in the vicinity of the boric acid and waste evaporators. This board permits the auxiliary operator to control and monitor the processing of wastes from a central location in the same general area where equipment is located.

1.2.2.4 Waste Disposal System

The Waste Disposal System provides equipment necessary to collect, process, and prepare for disposal potentially radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation.

Liquid wastes are collected and processed through a waste evaporator, or a filter/demineralizer system. Mop water and decontamination solutions may also be processed using charcoal to meet NPDES limits. The effluent from the waste evaporator, the filter/demineralizer system, or the mop water treatment process is sampled to determine residual activity and monitored during discharge to the lake via the condenser discharge to assure concentrations are below 10CFR20 limits. The evaporator residues are drummed for ultimate disposal in an authorized location. Filters, spent resin and charcoal are also shipped for disposal in an authorized location.

Gaseous wastes are collected and stored in Waste Gas Decay Tanks until release or they are discharged to the environment in a manner that does not create radioactivity concentrations above 10CFR20 limits.

1.2.2.5 Fuel Handling System

The reactor is refueled with equipment designed to handle spent fuel under water. 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 storage of new fuel.

1.2.2.6 Turbine and Auxiliaries

The turbine is a tandem-compound, 3-element, 1,800 rpm unit having 56 in. exhaust blading in the low pressure elements. Four combination moisture separator-reheater units are employed to dry and superheat the steam between the high and low pressure turbine cylinders.

A single-pass deaerating, radial flow surface condenser, vacuum pump air ejector, two 55 percent capacity condensate pumps, two 55 percent capacity motor-driven boiler feed pumps, and six stages of feedwater heaters are provided. Two auxiliary (motor-driven) feedwater pumps are provided, in addition to an auxiliary feedwater pump which is steam driven. The steam-driven pump may be used in the unlikely event that power to both motor-driven pumps is interrupted.

1.2.2.7 Electrical System

The main generator is an 1,800 rpm, 3 phase, 60 cycle, hydrogen innercooled unit. Three single phase main step-up transformers deliver power to the 230 kV switchyard.

The Station Service System consists of auxiliary transformers, 4l60 V switchgear, 480 V switchgear and motor control centers, 120 Vac Panels and 125 V DC equipment.

Emergency power supplied by alternate sources including emergency diesel generators provides power required for safe shutdown of the unit and for operating post-accident containment cooling equipment, as well as for both high head and low head safety injection pumps to ensure an acceptable post-loss-of-coolant containment pressure transient.

1.2.2.8 Engineered Safety Features Protection Systems

The Engineered Safety Features Protection Systems provided for this unit have sufficient redundancy of component and power sources that, under the conditions of a hypothetical LOCA, the system can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure of the public below the limits of 10CFR100.

The systems provided are summarized below:

a) The Containment System provides a highly reliable, essentially leak-tight barrier against the escape of fission products. The containment vessel penetrations are tested in accordance with 10 CFR 50, Appendix J. Pipes penetrating the containment which could become potential paths for leakage to the environment following a LOCA are provided with Isolation Valve Seal Water System (IVSW) connections. The system provides a simple and reliable means for injection of seal water between the seats and stem packing of the closed globe and double disc types of isolation valves, and into the piping between closed diaphragm type isolation valves. The operation of the system can be monitored after the accident, and provisions are included for manually replenishing the seal water if required. These provisions minimize leakage to the environment.

b) The Safety Injection System provides borated water to insert negative reactivity and cool the core by injection into the cold and hot legs of the reactor coolant loops.

c) The Containment Air Recirculation Cooling System provides a dynamic heat sink to cool the containment atmosphere under the conditions of a LOCA. The system utilizes the normal containment ventilation and cooling equipment.

d) The Containment Spray System provides a spray of cool, chemically treated borated water to the containment atmosphere to provide iodine removal capability and backup to the Containment Air Recirculation Cooling System.

1.2.2.9 Independent Spent Fuel Storage Installation (ISFSI)

There are two separate dry fuel storage facilities for the HBR2 site. The first facility was licensed in 1986 and is designated the 7P-ISFSI, as each canister contains 7 fuel assemblies. The second facility began operation in 2005 and is designated the 24P-ISFSI, as each canister contains 24 fuel assemblies. Both facilities employ the NUHOMS storage system. The fuel assemblies are confined in a helium atmosphere by a stainless steel canister. The canister is protected and shielded by a concrete horizontal storage module. Decay heat is removed by thermal radiation, conduction, and convection from the canister to an air plenum inside the concrete module. Air flows through this internal plenum by natural draft convection.

The canister containing irradiated fuel assemblies is transferred from the spent fuel pool to the concrete module in a transfer cask.

The 7P-ISFSI is operated under site-specific Materials License No. SNM-2502 and has its own Safety Analysis Report and Technical Specifications. The 24PISFSI is operated under the general license provisions of 10 CFR 72 and must meet the requirements for Certificate of Compliance 1004 for the NUHOMS 24-PTH system.

1.3 COMPARISON TABLES

Section 1.3 information was deleted in Revision 20.

1.4 IDENTIFICATION OF CONTRACTORS

The Carolina Power & Light Company (CP&L), as owner, engaged or approved the engagement of, the contractors and consultants identified below in connection with the design and construction of HBR Unit 2. However, irrespective of the explanation of contractual arrangements offered below, CP&L was the sole applicant for the construction permit and is the operating licensee, and as owner and licensee is responsible for the design, construction, and operation of the Unit.

HBR 2 was designed and built by the Westinghouse Electric Corporation as prime contractor for CP&L. Westinghouse undertook to provide a complete, safe, and operable nuclear power plant ready for commercial service. The project was directed by Westinghouse from the offices of its Atomic Power Division in Pittsburgh, Pennsylvania, and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse engaged the engineering firm of Electric Bond and Share Company (Ebasco) Services Incorporated of New York City, New York, to provide the design of the structures and non-nuclear portions of the plant to prepare specifications for the purchase and construction thereof. CP&L reviewed the designs and specifications prepared by Westinghouse and Ebasco Services to assure that the general plant arrangements, equipment, and operating provisions were satisfactory to them. CP&L inspected the construction work to assure that the plant was built in accordance with the approved plans and specifications.

The plant was constructed under the general direction of Westinghouse through a general contractor who was responsible for the management of all site construction activities and who either performed or subcontracted the work of construction and equipment erection. Preoperational testing of equipment and systems and initial plant operation was performed by CP&L personnel under the technical direction of Westinghouse.

As consultants on studies of plant site geology, hydrology, and seismology, the firm of Dames and Moore of New York, New York, was engaged by CP&L to work in conjunction with Ebasco Services.

As additional consultants on seismology and geology, Dr. G. W. Housner of the California Institute of Technology; Dr. J. L. Stuckey and Dr. L. L. Smith, Consultants, Raleigh, North Carolina; and Dr. P. Byerly, Consultant, Oakland, California, were engaged by CP&L.

As consultants on reactor and plant engineering, site meteorology, and general site studies, the firm of Pickard, Lowe, and Associates of Washington, D. C. was engaged by CP&L.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

Research and development (as defined in Section 50.2 of the commission's regulations) was conducted regarding final core design details and parameters (no longer applicable due to refueling changes), analytical methods for kinetics calculation, safety injection (emergency core cooling) system, xenon stability and related control systems, containment spray additive effectiveness, and capability of reactor internals to resist blowdown forces. The information submitted in Supplement No. 1, Tab 11 Item 1 of the original FSAR clarified the status of Westinghouse safety-related research and development programs as they related to the HBR 2. That information is now obsolete and not included in the updated FSAR.

1.5.1 DELETED BY AMENDMENT NO. 11.

1.5.2 SAFETY INJECTION SYSTEM DESIGN

The design of the safety injection system is essentially that proposed at the time the construction permit was issued; that is, it includesnitrogen-pressurized accumulators to inject borated water into the RCS to rapidly and reliably reflood the core following a loss-of-coolant accident (LOCA). Additional analyses have been performed to demonstrate that the accumulators in conjunction with other components of the emergency core cooling system can adequately cool the core for any pipe rupture.

Research and development work has also been performed on the integrity of Zircaloy-clad fuel under conditions simulating those during a LOCA. Under the conservatively evaluated temperature predicted for the fuel rods during LOCA, the clad may burst due to a combination of fuel rod internal gas pressure and the reduction of clad strength with temperature. Burst cladding could block flow channels in the core, so that core cooling by the safety injection system would be insufficient to prevent fuel rod melting.

Rod burst experiments have therefore been conducted on Zircaloy rods to evaluate the mechanism and effects of this potential channel blockage. Single rod tests are presented in detail in Reference 1.5.2-1.

1.5.3 SYSTEMS FOR REACTOR CONTROL DURING XENON INSTABILITIES

In the transition to large Zircaloy-clad-fuel cores, the potential of power spatial redistribution caused by instabilities in local xenon concentration was created.

Extensive analytical work has been performed on reactor core stability (References 1.5.3-1, 1.5.3-2, and 1.5.3-3). These references indicate that a core of this size may be unstable against axial power redistribution, but is nominally stable against transverse (denoted X-Y) power oscillations. The plant is therefore provided with instrumentation which will allow the operator to detect the axial power oscillations, and procedures exist for suppressing these oscillations (Section 7.7.1).

Control information for suppression of power oscillation is obtained from four long ion chambers, each divided into an upper and lower section, mounted vertically outside the core. Both calculation and experimental measurements at SENA, San Onofre and Haddam Neck have shown that this out-of-core instrumentation represents in core power distribution, which is adequate for power distribution control (Reference 1.5.3-3).

The control strategy is based on the difference in output between the top and bottom sections of the long ion chambers. If the operator allows axial power imbalance to exceed operating limits, various levels of protection are invoked automatically. These include generation of alarms, turbine power cutback and blocking of control rod withdrawal.

1.5.4 CONTAINMENT SPRAY ADDITIVE FOR IODINE REMOVAL

Initially sodium thiosulphate, $Na_2S_2O_3$, was proposed as the iodine removal additive to the boric acid containment spray, but an evaluation program led to the selection of sodium hydroxide, NaOH. The results of the evaluation program are detailed in Reference 1.5.4-1 and are summarized briefly below:

a. Chemical characteristics - The $Na_2S_2O_3$ solution was found to be oxidized by air at the post-accident temperature in containment. The NaOH was not unstable in this way.

b. Iodine removal characteristics - The removal efficiency of the NaOH solution (at pH not less than 9.5) was comparable to that of the $Na_2S_2O_3$ solution.

c. Materials compatibility - Corrosion rates of copper and copper-alloy heat exchanger tubing were reduced by more than an order of magnitude compared with high pH $Na_2S_2O_3$ solution and were acceptably low (<0.01 mils/month at 200°F) for the application. These tests showed that pitting or local corrosion did not occur.

d. Radiolysis - The NaOH solution was radiolytically stable, and liberates significantly less net hydrogen than the unstable $Na_2S_2O_3$ solution.

Therefore, further testing was centered on the use of NaOH as the spray additive leading to the development of a technical basis for its inclusion in the plant engineered safety features as a means of "fixing" absorbed iodine, enhancing the natural rate of deposition of I_2, and thus lowering the calculated offsite thyroid dose resulting from a postulated release of fission products to the containment atmosphere.

Chapter 6.0 gives a further discussion of iodine removal by the Containment Spray System.

1.5.5 BLOWDOWN CAPABILITY OF REACTOR INTERNALS

As documented in the response to AEC Questions dated 3/24/69, the computer code BLODWN-2 was used to evaluate the blowdown capability of the reactor internals. In the response to AEC Questions dated 8/22/69, it was indicated that the BLODWN-2 analysis performed for Indian Point was applicable to Robinson. Detailed data from that evaluation was included in the response to AEC Questions which indicates that the "Blowdown Capability of Reactor Internals" was evaluated with BLODWN-2.

Furthermore, in June 1980, the response of the reactor vessel (including the internals) to a guillotine break at the reactor vessel inlet nozzle was evaluated by a different set of computer codes in WCAP-9748 (Westinghouse Owners Group Asymmetric LOCA Loads Evaluation). Subsequent review by the NRC determined that the probability of failure in the primary system is low enough that a double ended guillotine break need not be postulated as a design basis event for defining structural loads (reference Generic Letter 84-04).

The response of Siemens fuel assemblies to LOCA blowdown forces was evaluated in XN-NF-76-47 (Combined Seismic-LOCA Mechanical Evaluation for Exxon Nuclear 15x15 Reload Fuel). This evaluation is cited in UFSAR Section 4.0.

1.5.6 PROGRAMS CONDUCTED DURING OPERATION

There are no ongoing programs specific to HBR 2 to demonstrate the acceptability of contemplated future changes in design or modes of operation.

REFERENCES: SECTION 1.5

1.5.2-1 WCAP-7379-L, Performance of Zircaloy Clad Fuel Rods During a Single Rod Test, October 6, 1969. (Westinghouse Proprietary).

1.5.3-1 Poncelet, C. G. and Christie, A. M., "Xenon Induced Spatial Instabilities in Large Pressurized Water Reactors," WCAP-3680-20, March 1968.

1.5.3-2 McGough, J. D., "The Effect of Xenon Spatial Variations and the Moderator Coefficient Core Stability," WCAP-2983, August 1966.

1.5.3-3 Westinghouse Proprietary Report, "Power Distribution Control in Westinghouse PWR," WCAP-720. OCTOBER 1968.

1.5.4-1 Westinghouse Confidential Report, "Investigation of Chemical Additives for Reactor Containment Sprays," WCAP-7153, March 1968.

1.6 MATERIAL INCORPORATED BY REFERENCE

Topical Reports and other material incorporated by reference are called out in individual subsections and listed in the section reference lists.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The "drawing package" is not part of the original FSAR. This section is not applicable to the Updated FSAR.

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Regulatory Guides (originally called Safety Guides) have been published beginning in late 1970. Since H. B. Robinson (HBR) was licensed for operation prior to that time, they were not addressed. Those Regulatory Guides which have been addressed during the operating phase are discussed below.

Regulatory Guide 1.8	PERSONNEL SELECTION AND TRAINING (SEPTEMBER 1975)
ANSI Standard N18.1-1971	PERSONNEL SELECTION AND TRAINING

The criteria for selection and training of personnel for operation of Robinson 2 are addressed in the Robinson Technical Specifications.

Regulatory Guide 1.13 FUEL STORAGE FACILITY DESIGN BASIS

Conform to Regulatory Guide 1.13 only as it relates to the new high-density spent fuel storage racks design for preventing damage resulting from the safe shutdown earthquake (SSE), and protecting the fuel from mechanical damage.

Regulatory Guide 1.28 QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION)

Conformance with Regulatory Guide 1.28 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.29 SEISMIC DESIGN CLASSIFICATION

Comply with Regulatory Guide 1.29, Revision 3, only as it relates to the new high-density spent fuel storage racks designated Seismic Category I as defined and outlined in the regulatory guide.

Regulatory Guide 1.30 — QUALITY ASSURANCE REQUIREMENTS FOR THE INSTALLATION, INSPECTION, AND TESTING OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Conformance with Regulatory Guide 1.30 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.33 QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATION)

Conformance with Regulatory Guide 1.33 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.37 QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.37 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.38 QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.38 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17

Regulatory Guide 1.39 HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.39 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.52	DESIGN, TESTING, AND MAINTENANCE CRITERIA FOR POST ACCIDENT ENGINEERED SAFETY - FEATURE ATMOSHPERE CLEANUP SYSTEM AIR FILTRATION AND ADSORPTION UNITS OF LIGHT-WATER-COOLED NUCLEAR POWER PLANTS (MARCH 1978)
ANSI Standard N510-1975/1980	TESTING OF NUCLEAR AIR-CLEANING SYSTEMS

The RNP Ventilation Filter Testing Program, as described in Section 5.5.11 of the Improved Technical Specification, shall comply with the frequencies specified in Positions C.5 and C.6 of Regulatory Guide 1.52 (March 1978) and conducted in general conformance with ANSI Standard N510-1975 or N510-1980. Applicable portions of the Fuel Handling Building Ventilation System and Containment Purge System shall be tested in accordance with ANSI Standard N510-1975. Applicable portions of the Control Room Emergency Filtration System shall be tested in accordance with ANSI Standard N510-1980.

Regulatory Guide 1.54	QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS (JUNE 1973)
ANSI Standard N101.4-1972	QUALITY ASSURANCE FOR PROTECTIVE COATINGS APPLIED TO NUCLEAR FACILITIES

HBR2 is not committed to compliance with RG 1.54 and ANSI 101.4.

The applicable surfaces at Robinson 2 are recoated with original type coating or approved equal in accordance with original specification requirements, or equivalent engineering requirements established for touch-up and repair.

Regulatory Guide 1.58 QUALIFICATION OF NUCLEAR POWER PLANT INSPECTION, EXAMINATION, AND TESTING PERSONNEL

Conformance with Regulatory Guide 1.58 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.60 — DESIGN RESPONSE SPECTRA FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Comply with Regulatory Guide 1.60 only as it relates to design spectra for the new high-density spent fuel storage racks.

Regulatory Guide 1.61 DAMPING VALUES FOR SEISMIC DESIGN OF NUCLEAR POWER PLANTS

Comply with Regulatory Guide 1.61 only as related to damping factors for the new high-density spent fuel storage racks.

Regulatory Guide 1.64 QUALITY ASSURANCE REQUIREMENTS FOR THE DESIGN OF NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.64 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.74 QUALITY ASSURANCE TERMS AND DEFINITIONS

Conformance with Regulatory Guide 1.74 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.75 PHYSICAL INDEPENDENCE OF ELECTRICAL SYSTEMS.

See Section 7.4.2.1 for application of Regulatory Guide 1.75.

Regulatory Guide 1.88 — REQUIREMENTS FOR COLLECTION, STORAGE, AND MAINTENANCE OF QUALITY ASSURANCE RECORDS FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.88 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.92 COMBINING MODAL RESPONSES AND SPATIAL COMPONENTS IN SEISMIC RESPONSE ANALYSIS

Guidelines set forth in Regulatory Guide 1.92 were used only as they related to the new high-density fuel storage rack seismic analysis.

Regulatory Guide 1.94 — QUALITY ASSURANCE REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF STRUCTURAL CONCRETE AND STRUCTURAL STEEL DURING THE CONSTRUCTION PHASE OF NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.94 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.97 INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT CONDITIONS DURING AND FOLLOWING AN ACCIDENT (REV 3)

The guidelines for selecting variables to be monitored were established in accordance with Regulatory Guide 1.97 (Revision 3) and NUREG 0737
Supplement 1.

The compliance with these documents is detailed in the following correspondence:

CP&L Letter NLS-84-509, E. E. Utley (CP&L) to S. A. Varga (NRC), "Response to Order Confirming Commitments on Emergency Response Facility," dated December 31, 1984.

CP&L Letter NLS-85-198, S. R. Zimmerman (CP&L) to S. A. Varga (NRC), "Revision to Compliance Report for Regulatory Guide 1.97, Revision 3," dated July 18, 1985.

CP&L Letter NLS-86-267, A. B. Cutter (CP&L) to S. A. Varga (NRC), "Revision 2 to Regulatory Guide 1.97 Submittal," dated July 28, 1986.

CP&L Letter NLS-87-093, A. B. Cutter (CP&L) to S. A. Varga (NRC), "Change Reactor Coolant Pump Seal Return Flow to Type D3 Variable," dated May 1, 1987.

CP&L Letter NLS-87-136, S. R. Zimmerman (CP&L) to S. A. Varga (NRC), "Revision 4 to Regulatory Guide 1.97 Submittal," dated October 9, 1987.

CP&L Letter NLS-87-065, A. B. Cutter to NRC, "CCW Temperature Instrumentation - R.G. 1.97," dated March 27, 1987.

NRC Letter, G. Requa (NRC) to E. E. Utley (CP&L) "Request for Information - R.G. 1.97," March 5, 1987.

NRC Letter, K. T. Eccleston (NRC) to E. E. Utley (CP&L), "Regulatory Guide 1.97," April 29, 1987.

CP&L Letter RNP-RA/01-0164, B. L. Fletcher III to NRC, “Request for Technical Specification Change to Eliminate Requirements for the Post-Accident Sampling System,” dated October 31, 2001.

NRC Letter, A. G. Hansen (NRC) to J. W. Moyer (CP&L), “H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of an Amendment RE: Elimination of Requirements for the Post-Accident Sampling System (TAC No. MB3380),” dated January 14, 2002

Regulatory Guide 1.116 QA REQUIREMENTS FOR INSTALLATION, INSPECTION, AND TESTING OF MECHANICAL EQUIPMENT AND SYSTEMS

Conformance with Regulatory Guide 1.116 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.123 QUALITY ASSURANCE REQUIREMENTS FOR CONTROL OR PROCUREMENT OF ITEMS AND SERVICES FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.123 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.124 — DESIGN LIMITS AND LOADING COMBINATIONS FOR CLASS I LINEAR-TYPE COMPONENT SUPPORTS

DEP complied with Regulatory Guide 1.124 only as it related to the design of the rack component supports for the new high-density fuel storage racks.

Regulatory Guide 1.133 LOOSE PARTS DETECTION PROGRAM FOR THE PRIMARY SYSTEM OF LIGHT-WATER-COOLED REACTORS (REVISION 1)

RG 1.133 Revision 1 Section D states "In cases where licensees of operating reactors (licensed prior to January 1, 1978) have not previously committed to install a loose-part detection system or where the design of an existing system precludes upgrading to an effective functional capability, the licensee should install a system in conformance with the programmatic aspects of the guide, specifically Sections C.2 and C.3, or propose an acceptable alternative. In cases where a loose part is known to be present or there exists a high probability that a part may become loose based on experience with other reactors of similar design, a loose-part detection system conforming to this guide should be installed."

The Loose Parts Detection System at HBR 2 is in compliance with the regulatory position C.2 and C.3 with the following exceptions:

Portions of the functional tests are performed every quarter instead of the recommended monthly frequency of Section C.3.a.(2).(d). Portions of the background checks are performed every six months instead of the recommended quarterly frequency of Section C.3.a.(2).(e). Special reports to the NRC are not submitted to setpoint changes as recommended in Section C.3.a.(2).(a).

Regulatory Guide 1.137 | FUEL OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (REVISION 1)

HBR 2 will comply with Regulatory Guide 1.137 with the following exceptions:

Regulatory Position C.1 is not applicable to HBR 2 per NRC's letter dated January 13, 1978 which distributed Regulatory Guide 1.137 and provided additional guidance regarding NRC's implementation of this guide for all nuclear power plants. Position C.1 was to be evaluated on a case by case basis for application to all construction permit cases under review whose Safety Evaluation Report had not been issued as of November 1, 1979. Since HBR 2 had an operating license as of this date, Position C.1 is <u>not</u> applicable.

Regulatory Position C.2 is applicable to HBR 2 per NRC's letter dated January 13, 1978 except as follows:

A. The analyses performed will be limited to API or specific gravity, water and sediment, viscosity and cloud point. The specifications that will be met will be in accordance with the Diesel Fuel Oil Testing Program (Improved Standard Technical Specifications Bases) which also are in accordance with the emergency diesel generator manufacturer's recommendations.

B. The Unit No. 2 diesel fuel oil storage tank is filled from site storage tanks, the sampling frequency will be as described below:

    1. New fuel oil received for storage in the fuel oil storage tanks and subsequently transferred to the Unit 2 DG fuel oil storage tank is verified to meet the analysis limits of the Diesel Fuel Oil Testing Program prior to adding to the storage tanks. This is accomplished either by verifying the integrity of the seal(s) on the tank truck against the certificate of compliance or by testing of the fuel oil prior to transfer from the tank truck.

    2. Stored fuel in the storage tanks and in the Unit 2 DG fuel oil storage tank is sampled every 31 days in accordance with the Diesel Fuel Oil Testing Program.

Diesel Generator (DG) fuel oil is controlled under the QA Program by virtue of the procedures for testing of DG fuel oil being incorporated in the Plant Operating Manual which is part of the approved QA Program.

Fuel Oil sampling from the discharge of the transfer pump was compared to the methodology in ASTM D270-1975 and verified to be an equivalent sampling methodology.

The above position is based on the following references:

(1) NRC letter, Robert B. Minogue (NRC) to Regulatory Guide Distribution List (Division 1), regarding Regulatory Guide 1.137 dated January 13, 1978.

(2) NRC letter, D. G. Eisenhut (NRC) to All Power Reactor Licenses, January 7, 1980, Incoming Document No. NLU-80-48.

(3) CP&L letter, NO-80-725, M. A. McDuffie (CP&L) to D. A. Eisenhut (NRC), "Quality Assurance Requirements for Diesel Generator Fuel Oil", May 14, 1980.

Regulatory Guide 1.137 FUEL OIL SYSTEMS FOR STANDBY DIESEL GENERATORS (REVISION 1)

(4) NRC letter, Steven A. Varga (NRC) to J. A. Jones (CP&L), September 30, 1981, Incoming Document No. NLU-81-482.

(5) CP&L letter, NO-81-1914, November 20, 1981, S. R. Zimmerman (CP&L) to S. A. Varga (NRC), "Quality Assurance Requirements Regarding Diesel Generator Fuel Oil"

(6) NRC letter, S. A. Varga (NRC) to J. A. Jones (CP&L), December 10, 1981, Incoming Document No. NLU-81-607.

(7) CP&L Letter, NO-92-1404, Charles R. Dietz (CP&L) to NRC, regarding Fuel Oil Sampling methodologies, May 15, 1992.

(8) NRC Inspection Report No. 50-261/92-32, December 28, 1992.

Regulatory Guide 1.144 AUDITING OF QUALITY ASSURANCE PROGRAMS FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.144 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.145 — ATMOSPHERIC DISPERSION MODELS FOR POTENTIAL ACCIDENT CONSEQUENCE ASSESSMENTS OF NUCLEAR POWER PLANTS, (RE-ISSUE FEBRUARY, 1983 TO CORRECT PAGE 1.145-7 IN REVISION 1, NOVEMBER 1982)

HBR-2 complies with the provisions of Regulatory Guide 1.145, February 1983, for the analysis of offsite meteorology conditions that support the Updated FSAR Chapter 15 Accident Analyses performed using the Alternative Source Term (AST) dose methodology described in Regulatory Guide 1.183. Compliance with the specific details contained within this Regulatory Guide is limited to the analyses as described in the applicable DEP licensing submittals for AST implementation.

Regulatory Guide 1.146 QUALIFICATION OF QA PROGRAM AUDIT PERSONNEL FOR NUCLEAR POWER PLANTS

Conformance with Regulatory Guide 1.146 is addressed in the description of the Quality Assurance Program incorporated by reference in Chapter 17.

Regulatory Guide 1.155 STATION BLACKOUT

HBR2 complies with the intent of NRC Regulatory Guide 1.155. In developing the Station Blackout (SBO) Coping Analysis (Document 8S19-P-101), the guidance of NUMARC 87-00 has been applied. The NUMARC 87-00 methodology has been utilized, with specific exceptions in areas including: (1) evaluation of the effects of loss of ventilation, and (2) evaluation of the containment isolation capability. The analytical method applied and the results of these analyses are documented in SBO Coping Analysis 8S19-P-101.

Regulatory Guide 1.183 — ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS, JULY 2000

HBR-2 complies with the provisions of Regulatory Guide 1.183, July 2000, for selected Updated FSAR Chapter 15 Accident Analyses. Compliance with the specific details contained within this Regulatory Guide is limited to the following bounding design basis accidents, as described in the applicable DEP licensing submittals.

FSAR Section References:

1. Common Dose Consequence Inputs for Alternative Source Term (AST) Analyses (UFSAR Section 15.0.12)
2. Main Steamline Break Event (UFSAR Section 15.1.5)
3. Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR Section 15.3.2)
4. Withdrawal of a Single Full-Length RCCA (UFSAR Section 15.4.3.1)
5. Steam Generator Tube Rupture (UFSAR Section 15.6.3)
6. Design Basis Fuel Handling Accidents (UFSAR Section 15.7.4)

Regulatory Guide 4.15 QUALITY ASSURANCE FOR RADIOLOGICAL MONITORING PROGRAMS – EFFLUENT STREAMS AND THE ENVIRONMENT

HBR 2 is not committed to Regulatory Guide 4.15. The guidance within Reg. Guide 4.15 was used as a source of information to aid in developing and maintaining quality assurance for the Radiological Environmental Monitoring Program. This program is implemented by procedures as required by the Robinson Technical Specifications.