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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

This Updated Safety Analysis Report (USAR) for Unit 1 of the Perry Nuclear Power Plant is submitted as an update to the Final Safety Analysis Report. It has been prepared in accordance with <10 CFR 50.71(e)> and is current to within six months for Unit 1, common facilities and those Unit 2 facilities that are required to support Unit 1 operations. Regarding Unit 2, by way of a letter (PY-CEI/NRR-1845L) to the Nuclear Regulatory Commission on August 12, 1994, the Cleveland Electric Illuminating Company formally withdrew its request for extension of the Construction Permit for Unit 2. A Site Stabilization Plan was subsequently transmitted on December 29, 1994 (PY-CEI/NRR-1899L). This plan provides the activities needed to redress portions of the Perry site affected by the Unit 2 construction activities. Systems and equipment that were originally to be shared by both units or intended for Unit 2 but are now used to support Unit 1 operations, will continue under the full control of the Unit 1 programs. Remaining Unit 2 areas, systems and equipment that no longer support Unit 1 are to be either “abandoned” in place or physically removed. The term “abandoned” from a licensing basis perspective implies that all systems, structures and components important to safety will remain intact, but are no longer required for system/plant operation. Plant layout drawings and other figures were revised to the extent possible to reflect this condition and the USAR was revised to delete obsolete references to Unit 2 and abandoned equipment.

Revisions to this USAR will be submitted in accordance with <10 CFR 50.71(e)>. Revisions will be submitted in the form of individual revised pages with the date and applicable revision number indicated on each page. Vertical lines in the right margin will indicate changes made during the current revision. Such revisions will be made in accordance with plant procedures.

The organization of the Updated Safety Analysis Report is similar to that of the Final Safety Analysis Report and follows the guidelines established in <Regulatory Guide 1.70>, (Revision 3), dated November 1978, and the applicable portions of the USNRC Rules and Regulations, <10 CFR 50>.

The report is divided into seventeen chapters. Each chapter is divided into sections, (e.g., Section 2.1, 2.2, etc.). Each section is subdivided into text, tables and applicable figures. The tables and figures are identified and numbered with the appropriate section (e.g., Table 2.1-1, Figure 2.1-1). In chapters where appendices are required, they are included immediately following the respective chapter and designated alphabetically (e.g., 2A, 2B, 2C, etc.).

1.1 INTRODUCTION

The Final Safety Analysis Report (FSAR) was submitted in September 1980, in support of the application of The Cleveland Electric Illuminating Company (CEI), acting in behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and The Toledo Edison Company, for an operating license for the Perry Nuclear Power Plant located near Lake Erie in Lake County, Ohio. The plant site is approximately 35 miles northeast of Cleveland, Ohio, and 21 miles southwest of Ashtabula, Ohio, as discussed in detail in <Section 2.1.1>. The plant is known as the Perry Nuclear Power Plant, Unit 1.

The Energy Harbor Nuclear Corp. is authorized to act as agent for Energy Harbor Nuclear Generation LLC, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

References to the previous owners: The Cleveland Electric Illuminating Company; OES Nuclear, Inc.; Pennsylvania Power Company; the Toledo Edison Company, and FirstEnergy Nuclear Generation, LLC. will remain throughout the Updated Safety Analysis Report along with references to the previous licensee FirstEnergy Nuclear Operating Company (FENOC). This is due to the references being either historical or a description of current activities provided by those companies in support of the Perry Nuclear Power Plant operations.

The plant has a boiling water reactor nuclear steam supply system as designed and supplied by the General Electric Company and designated BWR/6, with a Mark III containment. The rated core thermal power of the unit is 3,758 MWt. The net electrical output is 1,277 MWe and the gross electrical output is 1,327.6 MWe.

The plant, including the reactor containment, was designed by Gilbert Associates, Inc., Reading, Pennsylvania, as architect-engineer and agent for the applicant. The reactor building complex includes the drywell, containment vessel, and shield building. The drywell is a reinforced concrete structure enclosing the reactor pressure vessel and the main reactor coolant loops. Outside the drywell, there is a domed cylindrical steel containment vessel supported by a reinforced concrete, steel-lined foundation mat, surrounded by a reinforced concrete shield building.

An operating license for Unit 1 was issued in March 1986. Commercial operation of Unit 1 commenced in November 1987.

A list of acronyms used throughout this Updated Safety Analysis Report is included in <Table 1.1-1>.

An index of action items from <NUREG-0737> - "TMI Action Plan Requirements For Applicants For New Operating Licenses," is provided in <Appendix 1A>.

TABLE 1.1-1

ACRONYMS

ABES	auxiliary building exhaust system
ABSS	auxiliary building supply system
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AE	architect engineer
AEC	Atomic Energy Commission (also USAEC)
AEGTS	annulus exhaust gas treatment system
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ALRM	automatic leak rate monitor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
API	automatic priority interpretation
APRM	average power range monitor
ARMS	area radiation monitoring system
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigeration, Air Conditioning Engineers
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AT/TU	analog transmitter/trip unit

TABLE 1.1-1 (Continued)

ATS	alarm trip settings
ATWS	anticipated transient(s) without scram
AWG	American Wire Gage
AWS	American Welding Society
BOL	beginning of life
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling water reactor
BWROG	BWR Owners Group
CAPCO	Central Area Power Coordination Group
CCCWS	control complex chilled water system
CCGCS	containment combustible gas control system
CCTV	closed circuit television
CCWS	component cooling water system
CEI	Cleveland Electric Illuminating Co.
CF	control factor
CFR	Code of Federal Regulations
CNRB	Company Nuclear Review Board
CO_2	carbon dioxide
CP	construction permit
CP-FES	Construction Permit Final Environmental Statement
CP-SER	Construction Permit Safety Evaluation Report
CPR	critical power ratio
CRD	control rod drive
CRD(M)	control rod drive (mechanism)

TABLE 1.1-1 (Continued)

CRDRL	control rod drive reactor level
CRDS	control rod drive system
CRERS	control room emergency recirc. system
CRT	cathode ray tube
CRVICS	containment and reactor vessel isolation control system
CSS	containment spray system
CWS	circulating water system
DAC	data acquisition and control as described in Ch. 8
DAC	derived air concentration
DBA	design basis accident
DBE	design basis earthquake
DCRDR	detailed control room design review
DEMA	Diesel Engine Manufacturers Association
DF	decontamination factor
DG	diesel generator
DGBVS	diesel generator building ventilation system
DGCWS	diesel generator cooling water system
DLF	dynamic load factor
DOP	dioctylphthalate
DTS	differential temperature switch
EAB	exclusion area boundary
EAS	essential auxiliary support
ECAR	East Central Area Reliability
ECC	emergency closed cooling

TABLE 1.1-1 (Continued)

ECCS	emergency core cooling system
ECCSCS	ECCS pump room cooling system
ECCWS	emergency closed cooling water system
EFDS	equipment and floor drainage system
EFPY	effective full-power year
EHC	electro-hydraulic control
EOF	emergency operations facility
EOL	end of life
EOOW	Engineering Officer of the Watch, USN
EQ	equipment qualification
ERIS	emergency response information system
ESF	engineered safety feature
ESFAS	engineered safety features(s) actuation signal
ESW	emergency service water
ESWS	emergency service water system
ESWVS	ESW pump ventilation system
FA	full arc
FATT	fracture appearance transition temperature
FCC	Federal Communications Commission
FCD	functional control diagram
F/D	filter/demineralizer
FENOC	FirstEnergy Nuclear Operating Company
FFWT	final feedwater temperature
FHACES	fuel handling area charcoal exhaust system
FHAES	fuel handling area exhaust system

TABLE 1.1-1 (Continued)

FHASS	fuel handling area supply system
FHB	fuel handling building
FM	Factory Mutual
FMEA	failure modes and effects analysis
FPCC	fuel pool cooling and cleanup
FPCS	fuel pool cooling system
FPER	fire protection evaluation report
FRS	floor response spectra
GDC	general design criteria
FSAR	Final Safety Analysis Report
FWLC	feedwater leakage control system
GAI	Gilbert Associates, Inc.
GE	General Electric Co.
GESSAR	General Electric Standard Safety Analysis Report
GETAB	General Electric Thermal Analysis Basis
GM	Geiger-Muller
GWMS	gaseous waste management system
GWPS	gaseous waste processing system
HAZ	heat-affected zone
HCOG	Hydrogen Control Owners Group
HCU	hydraulic control unit
HELB	high-energy line break
HEPA	high efficiency particulate air (filter)
HI-STORM	HI-STORM 1005 Version B storage cask

TABLE 1.1-1 (Continued)

HI-TRAC	HI-TRAC 125D transfer cask
HPCS	high pressure core spray
HPSP	high power set point
HVAC	heating, ventilation and air conditioning
HWL	high water level
HX	heat exchanger
I&C	instrumentation and control
IC&R	initial checkout and run-in
IDC	independent detection circuitry
IE	Office of Inspection and Enforcement
IEB	IE Bulletin
IED	instrumentation and electrical diagram
IEEE	Institute of Electrical and Electronics Engineers
IFTS	inclined fuel transfer system
IGLD	International Great Lakes Datum
IGSCC	intergranular stress-corrosion cracking
IMCPR	initial minimum critical power ratio
INPO	Institute of Nuclear Power Operation
I/O	input/output
IRM	intermediate range monitor
ISE	Independent Safety Engineering
ISFSI	Independent Spent Fuel Storage Installation
ISI	inservice inspection
ITP	initial test program

TABLE 1.1-1 (Continued)

IV	instrumented volume
KEI	Kaiser Engineers, Inc.
LCD	local climatological data
LCO	limiting condition for operation
LCS	leakage collection system
LER	licensee event report
LFMG	low frequency motor generator
LFP	Local Fire Panel
LHGR	linear heat generation rate
LLWL	low-low water level
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPMS	loose-parts monitoring system
LPRM	local power range monitor
LPSP	low power set point
LPZ	low population zone
LRW	liquid radioactive waste
LWD	low water datum
LWL	low water level
LWMS	liquid waste management system
MAPLHGR	maximum average planar linear heat generation rate
MCC	motor control center
MCES	main condenser evacuation system

TABLE 1.1-1 (Continued)

MCPR	minimum critical power ratio
M/G	motor generator
MLHGR	maximum linear heat generation rate
MMMD	mean maximum mixing depth
MOV	motor operated valve
MPC	maximum permissible concentration or multi-purpose canister
MSIV	main steam isolation valve

TABLE 1.1-1 (Continued)

MSIV-LCS	main steam isolation valve - leakage control system
MSL	mean sea level
MSLB	main steamline break
MSLIV	main steam line isolation valve
MSLR	main steamline radiation
M&TE	measuring & test equipment
NB(R)	nuclear boiler (rated)
NCC	nuclear closed cooling
NCCS	nuclear closed cooling system
NDC	net dependable capability
NDT	nil ductility transition
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NMS	neutron monitoring system
NPDES	National Pollution Discharge Elimination System
NPSH	net positive suction head
NQAD	Nuclear Quality Assurance Department, CEI
NRC	U.S. Nuclear Regulatory Commission
NSC	nonsafety class
NSOA	nuclear safety operational analysis
NSSS	nuclear steam supply system
NSSSS	nuclear steam supply shutoff system
NTOL	near-term operating licensees
OBE	operating basis earthquake

TABLE 1.1-1 (Continued)

OGSR	offgas system radiation
OL	operating license
OPRM	oscillation power range monitor
OSC	operational support center
OSHA	Occupational Safety and Health Administration
PA	public address
PAR	Programmed and Remote Analysis, Inc.
PASS	postaccident sampling system
PBX	private branch exchange
PCA	primary coolant activity
PCI	pellet/cladding interaction
PCT	peak cladding temperature
PDA	preliminary design assessment
PDR	Public Document Room
PERMISS	process and effluent radiological monitoring instrumentation and sampling systems
P-I	proportional-integral
P&ID	piping and instrumentation diagram
PGCC	power generation control complex
PLC	programmable logic controller
PMF	probable maximum flood
PMP	probable maximum precipitation
PMS	performance monitoring system
PMWP	probable maximum winter precipitation
PNPP	Perry Nuclear Power Plant

TABLE 1.1-1 (Continued)

PORC	Plant Operation Review Committee
PRA	Probabilistic Risk Assessment
PRM	process radiation monitor
PRV	primary relief valve
PSI	preservice inspection
QA	quality assurance
RBES	radwaste building exhaust system
RBSS	radwaste building supply system
RCIC	reactor core isolation cooling
RC&IS	rod control and information system
RCP	radwaste control panel
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDA	rod drop accident
RECHAR	refrigerated charcoal
RFCS	recirculation flow control system
RHR	residual heat removal
RHRS	residual heat removal system
RIA	reactivity initiated accident
RO	reactor operator
RPCS	rod pattern control system
RPM	Radiation Protection Manager
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel

TABLE 1.1-1 (Continued)

RR	reactor recirculation
RRS	required response spectra
RSS	reactor shutdown system
RV	relief valve
RWBCR	radwaste building control room
RWCU	reactor water clean-up system
RWE	rod withdrawal error
RWL	rod withdrawal limiter
RWP	radiation work permit
SACF	single active component failure
SAR	Safety Analysis Report
SBA	small break analysis
SBLOCA	small-break loss-of-coolant accident
SCR	silicon controlled rectifier
SDIV	scram discharge instrument volume
SDV	scram discharge volume
SEF	single equipment failure
SER	safety evaluation report
SFDS	spent fuel dry storage
SFP	spent fuel pool
SJAE	steam jet air ejector
SLCS	standby liquid control system
SOE	single operator error
SPDS	safety parameter display system

TABLE 1.1-1 (Continued)

SPMU	suppression pool make-up system
SPT	standard penetration test
SQRT	Seismic Qualification Review Team
SRDI	safety-related display instrumentation
SRM	source range monitor
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SRV	safety relief valve
SRW	solid radwaste system
SSE	safe shutdown earthquake
SSER	Supplement to Safety Evaluation Report
SSI	soil/structure interaction
STA	Shift Technical Advisor
SV	safety valve
SWMS	solid waste management system
TAF	top of active fuel
TCV	turbine control valve
TDI	Transamerica De Laval, Inc.
TEC	training center
TEMA	Tubular Exchanger Manufacturers Association
TGSS	turbine gland sealing system
TG	turbine generator
TIP	traversing incore probe

TABLE 1.1-1 (Continued)

TLD	thermoluminescent dosimeter
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2
TPM	thermal power monitor
TRS	test response spectra
TS	temperature switch
UAT	unit auxiliary transformer
UHF	ultra-high frequency
USAR	Updated Safety Analysis Report
USGS	United States Geological Survey
UT	ultrasonic test
UTM	universal transverse Mercator
VCT	vertical cask transporter
VHF	very high frequency
WARF/RISB	Waste Abatement and Reclamation Facility/Radwaste Interim Storage Building
ZPA	zero period acceleration
ZPT	zero profile transporter

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PRINCIPAL DESIGN CRITERIA

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

Power generation design criteria are:

a. The plant is designed to produce steam for direct use in a turbine generator unit.

b. Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.

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

d. The fuel cladding, in conjunction with other plant systems, is designed to retain its structural integrity, such that any failures will be within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

e. Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.

f. Reactor power level is manually controllable.

g. Control of the reactor is possible from a single location.

h. Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.

i. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineered safety features.

j. The station is designed for routine continuous operation whereby steam activation products, fission products, corrosion products, and coolant dissociation products are processed within acceptable limits.

1.2.1.1.2 Safety Design Criteria

Safety design criteria are:

a. The station design conforms to applicable codes and regulations.

b. The station is designed, fabricated, erected, and operated in such a way that the release of radioactive materials to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and for abnormal transients and accidents.

c. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

d. The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems.

e. Gaseous, liquid and solid waste disposal facilities are designed so that the discharge of radioactive effluents and offsite shipment of radioactive materials can be made, in accordance with applicable regulations.

f. The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

g. Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.

h. Radiation shielding is provided and access control patterns are established, to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations.

i. Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material containment barrier following abnormal operational transients and accidents.

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

k. Where positive, precise action is immediately required, in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

l. Essential safety actions are provided by equipment of sufficient redundance and independence such that no single failure of (a) active components or (b) passive components in certain cases, in the long term, will prevent the required actions. For systems or components to which IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations" and/or IEEE-308, "Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," applies, single failures of either active or passive electrical components are considered in recognition of the higher anticipated failure rates of passive electrical components, relative to passive mechanical components.

m. Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room.

n. Nuclear safety systems and engineered safety features are designed to permit demonstration of their functional performance requirements.

o. The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances, such as earthquakes, floods and storms at the station site.

p. Standby electrical power sources have sufficient capacity to power all nuclear safety systems and engineered safety features requiring electrical power concurrently.

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

r. A containment is provided that completely encloses the reactor system, drywell and suppression pool. The containment employs the pressure suppression concept.

s. It is possible to test primary containment integrity and leak tightness at periodic intervals.

t. A secondary containment is provided that completely encloses the primary containment. This secondary containment contains a system for controlling the release of radioactive materials from the primary containment.

u. The primary containment and secondary containment, in conjunction with other engineered safety features, limit radiological effects of accidents resulting in the release of radioactive material to the containment volumes to less than the prescribed acceptable limits.

v. Provisions are made for removing energy from the primary containment, as necessary, to maintain the integrity of the containment system following accidents that release energy to the containment.

w. Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is imminent. Such isolation is performed in time to limit radiological effects to less than the specified acceptable limits.

x. Emergency core cooling systems are provided to limit fuel cladding temperature to less than that which would cause fragmentation in the event of a loss-of-coolant accident.

y. The emergency core cooling systems provide for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.

z. Operation of the emergency core cooling systems is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the station.

aa. The control room is shielded against radiation so that continued occupancy under accident conditions is possible.

bb. In the event that the control room becomes inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing the local controls and equipment that are available outside the control room.

cc. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition.

dd. Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

ee. Systems that have redundant or backup safety functions are physically separated and arranged, such that any credible events causing damage to any one region of the reactor island complex has minimum prospect for compromising the functional capability of the designated counterpart system.

1.2.1.2 <u>System Criteria</u>

The principal design criteria for particular systems are listed in the sections that follow.

1.2.1.2.1 Nuclear System Criteria

Nuclear system criteria are:

a. The fuel cladding is designed to retain integrity, such that any failures are within acceptable limits as a radioactive material barrier throughout the design power range.

b. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity, such that any failures are within acceptable limits throughout any abnormal operational transient.

c. Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material barrier during normal operation and following abnormal operational transients and accidents.

d. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational transients as well as for abnormal operational transients. The capacity of such systems is adequate to prevent fuel cladding damage.

e. Heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperable. The capacity of such systems is adequate to prevent fuel cladding damage. The reactor is capable of being shut down automatically in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems.

f. The reactor core and reactivity control systems are designed so that control rod action is capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

g. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

h. The nuclear system is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

1.2.1.2.2 Power Conversion Systems Criteria

Components of the power conversion systems are designed to perform these basic objectives:

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

b. Assure that any fission product or radioactive material associated with the steam and condensate during normal operation is safely contained inside the system or is released under controlled conditions, in accordance with waste disposal procedures.

1.2.1.2.3 Electrical Power Systems Criteria

Sufficient normal auxiliary and standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required essential safety actions under all postulated accident conditions.

1.2.1.2.4 Radwaste System Criteria

Radwaste system criteria are:

a. The gaseous and liquid radwaste systems are designed to limit the release of radioactive effluents from the station to the environs to the lowest reasonably achievable values. Such releases, as may be necessary during normal operation, are limited to values that meet the requirements of applicable regulations, including <10 CFR 20> and <10 CFR 50>.

b. The solid radwaste disposal systems are designed so that in-plant processing and offsite shipments are in accordance with all applicable regulations, including <10 CFR 20>, <10 CFR 71> and <49 CFR 171>, <49 CFR 172>, <49 CFR 173>, <49 CFR 174>, <49 CFR 175>, <49 CFR 176>, <49 CFR 177>, <49 CFR 178>, and <49 CFR 179>, and DOT Regulations, as appropriate.

c. The systems' design provides the means by which station operations personnel are alerted whenever specified limits on the release of radioactive material may be approached.

1.2.1.2.5 Auxiliary Systems Criteria

Auxiliary systems criteria are:

a. Fuel handling and storage facilities are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel. Provisions are made for maintaining the cleanliness of spent fuel cooling and shielding water.

b. Other auxiliary systems, such as service water, cooling water, fire protection, heating and ventilating, communications, and lighting, are designed to function during normal and/or accident conditions.

c. Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe condition are designed such that a failure of these systems will not prevent the essential auxiliary systems from performing their design functions.

1.2.1.2.6 Shielding and Access Control Criteria

Shielding and access control criteria are:

a. Radiation shielding is provided, and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of published regulations in any normal mode of plant operation. <Section 12.3.1>

b. The control room is shielded against radiation so that occupancy is possible under accident conditions.

1.2.1.2.7 Nuclear Safety Systems and Engineered Safety Features Criteria

Principal design criteria for nuclear safety systems and engineered safety features are as follows:

a. These criteria correspond to criteria j. through q., x. through z., bb. and cc. in <Section 1.2.1.1.2>.

b. Standby electrical power sources have sufficient capacity to power all Class 1E and all engineered safety features requiring electrical power concurrently.

c. Standby electrical power sources are provided, as necessary, for support of engineered safety feature functions (e.g., decay heat removal) under circumstances where normal auxiliary power is not available.

d. In the event that the control room is inaccessible, it is possible to bring the reactor from power range operation to a hot shutdown condition by use of controls and equipment that are available outside the control room. Furthermore, station design includes the ability, in this event, for operators to bring the reactor to a cold shutdown condition from the hot shutdown condition from outside the control room.

e. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from operating conditions and, subsequently, to maintain the shutdown condition.

1.2.1.2.8 Process Control System Criteria

The principal design criteria for the process control systems are in the sections that follow.

1.2.1.2.8.1 Nuclear System Process Control Criteria

Nuclear system process control criteria are:

a. Control equipment is provided to allow the reactor to respond automatically to main load changes within design limits.

b. Manual control of the reactor power level is possible.

c. Control of the nuclear system is possible from a central location.

d. Nuclear systems' process controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

e. Interlocks or other automatic equipment are provided as a backup to procedural controls to avoid conditions requiring the actuation of engineered safety features.

1.2.1.2.8.2 Power Conversion Systems Process Control Criteria

Power conversion systems process control criteria are:

a. Control equipment is provided to control the reactor pressure throughout its operating range.

b. The turbine is able to respond automatically to minor changes in load.

c. Control equipment in the feedwater system maintains the water level in the reactor vessel at the optimum level required by steam separators.

d. Control of the power conversion equipment is possible from a central location.

e. Interlocks or other automatic equipment are provided in addition to procedural controls to avoid conditions requiring the actuation of engineered safety features.

1.2.1.2.8.3 Electrical Power System Process Control Criteria

Electrical power system process control criteria are:

a. Class 1E power systems are designed as an "n" bus system, with any "n-1" buses being adequate to safely shut down the unit.

b. Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance, in the event of equipment failure.

c. Voltage relays are used on the emergency equipment buses to isolate these buses from the normal electrical system, in the event of loss of offsite power, and to initiate starting of the standby emergency power system diesel generators.

d. Standby emergency power diesel generators are started and loaded automatically to meet the existing emergency condition.

e. Electrically operated breakers are controllable from the control room.

f. Monitoring of essential generators, transformers and circuits is provided in the control room.

1.2.1.2.9 Spent Fuel Dry Storage System Criteria

The principle design criteria of the Spent Fuel Dry Storage (SFDS) are addressed in the HI-STORM FSAR. Final Safety Analysis Report for HOLTEC International storage and transfer operation reinforced module cask system (HI-STORM 100 cask system) HOLTEC International Report No. HI-2002444, Docket 72-1014, Revision 7, August 2008.

1.2.2 PLANT DESCRIPTION

1.2.2.1 Site Characteristics

The information in this Section is historical. That is, information originally provided in the Final Safety Analysis Report (FSAR) to meet the requirements of <10 CFR 50.34(b)> and was accurate at the time the plant was originally licensed, but is not intended to be updated for the life of the plant.

1.2.2.1.1 Location of Site

The plant site is located along the southern shoreline of Lake Erie in a rural area of Lake County, Ohio, approximately 7 miles northeast of Painesville and 35 miles northeast of Cleveland, Ohio. The centerline of the reactor for Unit 1 is located at latitude 41°48′04.2″ and longitude 81°08′36″ and the centerline of the reactor for Unit 2 at latitude 41°48′02.3″ and longitude 81°08′35.6″, as shown on <Figure 2.1-3>.

A more complete description of the site location is given in <Section 2.1>.

1.2.2.1.2 Description of Plant Environs

The plant is located on a relatively flat site of approximately 1,030 acres, located about 50 feet above the low water datum of Lake Erie, with a very gentle slope toward the lake. About 45 percent of the site area is covered with light to heavy woodland.

There are no domestic residences within the site boundaries.

The exclusion area is established as the area within a 2,900-foot radius centered on the Unit 1 and Unit 2 reactors.

1.2.2.1.3 Design Bases Dependent on Site

Only two small streams run close to the site, neither of which have any upstream dams. This, together with the plant location about 50 feet above Lake Erie, results in a negligible possibility of flooding. There are no capable faults at or near the site. The site is located in a temperate climate zone.

These factors make it unnecessary to establish any unusual design bases for the plant.

1.2.2.2 General Arrangement of Structures and Equipment

The plot plan and general arrangements of structures and equipment for the Perry Nuclear Power Plant are shown on <Figure 1.2-2>, <Figure 1.2-3>, <Figure 1.2-4>, <Figure 1.2-5>, <Figure 1.2-6>, <Figure 1.2-7>, <Figure 1.2-8>, <Figure 1.2-9>, <Figure 1.2-10>, <Figure 1.2-11>, <Figure 1.2-12>, <Figure 1.2-13>, <Figure 1.2-14>, <Figure 1.2-15>, <Figure 1.2-16>, and <Figure 1.2-17>. A plot plan of the site, showing radioactive and nonradioactive release points to the environment, is shown on <Figure 1.2-18>.

1.2.2.3 Nuclear System

The nuclear system includes a direct cycle, forced circulation, General Electric boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown as <Figure 1.2-19>.

1.2.2.3.1 Reactor Core and Control Rods

Fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircaloy tubes. These tubes (or fuel rods) are assembled into individual fuel assemblies. Gross control of the core is achieved by movable, bottom entry control rods. The control rods are cruciform in shape and are dispersed throughout the lattice of fuel assemblies. The control rods are positioned by individual control rod drives.

Some fuel assemblies have several fuel rods with gadolinia (Gd_2O_3) mixed in solid solution with the UO_2. The Gd_2O_3 is burnable poison which diminishes the reactivity of the fresh fuel. It is depleted as the fuel reaches the end of its first cycle.

A conservative limit of plastic strain is the design criterion used for fuel rod cladding failure. The peak linear heat generation for steady-state operation is well below the fuel damage limit, even late in life. Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.2.3.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include steam lines, coolant recirculation lines, feedwater lines, control rod drive and in-core nuclear instrument housings, core spray lines, core differential pressure line, jet pump pressure sensing lines, and water level instrumentation lines.

The reactor vessel is designed and fabricated in accordance with applicable codes. The nominal operating pressure in the steam space above the separators is 1,040 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head, nozzles and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The initial and largest separation of liquid from vapor is made by the steam separators. The steam is then dried to greater than 99.9 percent quality by steam dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with three isolation valves in series: one on the inside and two on the outside of the containment barrier.

1.2.2.3.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one high capacity motor-driven recirculation pump, two motor-operated maintenance valves and one hydraulically operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low frequency motor-generator set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are reactor vessel internals. The jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break would still allow core flooding to approximately two-thirds of the core height (the level of the inlet of the jet pumps).

1.2.2.3.4 Residual Heat Removal System

The residual heat removal (RHR) system is a system of pumps, heat exchangers and piping that fulfills the following functions:

a. Removes decay and sensible heat during and after plant shutdown.

b. Injects water into the reactor vessel, following a loss-of-coolant accident, to reflood the core and maintain fuel cladding below fragmentation temperature independent of other core cooling systems. This is discussed in <Section 1.2.2.4.8>.

c. Removes heat from the containment, following a loss-of-coolant accident, to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the containment air space (containment spray) with suppression pool water.

1.2.2.3.5 Reactor Water Cleanup System

The reactor water cleanup system recirculates a portion of reactor coolant through a filter demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

1.2.2.3.6 Nuclear Leak Detection System

The nuclear leak detection and monitoring system consists of temperature, pressure, flow, and fission product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

a. Main steam lines

b. Reactor water cleanup (RWCU) system

c. Residual heat removal (RHR) system

d. Reactor core isolation cooling (RCIC) system

e. Feedwater system

f. ECCS systems

g. Miscellaneous systems

Small leaks generally are detected by monitoring the air coolers condensate flow, radiation levels and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.4 Nuclear Safety Systems and Engineered Safety Features

1.2.2.4.1 Reactor Protection System

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

1.2.2.4.2 Neutron Monitoring System

Those portions of the neutron monitoring system that are part of the reactor trip system qualify as a nuclear safety system. The intermediate range monitors (IRM) and the average power range monitors (APRM), which monitor neutron flux by in-core detectors, provide scram logic inputs to the reactor trip system to initiate a scram in time to prevent excessive fuel clad damage, as a result of over-power transients. The APRM system also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals. The LPRM system provides signals to the oscillation power range monitors (OPRM’s) which detect evidence of reactor thermal-hydraulic instability and provide logic input to the RPS to scram the reactor if the instability is of sufficient magnitude.

1.2.2.4.3 Control Rod Drive System

When a scram is initiated by the reactor protection system, the control rod drive system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high pressure water stored in an accumulator in the hydraulic control unit or reactor pressure forces its control rod into the core.

1.2.2.4.4 Control Rod Drive Housing Supports

Control rod drive housing supports are located underneath the reactor vessel, near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure, and thus, protect the fuel barrier.

1.2.2.4.5 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. This action limits the rate of reactivity insertion resulting from a rod drop accident. The limiters contain no moving parts.

1.2.2.4.6 Nuclear System Pressure Relief System

A pressure relief system, consisting of safety relief valves mounted on the main steam lines, is provided to prevent excessive pressure inside the nuclear system for operational transients or accidents.

1.2.2.4.7 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCIC) provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel, for events defined in <Section 5.4.6.1>.

1.2.2.4.8 Emergency Core Cooling Systems

Four emergency core cooling systems are provided to maintain fuel cladding below the temperature limit in <10 CFR 50.46>, in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. The systems are:

a. High Pressure Core Spray

   The high pressure core spray (HPCS) system provides and maintains an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant pressure boundary. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero. The HPCS cooling decreases vessel pressure to enable the low pressure cooling systems to function. The HPCS system pump motor is powered by a diesel generator if auxiliary power is not available; the system may also be used as a backup for the RCIC system.

b. Automatic Depressurization

   The automatic depressurization system (ADS) rapidly reduces reactor vessel pressure in a loss-of-coolant accident (LOCA) in which the HPCS system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low pressure emergency core cooling systems to deliver cooling water to the reactor vessel. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating both, that a break in the reactor coolant pressure boundary has occurred and that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS will not be activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after the depressurization.

c. Low Pressure Core Spray

The low pressure core spray (LPCS) system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the reactor coolant pressure boundary. However, water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into each fuel channel. The LPCS loop functioning, in conjunction with ADS or HPCS, can provide sufficient fuel cladding cooling following a loss-of-coolant accident.

d. Low Pressure Coolant Injection

Low pressure coolant injection (LPCI) is an operating mode of the RHR system, but is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with the other emergency core cooling systems. LPCI uses the pump loops of RHR to inject cooling water into the pressure vessel. LPCI is actuated by conditions indicating a breach in the reactor coolant pressure boundary, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation provides the capability of core reflooding, following a loss-of-coolant accident, in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.4.9 Containment

1.2.2.4.9.1 Containment Design

The containment system consists of the following components:

a. A drywell enclosing the reactor pressure vessel, the reactor coolant recirculation loops and pumps, and other branch connections of the reactor primary system. The drywell is a cylindrical reinforced concrete structure with a removable steel head.

b. A suppression pool containing a large amount of water used to rapidly condense steam from reactor vessel blowdown or from a break in a major pipe.

c. A leaktight containment vessel completely surrounding the drywell and the suppression pool. The containment vessel is a cylindrical steel structure with a dome and flat bottom supported by a reinforced concrete foundation mat.

Part of the suppression pool water is inside the drywell (retained by a cylindrical concrete retaining wall), but the major part is outside the drywell between the drywell wall and the containment wall. A system of vents, located below the suppression pool water level, connects the inner and outer parts of the suppression pool. In the event of a process piping failure within the drywell, the increased pressure inside the drywell will force a mixture of air, steam and water through the vents to the major volume of the suppression pool where the steam will be rapidly condensed. The noncondensible gases will escape into the free air volume inside the containment vessel where they will be contained.

Equipment and facilities, located inside the containment vessel but outside the drywell, include the control rod drive modules, major components of the reactor water cleanup system, the standby liquid control system, and the reactor refueling facilities.

1.2.2.4.9.2 Heat Removal

The containment heat removal system is summarized in <Section 1.2.2.4.14>.

1.2.2.4.9.3 Shield Building

The shield building is a cylindrical concrete structure, with a domed top, completely enclosing the containment vessel. The annular space between the shield building and the containment vessel is normally kept at a slightly negative pressure, relative to atmospheric pressure, so that any leakage through the shield building or the containment vessel is into this space. The ventilation exhaust from this area is treated by the annulus exhaust gas treatment system through roughing, HEPA and charcoal filters and through HEPA after-filters. Instrumentation is provided to monitor the radioactivity level in the exhaust and alarms in the control room, in the unlikely event of abnormally high radioactivity levels.

The shield building structure provides shielding to minimize direct radiation to operating personnel and/or the public under normal operating and accident conditions. It also provides weather and external missile protection for the containment vessel.

1.2.2.4.9.4 Containment Spray

A containment spray system is provided to operate in conjunction with the combustible gas control system. The containment spray system will function, by automatic initiation, to condense steam to reduce pressure that has built up in the containment. The containment spray system consists of two redundant subsystems, each with its own full capacity spray header. Each subsystem is supplied from a separate redundant RHR subsystem.

1.2.2.4.9.5 Combustible Gas Control

In order to assure containment integrity following a LOCA, means are provided, as necessary, for controlling the concentration of combustible gas in the containment after the LOCA. Initial control will be accomplished by mixing volumes of relatively high combustible gas concentration with those of low concentration. When the amount of combustible gas present becomes too large for mixing to be of further value, combustible gas control equipment will be put into operation to reduce the combustible gas concentration.

1.2.2.4.10 Containment and Reactor Vessel Isolation Control System

The containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all process lines which are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a breach in the reactor coolant pressure boundary.

1.2.2.4.11 Main Steam Isolation Valves

All pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities. Additionally, the main steam lines are given special isolation consideration because of their large size and large mass flow rates. Automatic isolation valves are provided in each main steam line. Each is powered by both air pressure and spring force. These valves fulfill the following objectives:

a. Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel. Such a loss may stem from either a major leak from the steam piping outside the containment or a malfunction of the pressure control system, and result in excessive steam flow from the reactor vessel.

b. Limit the release of radioactive material by isolating the reactor coolant pressure boundary in case of a gross release of radioactive material from the fuel to the reactor cooling water and steam.

c. Limit the release of radioactive material by closing the containment barrier in case of a major leak from the nuclear system inside the containment.

1.2.2.4.12 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the reactor vessel before the main steam isolation valves are closed, in case of a main steam line break outside the containment.

1.2.2.4.13 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors isolate and trip the mechanical vacuum pumps.

1.2.2.4.14 Residual Heat Removal System (Containment Cooling)

The containment cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool and of the atmosphere in the drywell and in the suppression chamber, following a design basis loss-of-coolant accident; to control the pool temperature during normal operation of the safety relief valves and the RCIC system; and to reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the service water system. The RHR fluid is discharged back to the suppression pool.

1.2.2.4.15 Ventilation Exhaust Radiation Monitoring System

The process ventilation radiation monitoring systems consist of a number of radiation monitors arranged to monitor the activity level of the air exhaust from the containment drywell, auxiliary building, fuel handling building, controlled access area, radwaste building, offgas building, turbine building, heater bay, and intermediate building.

1.2.2.4.16 Annulus Exhaust Gas Treatment System

The annulus exhaust gas treatment system consists of two redundant subsystems designed to filter any airborne radioactive iodine and particulates from the air which leak out of the containment vessel. Each subsystem, consisting of a 100 percent capacity exhaust fan, roughing, HEPA, and charcoal filters, and exhausts to the plant vent. The treated exhaust gases are monitored prior to release to the atmosphere, and if the radiation level in the operating subsystem exhaust should reach or exceed the selected setpoint, an alarm will be sounded in the control room.

1.2.2.4.17 Standby AC Power Supply

The auxiliaries connected to the engineered safety features buses are normally supplied from the unit’s startup transformer through a 13.8/4.16 kV bus tie transformer. These auxiliaries do not go through an automatic transfer on loss of the generator source. However, on complete loss of offsite power, the engineered safety features loads are automatically transferred to the ESF diesel generators.

The engineered safety feature systems consist of three redundant and independent load groups per generator unit designated as Division 1, 2 and 3. Each group consists of 4.16 kV, 480 volt and 120 volt ac and 125 volt dc systems.

1.2.2.4.18 DC Power Supply

Station batteries are included for circuit breaker control power, selected emergency lighting and operating power for vital instrumentation and control until offsite power is restored or onsite emergency generation is available. Onsite emergency power is supplied by diesel generators. Critical instrumentation is fed from buses which are powered from the station batteries through inverters to provide a reliable and stable power supply. This guarantees continuous monitoring and control of critical instrument channels.

1.2.2.4.19 Standby Liquid Control System

The standby liquid control system provides backup capability for reactivity control, independent of normal reactivity control provisions, and is able to shut down the reactor if normal control becomes inoperative. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The backup system has the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to ensure complete shutdown from the most reactive condition at any time in core life.

1.2.2.4.20 Safe Shutdown from Outside the Control Room

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by the use of the local controls and equipment that are available outside the control room.

1.2.2.4.21 Main Steam Line Isolation Valve Leakage Control System

The main steam line isolation valve leakage control system (MSIV-LCS) has been eliminated and is abandoned in place.

1.2.2.5 Power Conversion Systems

The unit utilizes a power conversion system which includes a turbine generator, a main condenser, condensate pumps, an air ejector, turbine gland seal condensers, condensate demineralizers, the feedwater heating system, and reactor feedpumps. These components produce electrical power from the steam coming from the reactor, condense the steam into water and return the heated feedwater to the reactor. The circulating water system removes the heat rejected to the main condenser.

1.2.2.5.1 Turbine Generator

The turbine is a General Electric tandem compound, six flow, double reheat, 1,800 rpm unit. The unit consists of one double flow, high pressure turbine and three double flow, low pressure turbines. Exhaust steam from the high pressure turbine passes through moisture separators and two stages of reheaters before it enters the three low pressure turbines. Steam is extracted for six stages of feedwater heating, and to supply two reactor feed pump drive turbines. Turbine controls include a speed governor, stop valves, control valves, and supervisory, protective and operating instruments. The generator is direct driven and conductor cooled, with a direct driven exciter unit.

1.2.2.5.2 Main Steam System

The main steam system consists of four main steam lines from the outermost containment isolation valves to the turbine stop valves, connecting lines to supply steam to the second stage reheater, condenser steam jet air ejectors, main turbine bypass valves, reactor feed-pump turbines, and the seal steam evaporator.

1.2.2.5.3 Main Condenser

The main condenser is a three shell, series flow, triple pressure design, with shells arranged beneath the low pressure elements of the turbine and tubes oriented transversely to the turbine generator axis. Each hotwell provides for tube leak detection and isolation of the circulating water passes. The main condenser hotwell maintains a minimum retention time of 3 minutes for radioactive decay.

Deaeration in each condenser shell provides for the removal of normal air inleakage plus the hydrogen and oxygen gases contained in the turbine steam due to radiological dissociation of water in the reactor.

1.2.2.5.4 Main Condenser Evacuation System

The main condenser gas removal system includes two steam jet air ejector units, complete with condensers, which remove air and noncondensible gases from the main condenser. Mechanical vacuum pumps which discharge to the atmosphere are provided to evacuate the condenser prior to startup. The air ejectors discharge to the offgas system.

1.2.2.5.5 Turbine Gland Sealing System

The turbine gland sealing system consists of a nonradioactive steam source, seal steam pressure regulator, steam seal header, steam packing exhausters, exhaust blowers, and the associated piping and valves. The turbine gland sealing system discharges to the atmosphere.

1.2.2.5.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation.

1.2.2.5.7 Circulating Water System

Heat rejected to the circulating water in the condenser is dissipated to the atmosphere by means of a natural draft cooling tower. The locations of the cooling towers and pumphouses are shown on <Figure 1.2-2>.

1.2.2.5.8 Condensate Polishing System

To ensure that the reactor receives water of the required purity, the unit is furnished with a full flow condensate demineralizer and condensate filtration system. Corrosion products that originate in the turbine, condenser, steam and drain piping, and the tube side of the feedwater heaters are removed from the condensate by this system. The system also protects the reactor against condenser tube leaks and removes impurities which might enter the condensate system with makeup water. The demineralizer and filter vessels are located in a shielded area.

1.2.2.5.9 Condensate and Feedwater System

To maintain reactor water level, the condensate and feedwater systems take water from the main condenser and deliver it to the reactor. The condensate pumps take suction from the condenser hotwell storage and discharge (through the steam jet air ejector condensers, the turbine steam packing exhausters, and the condensate cleanup system) to the suction at the condensate booster pumps. These pumps discharge through the low pressure feedwater heaters to the open, direct contact heater. The direct contact heater is mounted on the hot surge tank. Feedwater is taken from the hot surge tank by the reactor feed booster pumps which discharge through an intermediate pressure feedwater heater to the reactor feed pumps. These pumps transmit feedwater through a set of high pressure feedwater heaters to the reactor.

The three low pressure feedwater heaters and the two high pressure feedwater heaters are of the closed shell and tube type. With the exception of the fifth stage high pressure feedwater heater, all closed heaters are provided with internal drain coolers. The fourth stage of feedwater heating is done in an open, direct contact heater mounted on top of a storage tank. The storage tank provides minimum retention time of 2 minutes for radioactive decay. Drains from the two high pressure heaters are cascaded to the open heater. Drains from the low pressure heaters are cascaded to the main condenser.

Two nominal half capacity horizontal reactor feed pumps are connected directly to variable speed turbine drives. The turbines normally take steam from the main turbine cross-around steam line after the moisture separators and reheaters. A control system regulates feedwater flow to maintain reactor water level by controlling the admission of steam to the turbine drives. There is also a 20 percent electric-driven feed pump.

1.2.2.6 Electrical Systems and Instrumentation and Control

1.2.2.6.1 Electrical Power Systems

The plant consists of one 1,277 MWe (net) operating unit which generates power at 22 kV. The power from the Unit is fed through an isolated phase bus to the unit's main transformer where it is stepped up to 345 kV and delivered to the adjacent 345 kV switchyard.

The 345 kV switchyard serves four transmission circuits with a provision for one future circuit. These four circuits connect to the existing CEI transmission network. The switchyard has a minimum breaker-and-a-half configuration and serves as the point of connection of the two preferred sources (Unit 1 and Unit 2 startup transformers) to the offsite transmission system.

Each startup transformer (345-13.8 kV) is fed from the 345 kV transmission station switchyard. These two startup transformers back up one another through the use of high speed automatic transfer. The transfer will not occur if an open phase condition is sensed on the high voltage side of either transformer. The startup transformers are sized to provide safe shutdown under all conditions with only one transformer inservice.

The power required during normal operation for a unit's auxiliaries, which are not connected to engineered safety features buses, is supplied from the generator through the station service transformer (22-13.8 kV). These auxiliaries are automatically transferred to the unit's startup transformer upon failure of the generator source.

All electric systems and components essential for plant safety are designed as Class 1E electrical power systems and are located in Seismic Category I structures so that their integrity is not impaired by the applicable design basis events.

1.2.2.6.2 Nuclear System Process Control and Instrumentation

1.2.2.6.2.1 Rod Control and Information System

The rod control and information system (RCIS) provides the means by which control rods are positioned from the control room for power control. The system operates valves in each hydraulic control unit to change control rod position. One gang of control rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.6.2.2 Recirculation Flow Control System

During normal power operation, a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core and thereby changes the core power level. The system can automatically adjust the reactor power output to the load demand. For startup and shutdown flow changes at lower power, the pump speed is changed by adjusting the frequency of the electrical power supply.

1.2.2.6.2.3 Neutron Monitoring System

The neutron monitoring system is a system of incore neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRMs) and the intermediate range monitors (IRMs) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRMs) and average power range monitors (APRMs) allow assessment of local and overall flux conditions during power range operation. The traversing incore probe system (TIP) provides a means to calibrate the individual LPRM sensors. The neutron monitoring system provides inputs to the reactor manual control system to initiate rod blocks if preset flux limits are exceeded. It also provides inputs to the reactor protection system to initiate a scram if other limits are exceeded.

1.2.2.6.2.4 Refueling Interlocks

A system of interlocks, is provided to prevent an inadvertent criticality during refueling operations. This system restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple, and control rods.

1.2.2.6.2.5 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage.

1.2.2.6.2.6 Process Computer System

An online process computer is provided to monitor and log process variables and to make certain analytical computations.

1.2.2.6.3 Power Conversion Systems Process Control and Instrumentation

1.2.2.6.3.1 Pressure Regulator and Turbine Generator Control

The pressure regulator maintains control of the turbine control and turbine bypass valves to allow proper generator and reactor response to system load demand changes while maintaining the nuclear system pressure essentially constant.

The turbine generator speed-load controls maintain constant turbine speed (generator frequency) and respond to load changes by adjusting the reactor recirculation flow control system and pressure regulator setpoint.

The turbine generator speed-load controls initiate rapid closure of the turbine control valves and rapid opening of the turbine bypass valves to prevent turbine overspeed on loss of the generator electric load.

1.2.2.6.3.2 Feedwater Control System

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A conventional, three element, control system is used to accomplish this function.

1.2.2.7 Fuel Handling and Storage Systems

1.2.2.7.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to Seismic Category I requirements, and assurance that k_{eff} will not exceed 0.95 under dry or flooded conditions. This subject is further discussed in <Section 9.1>.

1.2.2.7.2 Fuel Handling System

The fuel handling equipment includes a 125 ton cask crane, fuel handling platform, fuel inspection stand, fuel preparation machine, fuel assembly transfer mechanism, containment refueling platform, a 125 ton containment crane, and other related tools for reactor servicing. All equipment conforms to applicable codes and standards.

The only function of the cask crane is to handle the spent fuel cask. The fuel handling platform transfers the fuel assemblies between the transfer pool, storage pools and cask. Fuel assemblies are transferred through the transfer tube between the reactor building and the fuel building. The fuel assemblies inside the containment are handled by the refueling platform.

The handling of the reactor head, removable internals and drywell head, during refueling, is accomplished using the containment crane.

All tools and servicing equipment necessary to meet the reactor general servicing requirements are designed for efficiency and safe serviceability.

1.2.2.8 <u>Cooling Water and Auxiliary Systems</u>

1.2.2.8.1 Emergency Closed Cooling System

The unit is equipped with an emergency closed cooling system that provides seal cooling water to the RHR pumps. The system also provides cooling water for the room coolers associated with the RCIC, RHR and LPCS pumps, the control complex chillers and the hydrogen analyzers.

Cooling is provided through the emergency closed cooling system heat exchangers mentioned in <Section 1.2.2.8.2>.

The system is designed with redundant capability to ensure operability of its cooling during all normal and emergency operations of the plant. Redundant power supplies are provided for use in the event of loss of offsite power.

1.2.2.8.2 Emergency Service Water System

The emergency service water system provides cooling water to equipment required for normal and emergency shutdown of the reactor. The unit is also equipped with a separate emergency service water system that provides cooling water to RHR heat exchangers (A and B) Fuel Pool Cooling Heat Exchangers, emergency diesel generator heat exchangers, and emergency closed cooling system heat exchangers. It can also provide water to the site Fire Protection System (P54), the Fuel Pool Cooling and Cleanup System (G41), the Emergency Closed Cooling Water System (P42), the Residual Heat Removal System (to provide containment flooding) (E12), and the Standby Liquid Control System (C41). The system is designed with sufficient redundancy to ensure heat removal capability during shutdown, hot standby, accident conditions, and refueling operations. Redundant power supplies are provided for use in the event of loss of offsite power.

1.2.2.8.3 Fuel Pool Cooling and Cleanup System

A fuel pool cooling and cleanup system is provided to remove decay heat from the spent fuel stored in the fuel pool and to maintain a specified water temperature, purity, clarity, and level. In the event of an abnormal heat load, the RHR system can be used to supplement the normal cooling system.

1.2.2.8.4 Service Water System

An open cycle cooling water system is provided to supply lake water to the unit for cooling the turbine building closed loop heat exchangers, the turbine lube oil coolers and the nuclear closed cooling heat exchangers. The system also supplies water to the screen wash pumps. This system consists of four, one-third capacity pumps, automatic self cleaning strainers, and a piping network to distribute cooling water to the tube side of the heat exchangers and back to the lake, by way of the condenser circulating water return lines.

1.2.2.8.5 Turbine Building Closed Cooling System

A closed cooling system is provided to supply the cooling water to the various turbine plant components that require it. The system consists of a closed loop network in which buffered condensate quality water is cooled with lake water in a shell and tube type heat exchanger and is circulated through the components to be cooled, with a set of centrifugal pumps.

Required static head is maintained on the system with an open surge tank which is located at the highest elevation in the system. Water level in the surge tank is maintained with a level controller using makeup water from the two-bed demineralized water system.

1.2.2.8.6 Nuclear Closed Cooling System

The nuclear closed cooling system provides cooling water to certain designated equipment located in the containment, the auxiliary building and in the fuel handling and radwaste buildings. Adequate capacity and redundancy is provided in heat exchangers and pumps to ensure performance of the cooling system under normal modes of plant operation. In the event of loss of offsite power, cooling is restored to designated equipment by the emergency closed cooling system.

1.2.2.8.7 Makeup Water Treatment System

A makeup water treatment system is provided to supply reactor quality water for plant makeup.

1.2.2.8.8 Potable and Sanitary Water System

A water system for drinking and sanitary uses is provided for the plant.

1.2.2.8.9 Process Sampling System

The process sampling system is furnished to provide process information that is required to monitor plant and equipment performance and changes in operating parameters. Representative liquid and gas samples are taken automatically and/or manually during normal plant operation for laboratory or online analyses.

1.2.2.8.10 Equipment and Floor Drains

The plant equipment and floor drainage system handles both radioactive and nonradioactive wastes. Wastes which may contain radioactive materials are pumped to the radwaste system for cleanup and then reused or discharged. After monitoring, nonradioactive effluents are discharged to the environs.

1.2.2.8.11 Service and Instrument Air Systems

A service air system and an instrument air system are provided to supply compressed air for general plant use and for operation of pneumatic instruments, valves and controllers. Redundancy is provided in compressors and receivers to ensure an air supply of suitable quantity, quality and pressure for plant operation.

1.2.2.8.12 Safety-Related Instrument Air Systems

A safety-related instrument air system is provided to continuously supply clean, dry, oil-free air for the initial charge and recharging of the automatic depressurization system (ADS) safety/relief valve accumulators when the depressurization function of the safety/relief valves is used. The “B” train safety-related instrument air system also provides postaccident makeup to the outboard MSIV air accumulators to ensure that accumulator air pressure remains above 45 psig for a period of seven (7) days after an accident. The system stores air at 160 to 170 psig in receiver tanks downstream of the purifier package. The volume of the air receiver tanks is designed to provide a sufficient quantity of air for recharging the ADS accumulators and outboard MSIV accumulators under accident conditions. In addition, the tanks contain a sufficient volume of air to provide makeup for system leakage for a period of 7 days after an accident occurs. After this initial 7 day period, the system can be recharged with the air compressors or commercially available compressed air bottles. Physically separate and redundant air lines are employed to distribute air at 150 psig to the ADS accumulators. The "B" train safety-related instrument air system also provides postaccident makeup to the Outboard MSIV air accumulators at an approximate setting of 85 psig (45 psig minimum). The unit utilizes one air compressor and purifier package for recharging the receiver tanks during normal operation.

The air system is safety-related except for the section between the air compressor and the dual isolation check valves of the air receiver tanks.

1.2.2.8.13 Fire Protection System

A fire protection system supplies fire fighting water to all areas inside and outside of the plant. Special fire protection systems are provided to protect hazardous areas.

1.2.2.8.14 Heating, Ventilating and Air Conditioning Systems

The plant heating, ventilating and air conditioning systems maintain suitable ambient temperatures for operating personnel and equipment throughout the plant. They also serve to control the flow and/or emission of airborne radioactivity.

1.2.2.8.15 Lighting System

Three lighting systems are employed: normal, fed from the unit auxiliary bus; essential, fed from the engineered safety features buses; and, emergency, fed from the Division 1 and Division 2 batteries. Essential lighting is used to supplement normal lighting to facilitate safe access and egress or the continuation of critical tasks. Emergency lighting is used in those pedestrian areas where lighting is required for safe personnel egress, for continuation of critical activities upon loss of all other light sources, or where a possible radiation hazard might endanger personnel safety.

Incandescent and Light Emitting Diodes (LED) light sources have been selected for the reactor building and other potentially high radiation areas.

1.2.2.8.16 Plant Communication System

Diverse communication systems and pathways are provided between selected plant areas, administrative office areas, the control room, and points offsite such as the system switching authority, local law enforcement, and emergency facilities.

Voice communication between various plant buildings and locations is provided by the intraplant communication system which provides a public address system and intercom system. Alarm and evacuation signals are broadcast over the intraplant public address system. A separate control room-to-plant communication system, provided for maintenance and instrument calibration, services most areas of the plant.

1.2.2.8.17 Alternate Decay Heat Removal System

The ADHR system provides additional decay heat removal options through a non-safety related alternate shutdown cooling system that can be used in MODE 4 and MODE 5 with the reactor depressurized and the reactor coolant system temperature ≤ 200°F.

1.2.2.9 Radioactive Waste Management Systems

1.2.2.9.1 Gaseous Radwaste System

The purpose of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the Radiologically Restricted Area does not exceed the maximum limits of the applicable 10 CFR regulations even with some defective fuel rods.

The offgases from the main condenser are the major source of gaseous radioactive waste. The treatment of these gases includes volume reduction through a catalytic hydrogen-oxygen recombiner, water vapor removal through a condenser, decay of short lived radioisotopes through a holdup line, further condensation and cooling, filtration, adsorption of isotopes on activated charcoal beds, further filtration through high efficiency filters, and final releases.

Continuous radiation monitors are provided which indicate radioactive release from the reactor and from the charcoal adsorbers. The radiation monitors are used to isolate the offgas system on high radioactivity, in order to prevent releasing gases of unacceptably high activity.

1.2.2.9.2 Liquid Radwaste System

The liquid radwaste system collects, monitors, treats, stores, and recycles or releases radioactive liquid wastes. These wastes are collected in sumps and drain tanks at various locations throughout the plant and then transferred to collection tanks in the radwaste facility for treatment, storage and recycle or release.

Wastes are processed on a batch basis with each batch being processed by methods appropriate for the quality and quantity of materials present. Most of the processed liquid is returned to the condensate system.

Equipment is selected, arranged and shielded to permit operation, inspection and maintenance within radiation allowances for personnel exposure. Processing equipment is selected and designed to require a minimum amount of maintenance.

Valving redundancy, instrumentation for detection, alarms of abnormal conditions, and procedural controls protect against the accidental discharge of liquid radioactive waste.

1.2.2.9.3 Solid Radwaste System

Solid radioactive wastes are collected, processed and packaged for storage prior to offsite shipment in approved shipping containers. Radwaste may be stored in various locations throughout the plant site as discussed in <Section 12.4.4.2>. Examples of these wastes are filter residue, spent resins and concentrated wastes.

Solid wastes originating from nuclear system equipment are stored for radioactive decay in the fuel storage pool and then prepared for offsite shipment. Examples of these wastes are spent fuel, spent control rods and incore ion chambers.

1.2.2.10 <u>Radiation Monitoring and Control</u>

1.2.2.10.1 Process Radiation Monitoring

Process radiation monitoring systems are provided to monitor and control radioactivity in process and effluent streams and to activate appropriate alarms and controls.

A process radiation monitoring system is provided for indicating and recording radiation levels associated with selected plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

Process radiation monitoring is also discussed in <Chapter 7>, <Chapter 9>, and <Chapter 11>.

1.2.2.10.2 Area Radiation Monitors

Radiation monitors are provided to detect abnormal radiation at various locations in the reactor building, turbine building, auxiliary building, radwaste, and fuel handling building. These monitors alarm locally and in the control room when abnormal radiation levels occur.

1.2.2.10.3 Site Environs Radiation Monitors

Radiation monitors are provided outside the plant buildings to monitor radiation levels. Data obtained are used to determine the plant contribution to onsite and offsite radiation levels.

1.2.2.11 Shielding

Shielding is provided throughout the plant, as required, to reduce radiation levels from direct and scattered radiation to dose rate levels well within the limits set in <10 CFR 20> and <10 CFR 100>. It is also designed to protect certain plant components from excess radiation damage or activation.

1.2.3 SYMBOLS USED IN ENGINEERING DRAWINGS

The symbols used in USAR figures that are applicable to the nuclear steam supply system (GE), are shown in <Figure 1.2-20> and <Figure 1.2-21>. The symbols used in USAR figures, that are applicable to the balance of plant are shown in <Figure 1.2-22> and <Figure 1.2-23>.

1.3 COMPARISON TABLES

The comparison tables represent information that was current at the time of FSAR submittal, September 1980. This information is historical in nature and is not updated to current plant design.

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This section highlights the principal design features of the plant and compares its major features with those of other boiling water reactor facilities. The design of this facility is based on proven technology obtained during the development, design, construction, and operation of boiling water reactors of similar types. The data, performance, characteristics, and other information presented here represent a firm design that was current September 1980.

1.3.1.1 Nuclear Steam Supply System Design Characteristics

<Table 1.3-1> summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to rated power output for a single plant unless otherwise noted.

1.3.1.2 Power Conversion System Design Characteristics

<Table 1.3-2> compares the power conversion system design characteristics.

1.3.1.3 Engineered Safety Features Design Characteristics

<Table 1.3-3> compares the engineered safety features design characteristics.

1.3.1.4 Containment Design Characteristics

<Table 1.3-4> compares the containment design characteristics.

1.3.1.5 Radioactive Waste Management Systems Design Characteristics

<Table 1.3-5> compares the radioactive waste management system's design characteristics.

1.3.1.6 Structural Design Characteristics

<Table 1.3-6> compares the structural design characteristics.

1.3.1.7 Electrical System Design Characteristics

<Table 1.3-7> compares the electrical system design characteristics.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

The significant changes that were made in the facility design from the last revision to the PSAR (Amendment 24, dated 7-18-75) to the submittal of FSAR are listed in <Table 1.3-8>. Each item listed in the table is cross referenced to the appropriate section of the FSAR.

TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS[1]

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
A. THERMAL AND HYDRAULIC DESIGN			
Rated power, MWt	3,579	3,833	2,436
Design power, MWt (ECCS design basis)	3,729	4,025	2,550
Steam flow rate, lb/hr	15.4×10^6	16.491×10^6	10.03×10^6
Core coolant flow rate, lb/hr	104.0×10^6	112.5×10^6	78.5×10^6
Feedwater flow rate, lb/hr	15.372×10^6	16.455×10^6	10.455×10^6
System pressure, nominal in steam dome, psia	1,040	1,040	1,020
Average power density, kW/liter	54.1	54.1	51.2
Maximum thermal output, kW/ft	13.4	13.4	13.4
Average thermal output, kW/ft	5.9	5.92	7.11
Maximum heat flux, Btu/hr-ft^2	361,600	362,000	428,300
Average heat flux, Btu/hr-ft^2	159,500	159,700	164,700
Maximum UO_2 temperature, °F	3,435	3,430	4,380

TABLE 1.3-1 (Continued)

		PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
A.	THERMAL AND HYDRAULIC DESIGN (Continued)			
	Average volumetric fuel temperature, °F	2,185	2,185	2,781
	Average cladding surface temperature, °F	565	558	558
	Minimum critical power ratio (MCPR)	1.20	1.23	1.9[(4)]
	Coolant enthalpy at core inlet, Btu/lb	527.7	527.9	526.2
	Core maximum exit voids within assemblies	79.0	76	79
	Core average exit quality, % steam	14.7	14.7	12.9
	Feedwater temperature, °F	420	420	387.4
	Design Power Peaking Factor			
	Maximum relative assembly power	1.40	1.40	1.40
	Local peaking factor	1.13	1.13	1.24

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
A. THERMAL AND HYDRAULIC DESIGN (Continued)			
Design Power Peaking Factor (Continued)			
Axial peaking factor	1.40	1.40	1.50
Total peaking factor	2.21	2.26	2.60
B. NUCLEAR DESIGN (First core)			
Water/UO_2 volume ratio (cold)	2.70	2.70	2.53
Reactivity with strongest control rod out, k_{eff}	<0.99	<0.99	<0.99
Dynamic void coefficient at core average voids, %, and	40.95	-41.31	38.0
rated output, ¢/%	-9.17	-7.14	-10.74
Fuel temperature doppler coefficient, end of cycle hot operating, ¢°C^{-1}	-0.412	-0.396	-0.403
Initial average U-235 enrichment wt. %	1.90	1.70	2.23
Initial cycle exposure, MWd/short ton	9,138	7,500	9,413 (Avg. first core)

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
C. CORE MECHANICAL DESIGN			
1. Fuel Assembly			
Number of fuel assemblies	748	800	560
Fuel rod array	8 x 8	8 x 8	7 x 7
Overall dimensions, in.	176	176	176
Weight of UO_2 per assembly, lb (pellet type)	457 (Chamfered)	458	483
Weight of fuel assembly, lb	697	699	681
2. Fuel Rods			
Number per fuel assembly	62	62	49
Outside diameter, in.	0.483	0.483	0.563
Cladding thickness, in.	0.032	0.032	0.032
Gap, pellet to cladding, in.	0.0045	.0045	.0060
Length of gas plenum, in.	10	10	16
Cladding material (free standing loading tubes)	Zircaloy-2	Zircaloy-2	Zircaloy-2

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
C. CORE MECHANICAL DESIGN (Continued)			
3. Fuel Pellets			
Material	UO_2	UO_2	UO_2
Density, % of theoretical	95	95	95
Diameter, in.	0.410	0.410	0.487
Length, in.	0.410	0.410	0.500
4. Fuel Channel			
Overall dimension, length, in.	167.36	166.9	166.9
Thickness, in.	0.120	0.120	0.080
Cross section dimensions, in.	5.455 x 5.455	5.46 x 5.46	5.44 x 5.44
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
5. Core Assembly			
Fuel weight as UO_2, lb	341,678	366,400	272,850
Core diameter (equivalent), in.	185.2	191.5	160.2

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
C. CORE MECHANICAL DESIGN (Continued)			
5. Core Assembly (Continued)			
Core height (active fuel), in.	150	150	144
6. Reactor Control System			
Method of variation of reactor power	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow
Number of movable control rods	177	193	137
Shape of movable control rods	Cruciform	Cruciform	Cruciform
Pitch of movable control rods	12.0	12.0	12.0
Control material in movable rods	B_4C granules compacted in SS tubes and/or B_4C capsules and hafnium metal rods in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
C. CORE MECHANICAL DESIGN (Continued)			
6. Reactor Control System (Continued)			
Type of control rod drives	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston
Type of temporary reactivity control for initial core	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison gadolinia-urania fuel
7. Incore Neutron Instrumentation			
Number of incore neutron detectors (fixed)	164	176	124
Number of incore detector assemblies	41	44	31
Total number of LPRM detectors	164	176	124
Number of incore LPRM penetrations	41	44	31
Number of LPRM detectors per penetration	4	4	4

TABLE 1.3-1 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
C. CORE MECHANICAL DESIGN (Continued)			
7. Incore Neutron Inst. (Continued)			
Number of SRM penetrations	4	6	4
Number of IRM penetrations	8	8	8
Total nuclear instrument penetrations	53	58	43
Source range monitor, range	4	Shutdown Through Criticality 6	4
Intermediate range monitor, range	8	Prior to Criticality to Low Power 8	8
Power range monitors, range		Approximately 1% Power to 15% Power	
Local power range monitors	164	176	124
Average power range monitors	8	8	6
Number and type of incore neutron sources	7 Sb-Be	7 Sb-Be	5 Sb-Be

TABLE 1.3-1 (Continued)

		PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
D.	REACTOR VESSEL DESIGN			
	Material	Low-alloy steel/ stainless clad	Low-alloy steel/ stainless clad	Carbon steel/ stainless clad
	Design pressure, psig	1,250	1,250	1,250
	Design temperature, °F	575	575	575
	Inside diameter, ft-in.	20-3/8	20-11	18-2
	Inside height, ft-in.	70-5	73	69-4
	Minimum base metal thickness (cylindrical section), in.	6.00	6.14	5.53
	Minimum cladding thickness, in.	1/8	1/8	1/8
E.	REACTOR COOLANT RECIRCULATION DESIGN			
	Number of recirculation loops	2	2	2
	Design pressure			
	Inlet leg, psig	1,250	1,250	1,148
	Outlet leg, psig	1,650(2) 1,550(3)	1,625(2) 1,525(3)	1,274(2)

TABLE 1.3-1 (Continued)

		PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
E.	REACTOR COOLANT RECIRCULATION DESIGN (Continued)			
	Design temperature, °F	575	575	562
	Pipe diameter, in.	24	24	28
	Pipe Material, ANSI	304/316	304/316	304/316
	Recirculation pump flow rate, gpm	42,000	44,900	42,200
	Number of jet pumps in reactor	20	24	20
F.	MAIN STEAMLINES			
	Number of steamlines	4	4	4
	Design pressure, psig	1,250	1,250	1,146
	Design temperature, °F	575	575	563
	Pipe diameter, in.	26	28	24
	Pipe material	Carbon steel	Carbon steel	Carbon steel

TABLE 1.3-1 (Continued)

NOTES:

[1] Parameters are related to rated power output for a single plant unless otherwise noted.

[2] Pump and discharge piping to, and including, discharge block valve.

[3] Discharge piping from discharge block valve to vessel.

[4] Minimum critical heat flux ratio (MCHFR).

TABLE 1.3-2

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Turbine generator <Section 10.2>			
Net generator output, MW	1,252[(1)]	1,306	626
Turbine cycle heat rate, Btu/KW-hr	9,770	10,029	9,602
Type/LSB length (line)	TC6F/43	TC6F/44	TC4F/28
Cylinders, No.	1-HP, 3-LP	1-HP, 3-LP	1-HP, 2-LP
Steam conditions at throttle valve			
Flow, lb/hr	14.68×10^6	15.54×10^6	8.29×10^6
Pressure, psia	965	965	965
Temperature, °F	540	540	510
Moisture content, %	0.40	0.51	0.40
Turbine cycle arrangement <Section 10.4>			
Steam reheat stages, No.	2	2	2
Feedwater heating stages, No.	6	6	6
Strings of feedwater heaters, No.	2-HP, 3-LP	2-HP, 3-LP	2
Heaters in condenser necks, No.	6	4	1
Heater drain system	Pumped forward	Pumped forward	Pumped forward
Condensate pumps, No.	3	3	3
Condensate booster pumps, No.	3	3	3

TABLE 1.3-2 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Turbine cycle arrangement (Continued)			
Heater drain pumps, No.	4	2	2
Reactor feed pumps, No.	3	2	2
Main steamline			
Steamlines, No.	4	4	4
Design pressure, psig	1,250	1,250	1,250
Design temperature, °F	575	575	575
Pipe diameter, in.	28	28	20
Pipe material	Carbon steel	Carbon steel	Carbon steel
Main steam bypass capacity, %	35	35	25
Final feedwater temperature, °F	420	420	420
Condenser <Section 10.4>			
Type	Multiple pressure	Multiple pressure	Single pressure
Condenser shells, No.	3	3	2
Design pressure, in. Hg abs	2.01/2.48/3.22	2.37/2.91/3.62	3.2
Total condenser duty, Btu/hr	8.1×10^9	8.506×10^9	4.25×10^9
Circulating water system <Section 10.4>			
Type	Closed/ND cooling tower	Closed/ND cooling tower	Closed/ND cooling tower

TABLE 1.3-2 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Circulating water system (Continued)			
Flow, gpm	555,000	571,000	376,000
Circulating water pumps, No.	3	2	2 (1/2 capacity)

NOTES:

[1] Original plant generator output. Net generator output after conversion to partial arc admission, incorporation of the 105% power uprate package, and replacement of the low pressure rotors is 1,277 MWe.

TABLE 1.3-3

COMPARISON OF ENGINEERED SAFETY FEATURES DESIGN CHARACTERISTICS

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
EMERGENCY CORE COOLING SYSTEMS (Systems sized on design power) <Section 6.3>			
1. Low Pressure Core Spray Systems			
Number of loops	1	1	2
Flow rate, gpm	6,000 at 122 psid	7,115 at 128 psid	4,625 at 120 psid
2. High Pressure Core Spray System			
Number of loops	1	1	1[(1)]
Flow rate, gpm	1,550 at 1,147 psid	1,650 at 1,147 psid	4,250
	6,000 at 200 psid	7,000 at 200 psid	
3. Automatic Depressurization System			
Number of relief valves	8	8	7

TABLE 1.3-3 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
EMERGENCY CORE COOLING SYSTEM (Continued)			
4. Low Pressure Coolant Injection[2]			
Number of loops	3	3	2
Number of pumps	3	3	4
Flow rate, gpm/pump	6,500 at 20 psid	7,450 at 24 psid	9,200 at 20 psid
5. Auxiliary Systems <Section 5.4> and <Section 9.2>			
6. Residual Heat Removal System			
Reactor shutdown cooling mode:			
Number of loops	2	2	2
Number of pumps	2	4	4
Flow rate, gpm/pump[3]	7,100	7,450	7,700

TABLE 1.3-3 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
EMERGENCY CORE COOLING SYSTEMS (Continued)			
6. Residual Heat Removal System (Continued)			
Duty, Btu/hr/heat exchanger(4)	46.9×10^6	50×10^6	32×10^6
Number of heat exchangers	2	2	2
Primary containment cooling mode:			
Flow rate, gpm	7,100(5)	7,450(5)	15,400(5)
7. Emergency Service Water System			
Flow rate, gpm	22,700 (total)	25,300 (total)	8,000
Number of pumps/unit	2/1	2/1	4
Flow rate, gpm/pump	2 @ 10,900/ 1 @ 900	2 @ 12,000/ 1 @ 1,300	See Note(6)

TABLE 1.3-3 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
8. Reactor Core Isolation Cooling System			
Flow rate, gpm	700 at 150-1,177 psig	800 at 1,120 psid	400 at 1,120 psid
9. Fuel Pool Cooling and Cleanup System			
Capacity, Btu/hr	26 x 10^6	12.5 x 10^6	5.7 x 10^6

NOTES:

(1) High pressure coolant injection system used.

(2) A mode of the RHR system.

(3) Capacity during reactor flooding mode with more than one pump running.

(4) Heat exchanger duty at 20 hours following reactor shutdown.

(5) Flow per heat exchanger.

(6) ESW system design is different and not readily correlated.

TABLE 1.3-4

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS[1]

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Type	Mark III, Steel containment, with pressure suppression, enclosed by reinforced concrete reactor building. Containment encloses drywell and suppression pool.	Mark III, Reinforced concrete containment, but with pressure suppression. Containment encloses drywell and suppression pool.	Mark II, Over-and-under primary containment, enclosed drywell and suppression pool. Enclosed by reactor building.
Leak rate, %/day	0.20	0.35	0.5
Containment			
Construction	Steel shell enclosed by reinforced concrete cylindrical structure (not prestressed) with hemispherical head.	Reinforced concrete cylindrical structure (not prestressed) with hemispherical head; steel lined.	Not applicable

TABLE 1.3-4 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Internal design temperature, °F	185	185	Not applicable
Design pressure, psig	+15, -0.8	15	Not applicable
Free (air) volume, ft^3	1.16×10^6	1.4×10^6 (excluding drywell)	Not applicable
Drywell			
Construction	Reinforced concrete. Basically cylindrical; flat concrete roof with a steel refueling head.	Reinforced concrete. Basically cylindrical; flat concrete roof with a steel refueling head.	Prestressed concrete. Drywell in frustum of a cone; steel lined.
Internal design temperature, °F	330	330	340
Design pressure, psig	+30, -21	30	+45, -2
Free (air) volume, total, ft^3	276,500	270,000	263,800

TABLE 1.3-4 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	BAILLY BWR 5
Suppression Pool			
Construction	Reinforced concrete, steel lined. Basically cylindrical.	Reinforced concrete, steel lined. Basically cylindrical.	Prestressed concrete. Pool is cylindrical; steel lined.
Internal design, temperature, °F	185	185	340
Design pressure, psig	15	15	+45, -2
Water volume, ft^3	120,000	136,000	73,500
Break area/total vent area	0.010	0.008	0.012

NOTE:

(1) Refer to <Chapter 3>.

TABLE 1.3-5

RADIOACTIVE WASTE MANAGEMENT SYSTEMS DESIGN CHARACTERISTICS

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
A. GASEOUS RADWASTE <Section 11.3>			
Design bases Noble gases μCi/sec	100,000 at 30 min	100,000 at 30 min	100,000 at 30 min
Process treatment	Recombiner, chilled charcoal	Chilled charcoal	Recombiner, ambient charcoal
Number of beds	8	8	12
Design condenser in-leakage, cfm	30	40	40
Release point-height above ground, ft	134	31.5	394
B. LIQUID RADWASTE <Section 11.2>			
Treatment of:			
1. Floor drains	F&D, E as req'd. R or D(1)	F,D,E and R	F,E, and R

TABLE 1.3-5 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
B. LIQUID RADWASTE (Continued)			
2. Equipment drains	F&D, E as req'd. R or D	F,D,E, and R	F,D, and R
3. Chemical drains	Neutralized, demineralized as req'd., recycled or discharged	Neutralized E, returned to equip. drain collector tank	F, discharged E, solid to radwaste
4. Laundry drains	F, if req'd., and discharged	None	Diluted and sent to circulating water discharge
5. Expected annual avg. release, μCi (excluding tritium)	500,000	110,000	20,000

TABLE 1.3-5 (Continued)

NOTE:

(1) Legend:

D - Demineralized
F - Filtered

R - Recycled, i.e., returned to condensate storage

TABLE 1.3-6

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

		PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
A.	SEISMIC DESIGN <Section 3.7>			
	Operating basis earthquake -			
	horizontal g	0.075	0.075	0.08
	vertical g	0.075	0.050	0.05
	(zero period)			
	Safe shutdown earthquake -			
	horizontal g	0.15	0.15	0.15
	vertical g	0.15	0.10	0.10
	(zero period)			
B.	WIND DESIGN <Section 3.3>			
	Maximum sustained - mph (at grade)	90	90	105
C.	TORNADOES			
	Translational - mph	70	60	60
	Tangential - mph	290	300	300

TABLE 1.3-7

COMPARISON OF ELECTRICAL SYSTEMS

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
Number of offsite circuits	5	first unit - 3 both units - 4	9
Number of auxiliary power sources	4 - 1 unit auxiliary transformer, 1 startup transformer (per unit)	3 service transformers (1 exclusively for ESF)	3-1 unit auxiliary transformer 1 - reserve auxiliary transformer 1 emergency reserve auxiliary transformer
Number of preferred power circuits of ESF buses	2	3	3
Number of ESF buses per unit	3	3	3
Number of standby ac power supplies	6 (1/ESF bus)	6 (1/ESF bus)	3 (1/ESF bus)

TABLE 1.3-7 (Continued)

	PERRY BWR 6 238-748	GRAND GULF BWR 6 251-800	HATCH 1 BWR 4 218-560
Number of 125V dc systems supplying buses	6 (1/ESF bus)	6 (1/ESF bus)	3 (1/ESF bus)
Sharing of standby power supplies and interconnections between safety buses	None	None	dc buses interconnected

TABLE 1.3-8

SIGNIFICANT DESIGN CHANGES FROM PSAR TO FSAR

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Nuclear fuel	The number of water rods in each fuel bundle has been changed from 1 to 2.	Improved fuel performance.	4.2.2, 4.3.4
Control Rod Drive Position Indication	Changed to 11 wire probe and solid state.	Improved reliability and increased frequency of checking actual rod position.	4.2
Feedwater Sparger	The thermal sleeve was changed to provide improved slip fit design of sparger to nozzle.	To eliminate failure, leakage and provide for possible inservice inspection.	5.3
RCIC System	Each component, except for the flow controller, has been made capable of functional testing.	Improved testability.	5.4.6
Automatic Depressurization System (ADS)	The interlocks on the automatic depressurization system were revised.	To meet IEEE Standard 279 requirements.	7.3.1

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Leak Detection System	The leak detection system was revised to upgrade the capability and incorporate the requirements of IEEE Standard 279. Added additional monitors to increase adequacy of detection.	To meet IEEE Standard 279 and <Regulatory Guide 1.45> requirements.	7.6.1
Control Rod Drive fast scram	Increased system pressure from 1,750 to 2,000 psi, enlarged insert/ withdraw draw lines, and increased accumulator volume to provide faster scram time.	Provides increased reactivity control, especially at end of fuel cycle. Provides increased thermal margin, and reduces amount of operation of steam relief.	3.9, 4.6
Reactor Recirc. pump trip	Pumps tripped on signals from turbine control or stop valves upon generator load rejection or turbine trip.	Reduces transient core flow and reactivity. Works with fast scram to provide increased thermal margin.	4.6.4, 5.4.1, 7.6.1
High Pressure Core Spray System	Changed logic for admission valve F-004 to close on occasion of high RPV water level only if drywell pressure is low.	NRC requirement.	7.3.1

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Reactor Protection System	Changes for control system instrument testability. Changed from switches to transmitters and added calibration units.	Provides improved testability reliability.	7.2.2
Ganged control rod withdrawal	Changed logic and control rod drive hydraulic system to move groups of control rods. Added stabilizing hydraulic valves.	Improves operating time for control maneuvering and startup.	3.9, 4.6
Reactor in-core monitors	Changed replacement from top to bottom of core monitor entry.	Improves time for replacement during outages.	7.6.1
Reactor Recirc. Pump	Added vibration sensors to record and alarm when high shaft vibration encountered on pump or motor.	Improves reliability.	Chapter 5
Reactor Recirc. Pump motor controls	Added motor-generator sets to provide control for reduced flow during startup and shutdown.	Provides improved operation.	7.7.1
Reactor Recirc. System	Removed pump bypass lines for reduction of region potentially sensitive to stainless steel stress corrosion problems.	Design improvement.	5.4.3

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Rod Block Monitor	Deleted sub-system from neutron monitor system	Design change.	5.6.1, 7.6
Instrument Line Con-tainment Isolation Valves	Excess flow check valves were re-placed with dual action solenoid valves and re-stricting ori-fices.	Design improvement.	6.2.4
Biological shield wall	Biological shield wall was filled with high density concrete.	To provide additional neutron and gamma shielding.	3.8.3
Suppression Pool Cleanup System	Added system.	To reduce doses inside containment.	6.2.7
Safety-Related Instrument Air System	Added system.	To support the air requirements of the ADS.	6.8
Combustible Gas Control System	Alternate type of hydrogen recombiner system used. Analyzer relocated outside containment.	To reduce plant cost with an equally qualified hydrogen recombiner system. Less severe operating environment.	6.2.5, 7.3.1
Spent Fuel Storage	Alternate type of spent fuel storage racks used.	High density spent fuel storage rack configuration used to increase onsite storage of spent fuel.	9.1.2

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Seismic loading on valves	Seismic loading requirement on valves lowered from 4.5g to 3.0g.	Piping systems have been designed to limit the accelerations to 3.0g.	3.9, 6.2.4
Insulation inside containment	Alternate type of insulation used.	Nu”K”on, fiberglass blanket insulation used instead of metal reflective to reduce plant cost and reduce heat loss into containment with an equally qualified product.	6.1.2, 6.2.2
18” diameter bypass in containment and drywell system	The containment and drywell purge system was modified to include an 18” bypass valve for use during normal operation. Also, the normal flow rate was reduced from 15,000 to 5,000 cfm through use of variable inlet vanes on the supply and exhaust fans.	To satisfy the requirements of Branch Technical Position CSB 6-4.	9.4.6, Figure 9.4-17

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Turbine Building and Heater Bay Ventilation	The exhaust system for the turbine building and heater bay was changed from roof ventilators in each area to a ducted exhaust system. The ducted system exhausts all areas through ducts which direct the exhaust to a single plenum which contains two centrifugal fans discharging through a single vent.	Improved radiation monitoring of exhaust air from the turbine building and the heater bay.	9.4.4, Figure 9.4-9
Change in safety classification	The safety classification of radwaste supply and exhaust systems (M31) was changed from Safety Class 3 to non-nuclear safety class.	The criteria of ANS-22 did not require this system to be Safety Class 3.	9.4.3, Figure 9.4-7

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Purge System Isolation Valves	The purge system isolation valve closing times were changed from 1 to 4 seconds for the valves isolating the drywell from the containment and from 2 seconds to 4 seconds for the valves isolating the containment from the outside.	Four-second closing time is in accordance with Branch Technical Position 6-4. Offsite releases would not exceed <10 CFR 100> guidelines.	9.4.6
Annulus Exhaust Gas Treatment System	The maximum discharge rate capability was revised from 650 to 2,000 cfm.	The change reflects the actual performance capability of the system and the revised discharge rate does not exceed the guidelines of <10 CFR 100>.	6.5.3, Figure 6.5-1
Annulus Exhaust Gas Treatment System	Delete the capability to automatically isolate the active filtering system and automatically start the standby filtering system on indication of high radiation in the exhaust from this system. Start both units automatically following an accident.	The change permitted the use of a non-safety-related radiation monitor and simplified the control system.	6.5.3, Figure 6.5-1

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Annulus Exhaust Gas Treatment System	Delete the automatic control of the electric heating coil from a humidistat. Energize the heating coil whenever the filter system operates.	The change permitted the use of nonsafety-related humidistats and simplified the control system.	7.3.1
Spent Fuel "Push-Pull" Ventilation System	The "push-pull" ventilation system serving the spent fuel pool was replaced by a system that supplies air around the pool periphery and exhausts it over the center of the pool.	The changes were made to accommodate ventilation requirements of refueling equipment.	9.4.2, Figure 9.4-4
Control Complex System Duct Changes	Modifications to portions of the duct distribution and exhaust system.	Duct system was modified to accommodate increased room heat loads and to eliminate smoke purge capability from the normal ventilation system. Smoke purge was subsequently provided with a separate system.	9.4.1, Figure 9.4-1, Figure 9.4-2
Containment Pool Supply and Exhaust System	The supply and exhaust system which developed air-flow across the containment fuel pool surface was eliminated.	The system was determined to not be necessary and its elimination will result in no safety hazard.	9.4.6, Figure 9.4-17

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Ice Protection	Addition of vertical concrete caissons around offshore intake and discharge structures.	To protect offshore structures from dynamic loads produced by floating ice islands crushing against the structures.	2.4.7, 3.8.4
Trash Racks	Trash racks have been eliminated from the offshore intake structures, although intakes have been constructed to allow for backfit if necessary.	Prevents the buildup of frazil ice at the intake structure ports. Low intake velocities will significantly reduced the possibility of debris entrapment.	2.4.7, 3.8.4
Class A Fill Permeability Coefficient and Groundwater Inflow Rate	Minimum Permeability Coefficient changed from $5x10^{-3}$ to $2x10^{-4}$ cm/sec and estimated inflow rate changed from 80 to 4 gpm.	4 gpm based upon field observations, thereby reducing required permeability coefficient.	2.5.4
Intake Structures	Three of shore intake heads have been changed to two heads.	Due to reduced flow requirements, two heads are adequate to supply the required flow and prevent the entrainment of fish due to intake water velocities.	2.4, 3.8, 9.2
Main Steam Relief Valves	Change in Supplier	Cost and quality advantages.	5.4.13

TABLE 1.3-8 (Continued)

Item	Change	Reason for Change	FSAR Section in Which Change is Discussed
Main Steam Flow Diverter	Introduction of special mechanical device to relieve biological shield wall annulus transient pressure.	Evolving analytical criteria, finalized after PSAR, required annulus venting for DBA.	6.2.1
Main Steam Relief Valve Discharge Piping and Suppression Pool Spargers (Quenchers)	Reclassification of discharge lines and spargers to Safety Class 3 from nonsafety.	Reevaluation of piping and quenchers established that, for plant safety, the piping and quenchers should be upgraded to Safety Class 3.	3.2
Quenchers in Suppression Pool	The rams head-type spargers were replaced by multi-arm spargers.	Improved steam suppression performance.	5.4.13, 3.8, 6.2

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The information in this Section is historical. That is, information originally provided in the Final Safety Analysis Report (FSAR) to meet the requirements of <10 CFR 50.34(b)> and was accurate at the time the plant was originally licensed, but is not intended to be updated for the life of the plant.

1.4.1 THE CLEVELAND ELECTRIC ILLUMINATING COMPANY - OWNER

The Cleveland Electric Illuminating Company (CEI) is engaged primarily in the production, transmission, distribution, and sale of electric energy for lighting, heating, cooling, and power purposes to residential, commercial and industrial customers.

The Company's service area is located in Northeastern Ohio in an area of approximately 1,700 square miles extending about 100 miles along the south shore of Lake Erie, west from Pennsylvania. In its area, CEI provides service for over 700,000 customers (1979).

The electric generating facilities of CEI include four fossil fuel plants located in Northeastern Ohio on the shore of Lake Erie. In addition, CEI owns 80 percent of a pumped-storage hydroelectric generating plant, the Seneca Power Plant, located in Warren County, Pennsylvania.

CEI is also the majority owner (51.38% in 1979) of Unit 1 of the Davis-Besse Nuclear Power Station, the first commercial nuclear power unit online in Ohio. The Toledo Edison Company owns the balance of Davis-Besse No. 1 and is the operator of the unit which is located nine miles west of Port Clinton, Ohio. Davis-Besse is a pressurized water reactor with 906,000 kilowatts of generating capacity.

CEI’s other nuclear participation includes part ownership in one additional unit at the Beaver Valley Power Station.

CEI has traditionally retained overall responsibility for the design, purchase, construction, and startup of its generating units. Through 1962, CEI performed all of the detailed design work with its own employees. Beginning in 1964 with the Seneca pumped-hydro plant, however, CEI elected to carry out power plant design employing the use of architect/engineer consultants with close supervision and participation by CEI's experienced power plant design engineers, with key decisions having been made by CEI. Further, the purchase of all plant equipment and the letting of contracts has always been done by CEI. As detailed elsewhere, construction of generating unit additions has been accomplished through individual subcontractors under the direction of CEI construction management employees.

CEI has also performed the startup and testing of its generating plant additions. This work was done by employees from the Steam Power Division and System Operation and Test Department with assistance from design engineering elements and the Production Engineering Section in the Civil and Mechanical Engineering Department.

1.4.2 GILBERT ASSOCIATES, INC. - ARCHITECT/ENGINEER

Gilbert Associates, Inc. (GAI), engineers and consultants, has been retained as the architect-engineer for the Perry Nuclear Power Plant.

Gilbert Associates, Inc., located in Reading, Pennsylvania, originally was known as W.S. Barstow and Company and was organized in 1906. The corporate name was changed to E.M. Gilbert Engineering Corporation in 1933, and in 1942, the corporate structure was revised and the name became Gilbert Associates, Inc. Gilbert Associates, Inc., and Commonwealth Associates, Inc., located in Jackson, Michigan, are part of The Gilbert/Commonwealth Companies (G/C). Commonwealth Associates originated in 1910 as Commonwealth Power Railway and Light Company. Commonwealth Associates was formed in 1949, and in 1973 was acquired by Gilbert Associates.

The collective experience and capabilities of The Gilbert/Commonwealth Companies offer complete consulting and engineering services to both investor-owned utilities and general industry, in such diverse fields as nuclear and conventional power generation, transmission, substation, and distribution systems, economic engineering, and management consulting service.

G/C is responsible for the design of many thermal generating units, both fossil and nuclear power. The Company's design experience includes one of the first reheat units, one of the first once through boiler units and one of the first supercritical steam pressure units. Individual unit designs range in ratings up to 1,280,000 kW and stations vary in complexity - nuclear, mine-mouth, closed cycle cooling tower, base-load, peaking, and others.

G/C has played an active and important role in the development of nuclear energy for private utilities, industry and governmental agencies. Projects include complete programs of nuclear power development involving analysis of sites, complete evaluations of proposals, contract and fuel program assistance, preparation of license applications, containment vessel design concepts, complete plant design, and procurement.

1.4.3 GENERAL ELECTRIC COMPANY - NUCLEAR STEAM SUPPLY SYSTEM

The General Electric Company (GE) has been awarded the contracts to design, fabricate and deliver the direct cycle boiling water nuclear steam supply system, to fabricate the first core of nuclear fuel and to provide technical direction of installation and startup of this equipment. GE has engaged in the development, design, construction, and operation of boiling water reactors since 1955. <Table 1.4-1> lists over 80 GE reactors that were completed, under construction or on order when the FSAR was originally submitted in January 1981. Thus, GE has substantial experience, knowledge and capability to design, manufacture and furnish technical assistance for the installation and startup of reactors.

1.4.4 RAYMOND KAISER ENGINEERS, INC.

Raymond Kaiser Engineers, Inc., was retained to provide construction management, expediting and quality assurance services.

Raymond Kaiser Engineers is a wholly owned subsidiary of Raymond International, Inc., and one of the major engineering and construction firms that has continuously served a wide range of clients in the aluminum, power, iron and steel, minerals, and other industries throughout the world since 1914.

In the nuclear field, Raymond Kaiser Engineers has provided services for private industry, as well as U.S. Government agencies in the areas of uranium processing, nuclear power and nuclear waste management. Raymond Kaiser Engineers has provided continuous construction and consulting services for the Department of Energy and its predecessor, the Atomic Energy Commission since 1950.

1.4.5 GENERAL ELECTRIC COMPANY - TURBINE GENERATOR VENDOR

The General Electric Company Large Steam Turbine Generator Department is the vendor for the Perry turbine generators and will provide technical direction and assistance in their installation and startup testing. The General Electric Company has been in the business of manufacturing and servicing steam turbine generators of all types and sizes since the early 1900's. Hundreds of large steam turbine generators have been built by the General Electric Company and are in service throughout the United States and the entire world.

The headquarters of the GE Large Steam Turbine Generator Department is in Schenectady, N.Y. In addition to having extensive manufacturing facilities in Schenectady, there are also numerous GE Research, Development and Test facilities including the Materials & Processes

Laboratory, the GE Research & Development Center, the Turbine Generator Development Laboratory, the Generator Test Balance Facility, and others.

General Electric had numerous nuclear steam turbine generator units in service, and on order, when the FSAR was originally submitted in January 1981 for both boiling water reactors (as installed at the Perry Plant), and pressurized water reactors. Twenty-one nuclear steam turbine-generator units were in service and operating with boiling water reactors, while twelve units were in service and operating with pressurized water reactors. In addition, 30 turbine generators were on order for boiling water reactors, and 45 turbine generators were on order for pressurized water reactors.

The first General Electric nuclear steam turbine generator went into service at the Commonwealth Edison Dresden Unit 1 station in April 1960. Since that time, a wealth of operating experience has been gained on the Dresden No. 1 and subsequent units.

1.4.6 NUS CORPORATION - ENVIRONMENTAL CONSULTANT

The NUS Corporation was retained as the environmental consultant for the Perry Project. NUS provides support to the Perry Project in the areas of land use and demography, meteorology, hydrology, noise, ecology, and radiological impact assessment.

NUS Corporation (NUS), an engineering and environmental consulting firm, provides professional services to industry, utilities and government in the areas of energy management, fossil and nuclear energy systems, environmental engineering, pollution control, training, water and wastewater management, and mining consulting. Corporate headquarters are located in Rockville, Maryland.

1.4.7 OTHER CONSULTANTS

Several consultants were retained to supplement CEI personnel in a variety of disciplines. Each consultant provided a relatively narrow scope of service, but collectively they provided a significant, well-qualified work force. Other consultants may be retained from time to time throughout the life of the project.

TABLE 1.4-1

COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION OR IN DESIGN BY GENERAL ELECTRIC

STATION	UTILITY	RATING (MWe)	YEAR OF ORDER	YEAR OF STARTUP
Dresden 1	Commonwealth Edison	207	1955	1960
Humboldt Bay	Pacific G&E	70	1958	1963
Kahl	Germany	15	1958	1961
Garigliano	Italy	150	1959	1964
Big Rock Point	Consumers Power	72	1959	1963
JPDR	Japan	11	1960	1963
KRB	Germany	237	1962	1967
Tarapur 1	India	190	1962	1969
Tarapur 2	India	190	1962	1969
GKN	Holland	52	1963	1968
Oyster Creek	JCP&L	640	1963	1969
Nine Mile Point 1	Niagara Mohawk	610	1963	1970
Dresden 2	Commonwealth Edison	794	1965	1970
Pilgrim	Boston Edison	670	1965	1972
Millstone 1	NUSCO	652	1965	1971
Tsuruga	Japan	340	1965	1970
Nuclenor	Spain	440	1965	1971
Fukushima 1	Japan	439	1966	1971
BKW KKM	Switzerland	306	1966	1972
Dresden 3	Commonwealth Edison	794	1966	1971
Monticello	Northern States	548	1966	1971
Quad Cities 1	Commonwealth Edison	789	1966	1972
Browns Ferry 1	TVA	1,067	1966	1974
Browns Ferry 2	TVA	1,067	1966	1975
Quad Cities 2	Commonwealth Edison	789	1966	1972
Vermont Yankee	Vermont Yankee	515	1966	1972
Peach Bottom 2	Philadelphia Electric	1,065	1966	1974
Peach Bottom 3	Philadelphia Electric	1,065	1966	1974
Fitzpatrick	PASNY	821	1968	1975
Shoreham	LILCO	820	1967	1984
Cooper	Nebraska PPD	778	1967	1974
Browns Ferry 3	TVA	1,067	1967	1977
Limerick 1	Philadelphia Electric	1,100	1967	1984
Hatch 1	Georgia	786	1967	1975
Fukushima 2	Japan	762	1967	1974
Brunswick 1	Carolina P&L	821	1968	1977
Brunswick 2	Carolina P&L	821	1968	1975
Arnold	Iowa	545	1968	1975
Fermi 2	Detroit Edison	1,093	1968	1985

TABLE 1.4-1 (Continued)

STATION	UTILITY	RATING (MWe)	YEAR OF ORDER	YEAR OF STARTUP
Limerick 2	Philadelphia Electric	1,100	1967	See Note[1]
Hope Creek 1	PSE&G	1,067	1969	1986, est.
Chinshan	Taiwan	610	1969	1978
Caorso 1	Italy	822	1969	1977
Hatch 2	Georgia	786	1970	1978
La Salle 1	Commonwealth Edison	1,078	1970	1982
La Salle 2	Commonwealth Edison	1,078	1970	1983
Susquehanna 1	Pennsylvania P&L	1,050	1967	1982
Susquehanna 2	Pennsylvania P&L	1,050	1968	1984
Chinshan 2	Taiwan	610	1970	1979
Hanford 2	WPPSS	1,100	1971	1984
Nine Mile Point 2	Niagara Mohawk	1,100	1971	1985, est.
Grand Gulf 1	Mississippi P&L	1,250	1971	1982
Grand Gulf 2	Mississippi P&L	1,250	1971	See Note[1]
Kaiseraugst	Switzerland	915	1971	See Note[1]
Fukushima 6	Japan	1,135	1971	1979
Tokai 2	Japan	1,135	1971	1977
Riverbend 1	Gulf States	940	1972	1985, est.
Perry 1	Cleveland Electric	1,205[3]	1972	1985
Perry 2	Cleveland Electric	1,205	1972	See Note[2]
Laguna Verde 1	Mexico	660	1972	1986, est.
Leibstadt	Switzerland	940	1972	1984
Kuosheng 1	Taiwan	992	1972	1981
Kuosheng 2	Taiwan	992	1972	1982
Clinton 1	Illinois Power	950	1973	1986, est.
Laguna Verde 2	Mexico	660	1973	1988, est.
Alto Lazio 1	Italy	982	1974	1989, est.
Alto Lazio 2	Italy	982	1974	1990, est.

NOTES:

[1] Not Available
[2] Under Evaluation
[3] Original plant generator output. Net generator output after conversion to partial arc admission, incorporation of the 105% power uprate package, and replacement of the low pressure rotors is 1,277 MWe.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 CURRENT DEVELOPMENT PROGRAMS

1.5.1.1 Instrumentation for Vibration

Vibration testing for reactor internals has been performed on virtually all GE-BWR plants. At the time of issue of <Regulatory Guide 1.20>, test programs for compliance were instituted. The first BWR 6 plant of each size is considered a prototype design and is instrumented and subjected to both cold and hot, two phase flow testing to demonstrate that flow induced vibrations, similar to those expected during operation, will not cause damage. Subsequent plants which have internals similar to those of the prototypes are tested in compliance to the requirements of <Regulatory Guide 1.20> to confirm the adequacy of the design with respect to vibration. Since Perry is the prototype of the Standard 238 size plant, it has been subjected to the prototype test program discussed in <Section 3.9.2>.

1.5.1.2 Core Spray Distribution

GE has performed a program to study BWR 6 core spray distributions using a combination of single nozzle steam and air tests, single and multiple nozzle analytical models, and full scale air tests. This methodology has been confirmed by a full scale 30° sector steam test as described in NEDO-24712 Core Spray Design Methodology Confirmation Test, August 1979. In a letter from Tedesco to Sherwood, January 30, 1981, the NRC concluded the tests documented in NEDO-24712, "constitute an adequate confirmation of the GE spray distribution methodology for BWR/6 type sprayers."

1.5.1.3 Core Spray and Core Flooding Heat Transfer Effectiveness

Due to the incorporation of an 8 x 8 fuel rod array with unheated "water rods," tests have been conducted to demonstrate the effectiveness of ECCS in the new geometry.

These tests are regarded as confirmatory only, since the geometry change is very slight and the "water rods" provide an additional heat sink in the inside of the bundle which improves heat transfer effectiveness.

There are two distinct programs involving the core spray. Testing of the core spray distribution has been accomplished, and the Licensing Topical Report NEDO-10846, "BWR Core Spray Distribution," has been submitted. The other program concerns the testing of core spray and core flooding heat transfer effectiveness. The results of testing, with stainless steel cladding, were reported in the Licensing Topical Report NEDO-10801, "Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness." The results of testing, using Zircaloy cladding, were reported in the Licensing Topical Report NEDO-20231, "Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy Clad, 8 x 8 Simulated BWR Fuel Bundle."

1.5.1.4 Verification of Pressure Suppression Design

The General Electric Company has conducted a large scale test program to verify the performance characteristics of the Mark III containment. Large scale testing was started in November 1973 following completion of a 2-year small scale test program.

The large scale test program utilizes a facility which represents a segment of a Mark III containment. The original character of the programs was to be a confirmatory exercise to verify the short term analytical model. The scope of the total program included testing beyond design basis conditions to investigate the margins available in pressure suppression systems. As a result of this testing, GE proposed a new analytical model to evaluate the Mark III design. This model is entitled "The General Electric Mark III Pressure Suppression Containment System Analytical Model," and is described in NEDO-20533.

During early tests, it was observed that containment structures could be subject to significant suppression pool hydrodynamic loads during blowdown. This resulted in several additional test series whose objective was to generate design basis loads to be incorporated in the design of the affected containment structures.

Sixteen large scale test series have been completed to date. The primary objective of three series of these tests was to verify short term analytical models for horizontal vents and centerline submergences. The objectives of two others were to obtain scoping data regarding pool dynamic response and impact loads on structures located above the suppression pool. Other tests were designed to measure froth impingement loads on the hydraulic control unit floor and to determine pool swell motion characteristics, to measure pool impact loads on representative containment structures, and to determine pool motion characteristics for large air mass fraction vent flows, and to compare these scale results to the previous full scale air tests.

Additional tests have been conducted to indicate comparability of liquid blowdown to steam blowdowns and to investigate pool stratification and vent chugging effects.

Tests have been performed with the suppression pool at an initial elevated temperature to determine steam condensation characteristics under such conditions. A multivent series have been run to consider possible vent interactions. The remaining Mark III testing was confirmatory in nature and was completed prior to the first operating license for a Mark III plant.

1.5.1.5 Critical Heat Flux Testing

A program for critical heat flux testing was established and was to be similar to that described in the report APED-5286, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors." Since that time, however, a new analysis has been performed and the GETAB program initiated. The results of that analysis and related testing is described in the approved Licensing Topical Report, NEDO-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application."

1.6 REFERENCE MATERIALS

Document references used in the development of the Updated Safety Analysis Report can be found within the chapter reference sections. These reference sections are normally found at the end of the chapter prior to the tables.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The actual drawings specified in the sections that follow were provided separately to the NRC in conjunction with the original submittal of the FSAR. The drawings and other detailed information represents information that was current at the time of the FSAR submittal, September 1980. This information is historical in nature and is not updated to current plant design.

1.7.1 ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

<Table 1.7-1> consists of a listing of electrical, instrumentation and control drawings that were considered necessary for the evaluation of safety-related features discussed in <Chapter 7> and <Chapter 8> of the USAR. In cases where these drawings are provided in the USAR, a cross-reference to the USAR figure number and current revision is shown. Whenever revised drawings are provided to the NRC, <Table 1.7-1> will be updated by Revision to reflect drawing revisions.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

<Table 1.7-2> consists of a list of piping and instrumentation diagrams provided in the USAR. GE and CEI drawing numbers are cross-referenced with their corresponding USAR figure numbers.

1.7.3 OTHER DETAILED INFORMATION

Other detailed information will be provided as requested by the NRC staff.

TABLE 1.7-1

LISTING OF ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

A. General Electric Drawings

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
105D5116	(1)	7	5/19/78	Leak Detection System
105D5116	(2)	7	5/19/78	Leak Detection System
105D5116	(3)	7	5/19/78	Leak Detection System
127D1780CA	(1)	0	1/14/77	Remote Shutdown System
127D1780CA	(2)	0	1/14/77	Remote Shutdown System
127D1780CA	(3)	0	1/14/77	Remote Shutdown System
127D1780CA	(4)	0	1/14/77	Remote Shutdown System
127D1780CA	(5)	0	1/14/77	Remote Shutdown System
131C7911	(1)	7	4/12/78	Nuclear Boiler System
131C7911C	(1)	4	10/05/76	Nuclear Boiler System
762E260	(1)	3	7/27/80	MSIV Leakage Control System
762E294BA	(1)	0	3/01/76	Low Pressure Core Spray System
762E294BA	(2)	0	3/01/76	Low Pressure Core Spray System
762E297BA	(1)	2	2/08/80	Reactor Core Isolation Cooling System
762E297BA	(2)	2	2/08/80	Reactor Core Isolation Cooling System
762E297BA	(3)	2	2/08/80	Reactor Core Isolation Cooling System
762E297BA	(4)	2	2/08/80	Reactor Core Isolation Cooling System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
762E297BA	(5)	2	2/08/80	Reactor Core Isolation Cooling System
762E419CA	(1)	0	7/22/77	Process Radiation Monitoring System
762E419CA	(2)	0	7/22/77	Process Radiation Monitoring System
762E419CA	(3)	0	7/22/77	Process Radiation Monitoring System
762E421CA	(1)	6	1/22/79	Process Diagram RCIC System
762E293	(1)	7	5/19/78	Leak Detection System
762E293	(2)	7	5/19/78	Leak Detection System
762E293	(3)	7	5/19/78	Leak Detection System
762E293	(3A)	7	5/19/78	Leak Detection System
762E293	(4)	7	5/19/78	Leak Detection System
762E425CA	(1)	12A	10/02/93	Residual Heat Removal System
762E425CA	(2)	11A	10/02/93	Residual Heat Removal System
762E425CA	(3)	2	10/15/87	Residual Heat Removal System
762E426BA	(1)	2	4/16/76	Residual Heat Removal System
762E426BA	(2)	2	4/16/76	Residual Heat Removal System
762E426BA	(3)	2	4/16/76	Residual Heat Removal System
762E426BA	(4)	2	4/16/76	Residual Heat Removal System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
762E426BA (5)	2	4/16/76	Residual Heat Removal System
762E427BA (1)	1	3/02/78	Reactor Protection System
762E427BA (2)	1	3/02/78	Reactor Protection System
762E427BA (3)	1	3/02/78	Reactor Protection System
762E427BA (4)	1	3/02/78	Reactor Protection System
762E434 (1)	8	4/15/78	Standby Liquid Control System
762E455 (1)	6	7/30/79	High Pressure Core Spray System
762E467C (1)	3	4/05/77	Low Pressure Core Spray System
828E226CA (1) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (2) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (2A) (B-208-011)	3	11/08/76	Auto Depressurization System
828E226CA (3) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (4) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (5) (B-208-011)	7	6/18/79	Auto Depressurization System
828E226CA (6) (B-208-011)	7	6/18/79	Auto Depressurization System
828E226CA (7) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (8) (B-208-011)	6	10/04/78	Auto Depressurization System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E226CA (9) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (10) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (11) (B-208-011)	6	10/04/78	Auto Depressurization System
828E226CA (12) (B-208-011)	7	6/18/79	Auto Depressurization System
828E226CA (13) (B-208-011)	6	10/04/78	Auto Depressurization System
828E226CA (14) (B-208-011)	7	6/18/79	Auto Depressurization System
828E226CA (15) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (16) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (17) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (18) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (19) (B-208-011)	9	1/31/80	Auto Depressurization System
828E226CA (20) (B-208-011)	8	10/26/79	Auto Depressurization System
828E226CA (21) (B-208-011)	9	1/31/80	Auto Depressurization System
828E234CA (1)	2	1/31/80	Standby Liquid Control System
828E234CA (2)	2	1/31/80	Standby Liquid Control System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E234CA (3)	2	1/31/80	Standby Liquid Control System
828E234CA (4)	2	1/31/80	Standby Liquid Control System
828E238CJ (1) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (2) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (3) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (4) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (5) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (6) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (7) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (8) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (9) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (10) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (11) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (12) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (13) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E238CJ (14) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (15) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (16) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (17) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (18) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (19) (B-208-038)	2	11/17/76	Power Range Neutron Monitoring System
828E238CJ (20) (B-208-038)	2	11/17/76	Power Range Neutron Monitoring System
828E238CJ (21) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (22) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (23) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (24) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (25) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (26) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (27) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (28) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E238CJ (29) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (30) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (31) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (32) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (33) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (34) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (35) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (36) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (37) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (38) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (39) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (40) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (41) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (42) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (43) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E238CJ (44) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (45) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (46) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (47) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (48) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (49) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (50) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (51) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (52) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (53) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (54) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E238CJ (55) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (56) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (57) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System
828E238CJ (58) (B-208-038)	6	8/13/79	Power Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E238CJ (59) (B-208-038)	7	2/08/80	Power Range Neutron Monitoring System
828E239CA (1) (B-208-039)	6	1/31/80	Remote Shutdown System
828E239CA (2) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (3) (B-208-039)	4	6/19/79	Remote Shutdown System
828E239CA (4) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (5) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (6) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (7) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (8) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (9) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (10) (B-208-039)	3	10/02/78	Remote Shutdown System
828E239CA (11) (B-208-039)	5	10/26/79	Remote Shutdown System
828E243CA (1) (B-208-054)	6	10/26/79	Process Radiation Monitoring System
828E243CA (2) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (3) (B-208-054)	5	10/26/79	Process Radiation Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E243CA (4) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (5) (B-208-054)	4	6/22/79	Process Radiation Monitoring System
828E243CA (6) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (7) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (8) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (9) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (10) (B-208-054)	6	10/26/79	Process Radiation Monitoring System
828E243CA (11) (B-208-054)	6	6/22/79	Process Radiation Monitoring System
828E243CA (12) (B-208-054)	3	10/24/79	Process Radiation Monitoring System
828E243CA (13) (B-208-054)	3	10/24/79	Process Radiation Monitoring System
828E243CA (14) (B-208-054)	3	10/24/79	Process Radiation Monitoring System
828E243CA (15) (B-208-054)	3	10/24/79	Process Radiation Monitoring System
828E243CA (16) (B-208-054)	4	6/22/79	Process Radiation Monitoring System
828E243CA (17) (B-208-054)	4	6/22/79	Process Radiation Monitoring System
828E243CA (18) (B-208-054)	3	10/24/79	Process Radiation Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E278 (1)	2	5/31/78	Remote Shutdown System
828E278 (2)	2	5/31/78	Remote Shutdown System
828E443CA (1) (B-208-010)	6	1/31/80	Nuclear Boiler Process Instrumentation
828E443CA (2) (B-208-010)	6	1/31/80	Nuclear Boiler Process Instrumentation
828E443CA (3) (B-208-010)	4	6/19/79	Nuclear Boiler Process Instrumentation
828E443CA (4) (B-208-010)	BB	6/19/79	Nuclear Boiler Process Instrumentation
828E443CA (5) (B-208-010)	4	6/19/79	Nuclear Boiler Process Instrumentation
828E445CA (1) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (1A) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (1B) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (2) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (3) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (3A) (B-208-013)	9	1/28/80	Nuclear Steam Supply System Shutoff
828E445CA (4) (B-208-013)	8	10/26/79	Nuclear Steam Supply Shutoff System
828E445CA (5) (B-208-013)	9	10/26/79	Nuclear Steam Supply Shutoff System
828E445CA (6) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E445CA (7) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (8) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (9) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (10) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (11) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (12) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (13) (B-208-013)	9	6/22/78	Nuclear Steam Supply Shutoff System
828E445CA (14) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (15) (B-208-013)	9	1/28/80	Nuclear Steam Supply Shutoff System
828E445CA (16) (B-208-013)	2	1/01/77	Nuclear Steam Supply Shutoff System
828E445CA (17) (B-208-013)	9	6/22/79	Nuclear Steam Supply Shutoff System
828E446CA (1) (B-208-015)	6	10/29/79	Reactor Recirculation System
828E446CA (2) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (3) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (4) (B-208-015)	4	6/22/79	Reactor Recirculation System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E446CA (5) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (6) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (7) (B-208-015)	5	10/29/79	Reactor Recirculation System
828E446CA (8) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (9) (B-208-015)	5	10/29/79	Reactor Recirculation System
828E446CA (10) (B-208-015)	5	10/29/79	Reactor Recirculation System
828E446CA (11) (B-208-015)	6	10/29/79	Reactor Recirculation System
828E446CA (12) (B-208-015)	U	6/22/79	Reactor Recirculation System
828E446CA (13) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (14) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (15) (B-208-015)	6	10/29/79	Reactor Recirculation System
828E446CA (16) (B-208-015)	K	6/22/79	Reactor Recirculation System
828E446CA (17) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (18) (B-208-015)	6	10/29/79	Reactor Recirculation System
828E446CA (19) (B-208-015)	3	10/12/79	Reactor Recirculation System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E446CA (20) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (21) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (22) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (23) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (24) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (25) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (26) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (27) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (28) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (29) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E446CA (30) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (31) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (32) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (33) (B-208-015)	4	6/22/79	Reactor Recirculation System
828E446CA (34) (B-208-015)	4	6/22/79	Reactor Recirculation System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E446CA (35) (B-208-015)	6	6/22/79	Reactor Recirculation System
828E525CA (1) (B-208-025)	GG	10/26/79	Feedwater Control System
828E525CA (2) (B-208-025)	FF	10/26/79	Feedwater Control System
828E525CA (3) (B-208-025)	FF	10/26/79	Feedwater Control System
828E525CA (4) (B-208-025)	Z	6/18/79	Feedwater Control System
828E525CA (5) (B-208-025)	HH	10/26/79	Feedwater Control System
828E525CA (6) (B-208-025)	Y	10/26/79	Feedwater Control System
828E525CA (7) (B-208-025)	EE	10/26/79	Feedwater Control System
828E525CA (8) (B-208-025)	EE	10/26/79	Feedwater Control System
828E531CA (1) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (2) (B-208-040)	2	10/13/77	Reactor Protection System
828E531CA (3) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (4) (B-208-040)	4	6/22/79	Reactor Protection System
828E531CA (5) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (6) (B-208-040)	6	1/22/80	Reactor Protection System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E531CA (7) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (8) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (9) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (10) (B-208-040)	4	6/22/79	Reactor Protection System
828E531CA (11) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (12) (B-208-040)	4	1/22/79	Reactor Protection System
828E531CA (13) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (14) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (15) (B-208-040)	4	6/22/79	Reactor Protection System
828E531CA (16) (B-208-040)	6	1/22/80	Reactor Protection System
828E531CA (16A) (B-208-040)	2	10/13/77	Reactor Protection System
828E531CA (17) (B-208-040)	4	6/22/79	Reactor Protection System
828E531CA (18) (B-208-040)	4	6/22/79	Reactor Protection System
828E534CA (1) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (2) (B-208-055)	2	10/13/77	Residual Heat Removal System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E534CA (3) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (4) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (5) (B-208-055)	5	10/29/79	Residual Heat Removal System
828E534CA (5A) (B-208-055)	5	10/29/79	Residual Heat Removal System
828E534CA (6) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (7) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (8) (B-208-055)	4	6/22/79	Residual Heat Removal System
828E534CA (9) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (10) (B-208-055)	6	6/22/79	Residual Heat Removal System
828E534CA (11) (B-208-055)	7	6/22/79	Residual Heat Removal System
828E534CA (12) (B-208-055)	7	6/22/79	Residual Heat Removal System
828E534CA (13) (B-208-055)	5	10/29/79	Residual Heat Removal System
828E534CA (14) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (15) (B-208-055)	6	10/29/79	Residual Heat Removal System
828E534CA (16) (B-208-055)	1	11/30/76	Residual Heat Removal System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E534CA (17) (B-208-055)	6	10/29/79	Residual Heat Removal System
828E534CA (17A) (B-208-055)	7	10/29/79	Residual Heat Removal System
828E534CA (18) (B-208-055)	4	6/22/79	Residual Heat Removal System
828E534CA (19) (B-208-055)	7	6/22/79	Residual Heat Removal System
828E534CA (20) (B-208-055)	6	6/22/79	Residual Heat Removal System
828E535CA (1) (B-208-060)	6	10/26/79	Low Pressure Core Spray System
828E535CA (1A) (B-208-060)	7	10/26/79	Low Pressure Core Spray System
828E535CA (2) (B-208-060)	6	6/19/79	Low Pressure Core Spray System
828E535CA (3) (B-208-060)	5	7/25/79	Low Pressure Core Spray System
828E535CA (3A) (B-208-060)	6	10/26/79	Low Pressure Core Spray System
828E535CA (4) (B-208-060)	8	6/19/79	Low Pressure Core Spray System
828E535CA (5) (B-208-060)	8	6/19/79	Low Pressure Core Spray System
828E535CA (6) (B-208-060)	8	6/19/79	Low Pressure Core Spray System
828E535CA (7) (B-208-060)	8	6/19/79	Low Pressure Core Spray System
828E536CA (1) (B-208-065)	5	1/31/80	High Pressure Core Spray System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E536CA (1A) (B-208-065)	4	1/31/80	High Pressure Core Spray System
828E536CA (2) (B-208-065)	3	6/19/79	High Pressure Core Spray System
828E536CA (3) (B-208-065)	5	6/19/79	High Pressure Core Spray System
828E536CA (4) (B-208-065)	4	1/31/80	High Pressure Core Spray System
828E536CA (5) (B-208-065)	4	1/31/80	High Pressure Core Spray System
828E536CA (6) (B-208-065)	4	1/31/80	High Pressure Core Spray System
828E536CA (7) (B-208-065)	3	6/19/79	High Pressure Core Spray System
828E539CA (1) (B-208-075)	6	1/31/80	Reactor Core Isolation Cooling System
828E539CA (2) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (3) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (4) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (5) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (6) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (7) (B-208-075)	5	1/31/80	Reactor Core Isolation System
828E539CA (8) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
828E539CA (9) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (10) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (11) (B-208-075)	6	1/31/80	Reactor Core Isolation Cooling System
828E539CA (12) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (13) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (14) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (15) (B-208-075)	5	1/31/80	Reactor Core Isolation Cooling System
828E539CA (16) (B-208-075)	6	1/31/80	Reactor Core Isolation Cooling System
851E478 (1) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (2) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (3) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (4) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (5) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (6) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (7) (B-208-020)	3	9/24/77	Rod Control & Information System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E478 (7A) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (8) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (9) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (10) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (11) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (12) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (13) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (14) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (15) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (16) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (17) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478 (18) (B-208-020)	3	9/24/77	Rod Control & Information System
851E478CA (1)	3	1/31/80	Rod Control & Information System
851E478CA (2)	3	1/31/80	Rod Control & Information System
851E478CA (3)	3	1/31/80	Rod Control & Information System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E478CA (4)	2	10/26/79	Rod Control & Information System
851E478CA (5)	2	10/26/79	Rod Control & Information System
851E478CA (6)	3	1/31/80	Rod Control & Information System
851E478CA (7)	2	10/26/79	Rod Control & Information System
851E478CA (8)	2	10/26/79	Rod Control & Information System
851E478CA (9)	2	10/26/79	Rod Control & Information System
851E478CA (10)	0	10/27/78	Rod Control & Information System
851E478CA (11)	3	1/31/80	Rod Control & Information System
851E478CA (12)	3	1/31/80	Rod Control & Information System
851E567 (1) (B-208-025(200)) <Figure 7.7-6 (1)>	M	5/24/99	Feedwater Control System
851E567 (2) (B-208-025(201)) <Figure 7.7-6 (2)>	D	9/24/91	Feedwater Control System
851E602 (1) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (2) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (3) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (4) (B-208-070)	4	7/20/77	Leak Detection System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E602 (5) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (6) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (7) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (8) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (9) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (10) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (11) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (12) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (13) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (14) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (15) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (16) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (17) (B-208-070)	4	7/20/77	Leak Detection System
851E602 (18) (B-208-070)	4	7/20/77	Leak Detection System
851E602CA (1)	3	1/28/80	Leak Detection System
851E602CA (2)	2	10/26/79	Leak Detection System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E602CA (3)	3	1/28/80	Leak Detection System
851E602CA (4)	3	1/28/80	Leak Detection System
851E602CA (5)	2	10/26/79	Leak Detection System
851E602CA (6)	2	10/26/79	Leak Detection System
851E602CA (7)	3	1/28/80	Leak Detection System
851E602CA (8)	3	1/28/80	Leak Detection System
851E602CA (9)	3	1/28/80	Leak Detection System
851E602CA (10)	3	1/28/80	Leak Detection System
851E602CA (11)	3	1/28/80	Leak Detection System
851E602CA (12)	3	1/28/80	Leak Detection System
851E602CA (13)	3	1/28/80	Leak Detection System
851E602CA (14)	3	1/28/80	Leak Detection System
851E602CA (15)	3	1/28/80	Leak Detection System
851E602CA (16)	3	1/28/80	Leak Detection System
851E884 (1) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (2) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (3) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (4) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (5) (B-208-037)	8	10/26/79	Startup Range Neutron Monitoring System
851E884 (6) (B-208-037)	8	10/26/79	Startup Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E884 (7) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (8) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (9) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (10) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (11) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (12) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (13) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (14) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (15) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (16) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (17) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (18) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (19) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (20) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (21) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
851E884 (22) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (23) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (24) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (25) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (26) (B-208-037)	8	10/26/79	Startup Range Neutron Monitoring System
851E884 (27) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (28) (B-208-037)	9	1/28/80	Startup Range Neutron Monitoring System
851E884 (29) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (30) (B-208-037)	7	7/30/79	Startup Range Neutron Monitoring System
851E884 (31) (B-208-037)	8	10/26/79	Startup Range Neutron Monitoring System
851E884 (32) (B-208-037)	8	10/26/79	Startup Range Neutron Monitoring System
851E892BA (1)	0	12/18/75	High Pressure Spray System
851E892BA (2)	0	12/18/75	High Pressure Spray System
851E892BA (3)	1	12/10/77	High Pressure Spray System
865E338CA (1) (B-208-071)	5	1/31/80	MSIV Leakage Control System
865E338CA (2) (B-208-071)	3	6/19/79	MSIV Leakage Control System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
865E338CA (5) (B-208-071)	4	10/26/79	MSIV Leakage Control System
865E338CA (6) (B-208-071)	3	6/19/79	MSIV Leakage Control System
865E338CA (7) (B-208-071)	5	1/31/80	MSIV Leakage Control System
865E338CA (8) (B-208-071)	3	6/19/79	MSIV Leakage Control System
865E338CA (9) (B-208-071)	4	10/26/79	MSIV Leakage Control System
865E338CA (10) (B-208-071)	4	10/26/79	MSIV Leakage Control System
865E338CA (11) (B-208-071)	5	1/31/80	MSIV Leakage Control System
865E343CA (1)	0	1/18/77	MSIV Leakage Control System
865E343CA (2)	0	1/18/77	MSIV Leakage Control System
865E343CA (3)	0	1/18/77	MSIV Leakage Control System
865E343CA (4)	0	1/18/77	MSIV Leakage Control System
865E352 (1)	0	2/15/78	Reactor Recirculation System
865E352 (2)	0	2/15/78	Reactor Recirculation System
865E352 (3)	0	2/15/78	Reactor Recirculation System
865E352 (4)	0	2/15/78	Reactor Recirculation System
865E352 (5)	0	2/15/78	Reactor Recirculation System
865E995 (1)	0	7/11/78	Nuclear Boiler System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
865E995	(2)	0	7/11/78	Nuclear Boiler System
865E995	(3)	0	7/11/78	Nuclear Boiler System
865E995	(4)	0	7/11/78	Nuclear Boiler System
865E995	(5)	0	7/11/78	Nuclear Boiler System
865E995	(6)	0	7/11/78	Nuclear Boiler System
865E995	(7)	0	7/11/78	Nuclear Boiler System
866E304	(1)	1	6/26/78	Reactor Recirculation System
866E304	(2)	1	6/26/78	Reactor Recirculation System
866E304	(3)	1	6/26/78	Reactor Recirculation System
866E304	(4)	0	11/10/77	Reactor Recirculation System
866E304	(5)	1	6/26/78	Reactor Recirculation System
866E304	(6)	1	6/26/78	Reactor Recirculation System
866E304	(7)	0	11/10/77	Reactor Recirculation System
D-201-131		R	8/05/82	Pull, Terminal and Junction Boxes-Division 1, 2, 3, and 4
D-201-138		G	8/05/82	Sections, Details, Notes, and References
D-206-010 <Figure 8.3-1>		Z	7/25/00	Main One Line Diagram 13.8 kV and 4.16 kV
D-206-012		-	11/16/76	Legend and Abbreviations
D-206-013		D	9/23/80	Generator

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-206-017 <Figure 8.3-10>	EE	1/31/00	Class 1E 4.16 kV Bus EH11 and EH12
D-206-018 <Figure 8.3-7>	Y	6/22/95	Class 1E 4.16 kV Bus EH13
D-206-019 <Figure 8.3-11>	H	7/16/87	Recirculation Pump Motor Feeders
D-206-020 <Figure 8.3-2>	CC	8/09/00	Main One Line Diagram 480V
D-206-021	C	10/03/80	Class 1E 480V Bus EF1A
D-206-023	C	10/03/80	Class 1E 480V Bus EF1B
D-206-025	C	10/03/80	Class 1E 480V Bus EF1C
D-206-027	C	10/03/80	Class 1E 480V Bus EF1D
D-206-029	C	10/03/80	Class 1E 480V Bus EHF1E
D-206-050 <Figure 8.3-22>	X	5/29/92	Class 1E DC System Div. 3
D-206-051 <Figure 8.3-21>	ZZ	9/24/97	Class 1E DC System Div. 1 and 2
D-206-052	YY	3/16/81	Non-Class 1E DC System Bus D1A & D1B
D-206-053 <Figure 8.3-12>	HH	8/09/00	Class 1E Div. 1 AC System
D-206-054 <Figure 8.3-13>	KK	5/18/99	Class 1E Div. 2 and 3 AC System
D-207-021	E	9/12/80	Unit Overall Differential
D-207-022	E	9/12/80	Generator Differential
D-207-023	E	9/23/80	Main and Unit Auxiliary Transformer Differential
D-207-024	E	9/12/80	Potential

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-207-025	E	9/12/80	Metering
D-207-026	E	9/12/80	Startup Transformer Differential Relaying
D-207-027	C	9/23/80	Inter Bus Transformer LH-1-A Differential
D-207-028	C	9/23/80	Inter Bus Transformer LH-1-B Differential
D-207-029	C	9/23/80	Inter Bus Transformer LH-1-C Differential
D-207-030	B	9/12/80	Startup Transformer Metering and Relaying
D-207-031	C	9/27/78	Synchronizing and Phasing-Non-Class 1E
D-207-032	D	10/03/80	Synchronizing and Phasing-Class 1E
D-207-033	C	10/03/80	Standby Diesel Generator Div. 1 Differential Relaying
D-207-034	C	10/03/80	Standby Diesel Generator Div. 1 Metering
D-207-035	C	10/03/80	Standby Diesel Generator Div. 1 Potential
D-207-036	C	10/03/80	Standby Diesel Generator Div. 2 Differential Relaying
D-207-037	C	10/03/80	Standby Diesel Generator Div. 2 Metering
D-207-038	C	10/03/80	Standby Diesel Generator Div. 2 Potential
D-207-039	C	10/03/80	HPCS Diesel Generator Div. 3 Differential Relaying

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
D-207-040		C	10/03/80	HPCS Diesel Generator Div. 3 Metering
D-207-041		D	10/03/80	HPCS Diesel Generator Div. 3 Potential
B-208-001		C	9/19/79	Graphic Standards
B-208-002		-	9/12/74	Relay Standards
B-208-003		-	10/02/79	480V Switchgear Standards
B-208-004		-	10/03/79	13.8 kV/4.16 kV Standards
B-208-005		-	N/I	Relay Standard
B-208-010		-	7/28/77	B21 Nuclear Boiler Process Instrumentation Index
B-208-010	(A01)	E	1/06/82	Control Tabulation & Power Distribution
B-208-010	(A02)	D	1/06/82	Instrumentation, Computer Inputs
B-208-010	(A09)	D	1/06/82	Feedwater Inlet Shutoff MOV F065A
B-208-010	(A10)	D	1/06/82	Feedwater Inlet Shutoff MOV F065B
B-208-011		-	9/13/74	B21 Nuclear Boiler Automatic Depressurization Index
B-208-011	(C01)	C	5/19/82	Notes, References, Switch Tabulations
B-208-011	(C04)	C	4/16/80	Power Distribution & Thermocouple Identification
B-208-011	(C05)	D	5/19/82	Relay Logic
B-208-011	(C06)	C	8/09/82	Relay Logic
B-208-011	(C07)	C	1/19/82	Analog-Relay Logic

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-011	(C08)	B	1/19/82	ADS Valve
B-208-011	(C09)	D	5/19/82	Relay Logic, Analog Circuits
B-208-011	(C10)	D	5/19/82	Relay Logic, Analog Circuits
B-208-011	(C11)	C	5/19/82	ADS Valves
B-208-011	(C12)	D	5/19/82	ADS Valves
B-208-011	(C13)	E	5/19/82	Safety Relief Valves
B-208-011	(C14)	C	5/19/82	Safety Relief Valves
B-208-011	(C15)	C	1/19/82	Transient Test Panel
B-208-011	(C18)	D	1/19/82	Computer Inputs
B-208-011	(C19)	D	1/19/82	Computer Inputs
B-208-011	(C20)	C	4/16/80	Transient Test Inputs
B-208-011	(C21)	C	1/19/82	Transient Test Inputs
B-208-013		-	1/18/80	Nuclear Boiler Steam Supply Shutoff System-Index
B-208-013	(H01)	D	8/02/82	Notes, References, Legend & Tabulations
B-208-013	(H02)	B	2/03/82	Relay Tabulations
B-208-013	(H03)	B	2/03/82	Relay Tabulations
B-208-013	(H04)	E	8/16/82	Switch Development
B-208-013	(H05)	D	8/16/82	Power Distribution
B-208-013	(H06)	D	8/16/82	Logic A & C Panels 1H13-P691 & 1H13-P693
B-208-013	(H07)	D	8/16/82	Logic B & D Panels 1H13-P692 & 1H13-P694

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-013	(H08)	D	8/16/82	Radwaste System Trip Logic & Status Light
B-208-013	(H09)	F	8/16/82	Reactor Water Sample Valves, Reset Circuit for Isolation Valves
B-208-013	(H10)	D	8/16/82	Main Steam Line Isolation Valves Inboard F022A, F022B, F022C, F022D
B-208-013	(H11)	E	8/16/82	Main Steam Line Isolation Valves Outboard F028A, F028B, F028C, F028D
B-208-013	(H12)	D	8/02/82	RHR/RWCU Isolation Signals
B-208-013	(H13)	E	8/16/82	Postaccident Monitoring Recorder Chart Speed Control Systems A & B
B-208-013	(H14)	D	8/16/82	Trip Units
B-208-013	(H15)	E	8/16/82	Nuclear Steam Supply Shutoff System Trip Units
B-208-013	(H16)	D	8/16/82	Nuclear Steam Supply Shutoff System Transient Test Inputs
B-208-013	(H17)	D	2/03/82	Main Steam Line Drain Isolation MOV (Outboard) F019
B-208-013	(H18)	E	9/16/82	Main Steam Line Drain Isolation MOV F016
B-208-013	(H19)	D	2/03/82	Main Steam Line A Drain Line Isolation MOV (Outboard) F067A
B-208-013	(H20)	D	2/03/82	Main Steam Line A Drain Line Isolation MOV (Outboard) F067B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-013	(H21)	F	9/16/82	RHR Suction Cooling Isolation (Inboard) Valve 1E12-F009
B-208-013	(H22)	E	2/03/82	RHR Suction Cooling Isolation MOV (Throttling) 1E12-F008
B-208-013	(H23)	E	2/03/82	RHR Reactor Head Spray Isolation Valve (Throttling) 1E12-F023
B-208-013	(H24)	C	2/03/82	RHR Discharge to Radwaste Isolation Valve (Inboard) 1E12-F049
B-208-013	(H25)	E	2/03/82	RHR to Radwaste Isolation Valve Outboard Throttling 1E12-F040
B-208-013	(H26)	F	9/16/82	RWCU Discharge Isolation MOV 1G33-F001
B-208-013	(H27)	E	8/02/82	RWCU Discharge Isolation MOV 1G33-F004
B-208-013	(H28)	E	9/16/82	RWCU Discharge to Reactor Feedwater Isolation MOV 1G33-F040
B-208-013	(H29)	D	2/03/82	Main Steam Line D Drain Line Isolation MOV F067D
B-208-013	(H30)	D	2/03/82	RWCU Discharge to Reactor Feedwater Isolation MOV 1G33-F039
B-208-013	(H31)	D	2/03/82	RWCU Discharge to Reactor Feedwater Isolation MOV 1G33-F039
B-208-013	(H32)	E	9/16/82	Main Steam Line D Drain Line Isolation MOV F067C
B-208-013	(H33)	D	2/03/82	RWCU System Valve (Outboard) 1G33-F034

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-013	(H34)	E	9/16/82	RWCU System Valve (Inboard) 1G33-F053
B-208-013	(H35)	D	2/03/82	RWCU System Valve (Outboard) 1G33-F054
B-208-013	(H100)	A	11/17/80	Isolation Signal
B-208-013	(H101)	A	11/17/80	Isolation Signal
B-208-013	(H102)	A	11/17/80	Isolation Signal
B-208-013	(H103)	A	11/17/80	Isolation Signal
B-208-013	(H104)	A	11/17/80	Isolation Signal
B-208-013	(H105)	A	11/17/80	Isolation Signal
B-208-013	(H110)	B	11/17/80	Inop & Bypass
B-208-013	(H111)	A	11/17/80	Inop & Bypass
B-208-013	(H112)	B	11/17/80	Inop & Bypass
B-208-013	(H113)	A	11/17/80	Inop & Bypass
B-208-013	(H114)	C	11/17/80	Inop & Bypass
B-208-015		-	5/08/78	B33 Reactor Recirculation Index
B-208-015	(A01)	D	8/02/82	Notes, References, Tabulations, Legend
B-208-015	(A02)	C	8/02/82	Switch Developments
B-208-015	(A03)	A	12/05/79	Switch Developments
B-208-015	(A04)	C	8/02/82	Relay Tabulations
B-208-015	(A05)	B	12/05/79	Power Distribution
B-208-015	(A06)	B	8/02/82	Power Distribution
B-208-015	(A07)	C	4/30/80	Power Distribution

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-015	(A11)	D	6/19/80	Temperature Recorders
B-208-015	(A27)	H	8/02/82	Recirculation Pump C001A Breaker Control 3A
B-208-015	(A28)	H	8/02/82	Recirculation Pump C001B Breaker Control 3B
B-208-015	(A29)	G	8/02/82	Recirculation Pump C001A Breaker 4A
B-208-015	(A30)	G	8/02/82	Recirculation Pump C001B Breaker 4B
B-208-015 <Figure 7.7-5 (2)>	(200)	F	4/19/96	Recirculation Flow Control Illustrations
B-208-015 <Figure 7.7-5 (4)>	(201)	E	12/13/89	Recirculation Flow Control Illustrations
B-208-015 <Figure 7.7-5 (5)>	(202)	A	12/13/89	Recirculation Flow Control Illustrations
B-208-015 <Figure 7.7-5 (6)>	(203)	-	7/02/87	Recirculation Flow Control Illustrations
B-208-015 <Figure 7.7-5 (3)>	(205)	B	12/13/89	Recirculation Flow Control Illustrations
B-208-020		A	5/08/78	C11A Rod Control and Information System Index
B-208-020	(A01)	G	8/02/82	CRD Temperature Recorder Notes, Symbols, Power Distribution and Reference Documents
B-208-020	(A02)	F	8/02/82	Cabling Diagram
B-208-020	(A03)	F	12/17/81	Rod Position Information and Temperature Recorder
B-208-020	(A04)	D	12/17/81	Rod Position Information and Scram Time Test

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-020	(A08)	F	8/02/82	Rod Control & Information System, CRD Hydraulic Stabilizer, Solonoid Valves & Output to Computer
B-208-021		A	2/01/80	C11B CRD Hydraulic Control System Index
B-208-021	(B01)	C	12/05/79	Notes, References, Power Distribution and Computer Input
B-208-021	(B07)	D	11/05/80	CRD Supply to Reactor Isolation MOV F083
B-208-030	(A00)	A	8/02/82	C41 Standby Liquid Control Index System
B-208-030	(A01)	E	8/02/82	Notes, References, Tabulations & Switch Development
B-208-030	(A02)	F	8/02/82	Power Distribution, Status Lights & Instrumentation
B-208-030	(A03)	G	9/16/82	Storage Tank Outlet Valve F001A
B-208-030	(A04)	G	9/16/82	Storage Tank Outlet Valve F001B
B-208-030	(A05)	F	9/16/82	Standby Liquid Control Pump C001A
B-208-030	(A06)	F	9/16/82	Standby Liquid Control Pump C001B
B-208-035		-	8/24/77	C51C Neutron Monitoring - Startup Range Detection Drive Index
B-208-035	(C01)	C	12/29/81	Notes & Relay Tabulations
B-208-035	(C02)	E	12/29/81	SRM Channel A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-035	(C05)	C	4/11/81	Position Switches & Relays SRM Channel B
B-208-035	(C06)	C	4/11/81	Position Switches & Relays SRM Channel C & D
B-208-037		-	8/24/77	C51A Neutron Monitoring - Startup Range Index
B-208-037	(A01)	-	7/06/82	Notes, Legend, Reference, Tables & Switch Tabulations
B-208-037	(A02)	C	7/15/82	120V AC RPS Power Distribution
B-208-037	(A03)	A	7/06/82	Non-Divisional Power Distribution & Recorder Bus Distribution
B-208-037	(A04)	A	7/06/82	DC Power Distribution
B-208-037	(A05)	A	7/06/82	Relay Tabulation
B-208-037	(A06)	A	7/06/82	Relay Tabulation
B-208-037	(A07)	B	7/06/82	SRM Channel A & C Analog
B-208-037	(A08)	A	7/06/82	SRM Channel B & D
B-208-037	(A09)	A	7/06/82	SRM Channel B Alarm Section
B-208-037	(A10)	A	7/06/82	SRM Channel C Alarm Section
B-208-037	(A11)	A	7/06/82	SRM Channel D Alarm Section
B-208-037	(A12)	B	7/06/82	IRM Channel A, E Analog Section
B-208-037	(A13)	B	7/06/82	IRM Channel B, F Analog Section
B-208-037	(A14)	B	7/06/82	IRM Channel C, G Analog Section
B-208-037	(A15)	B	7/06/82	IRM Channel D, H Analog Section

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-037	(A16)	A	7/06/82	IRM Channel A, E Alarm Section
B-208-037	(A17)	A	7/06/82	IRM Channel B, F Alarm Section
B-208-037	(A18)	A	7/06/82	IRM Channel C, G Alarm Section
B-208-037	(A19)	A	7/06/82	IRM Channel D, H Alarm Section
B-208-037	(A20)	A	7/06/82	Auxiliary Relays, Channel A, E
B-208-037	(A21)	A	7/06/82	Auxiliary Relays, Channel B, F & D, H
B-208-037	(A22)	A	7/06/82	Auxiliary Relays, Channel C, G
B-208-037	(A23)	A	7/06/82	Auxiliary Relays, Channel D, H
B-208-037	(A29)	A	7/06/82	Rod Withdrawal Block Outputs to RCIS
B-208-037	(A30)	A	7/06/82	Trip Outputs to Protection System
B-208-038	(B00)	B	8/02/82	C51B Neutron Monitoring - Power Range Index
B-208-038	(B01)	D	8/02/82	Notes, References, Switch Developments
B-208-038	(B02)	D	8/02/82	Reactor Core Plan View, Relay Tabulation
B-208-038	(B03)	E	8/02/82	Relay Tabulation
B-208-038	(B04)	D	8/02/82	120V AC UPS Bus 1 - Power Distribution

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-038	(B05)	C	8/02/82	120V AC UPS Bus 2 - Power Distribution
B-208-038	(B06)	B	1/30/80	Instrument Bus & Recorder Power Distribution
B-208-038	(B07)	E	7/19/82	DC Power Distribution Panel 1H13-669
B-208-038	(B08)	C	7/19/82	DC Power Distribution Panel 1H13-P669
B-208-038	(B09)	E	7/19/82	DC Power Distribution Panel 1H13-P670
B-208-038	(B10)	E	7/19/82	DC Power Distribution Panel 1H13-P670
B-208-038	(B11)	D	7/19/82	DC Power Distribution Panel 1H13-P671
B-208-038	(B12)	C	7/19/82	DC Power Distribution Panel 1H13-P671
B-208-038	(B13)	D	7/19/82	DC Power Distribution Panel 1H13-P672
B-208-038	(B14)	C	7/19/82	DC Power Distribution Panel 1H13-P672
B-208-038	(B15)	D	12/29/81	Flow Channel A
B-208-038	(B16)	D	12/29/81	Flow Channel B
B-208-038	(B17)	D	12/29/81	Flow Channel C
B-208-038	(B18)	D	12/29/81	Flow Channel D
B-208-038	(B21)	B	12/29/81	APRM Trip Reference - Bus A-Channels A, C, E, G
B-208-038	(B22)	B	12/29/81	APRM Trip Reference - Bus B-Channels B, D, F, H
B-208-038	(B23)	C	12/29/81	APRM Channel A – LPRM Section

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-038	(B24)	D	8/02/82	APRM Channel A – Trip Section
B-208-038	(B25)	B	12/29/81	APRM Channel A - Alarm Section
B-208-038	(B26)	D	12/29/81	APRM Channel B - LPRM Section
B-208-038	(B27)	D	8/02/82	APRM Channel B - Trip Section
B-208-038	(B28)	B	12/29/81	APRM Channel B - Alarm Section
B-208-038	(B29)	E	12/29/81	APRM Channel C - LPRM Section
B-208-038	(B30)	D	8/02/82	APRM Channel C - Trip Section
B-208-038	(B31)	B	12/29/81	APRM Channel C - Alarm Section
B-208-038	(B32)	D	12/29/81	APRM Channel D - LPRM Section
B-208-038	(B33)	D	8/02/82	APRM Channel D - Trip Section
B-208-038	(B34)	B	12/29/81	APRM Channel D - Alarm Section
B-208-038	(B35)	C	2/29/81	APRM Channel E - LPRM Section
B-208-038	(B36)	D	8/02/82	APRM Channel E - Trip Section
B-208-038	(B37)	B	12/29/81	APRM Channel E - Alarm Section
B-208-038	(B38)	D	12/29/81	APRM Channel F - LPRM Section

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-038	(B39)	D	8/02/82	APRM Channel F - Trip Section
B-208-038	(B40)	B	12/29/81	APRM Channel F - Alarm Section
B-208-038	(B41)	D	12/29/81	APRM Channel G - LPRM Section
B-208-038	(B42)	D	8/02/82	APRM Channel G - Trip Section
B-208-038	(B43)	B	12/29/81	APRM Channel G - Alarm
B-208-038	(B44)	D	12/29/81	APRM Channel H - LPRM Section
B-208-038	(B45)	D	8/02/82	APRM Channel H - Trip Section
B-208-038	(B46)	B	12/29/81	APRM Channel H - Alarm Section
B-208-038	(B47)	F	8/02/82	APRM Auxiliary Relays
B-208-038	(B48)	C	12/29/81	APRM Remote Indicators
B-208-038	(B49)	A	1/30/80	APRM/LPRM Calibrator
B-208-038	(B51)	D	12/29/81	Computer Analog Inputs & Transient Test Outputs & APRM Recorders
B-208-038	(B53)	C	12/29/81	Computer Digital Inputs
B-208-038	(B54)	B	12/29/81	RPS Outputs
B-208-038	(B55)	B	12/29/81	RCIS Outputs
B-208-038	(B56)	B	4/11/81	LPRM Display Multiplexer
B-208-038	(B57)	B	12/29/81	LPRM Display Outputs

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-038	(B58)	D	7/19/82	Cabling Diagram, Wire, Cable & Field Run Cable Tabulations
B-208-039		-	3/16/78	C61 Remote Reactor Shutdown Index
B-208-039	(A01)	F	2/03/82	Notes, References & Legend Power Distribution
B-208-039	(A02)	D	12/30/81	Switch Development
B-208-039	(A03)	E	12/30/81	Flow Control and Instrumentation
B-208-040		-	6/30/77	C71A Reactor Protection Index
B-208-040	(A01)	D	8/04/82	Notes, References, Valve Tabulations & Rod Scram Group Arrangements
B-208-040	(A02)	C	8/04/82	Relay Tabulations
B-208-040	(A03)	G	8/04/82	Switch Development & Relay Tabulations
B-208-040	(A04)	D	8/04/82	Power Distribution
B-208-040	(A05)	G	8/04/82	Channel “A” Sensor Relays
B-208-040	(A06)	G	8/04/82	Channel “B” Sensor Relays
B-208-040	(A07)	G	8/04/82	Channel “C” Sensor Relays
B-208-040	(A08)	G	8/04/82	Channel “D” Sensor Relays
B-208-040	(A09)	C	8/04/82	Channel A, B, C & D Scram Trip Logic
B-208-040	(A10)	E	8/04/82	Scram Solenoids
B-208-040	(A11)	E	8/04/82	Scram Discharge Volume Isolation and Backup Valves Recirculation Pumps A & B Trip Logic

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-040	(A12)	F	8/04/82	Computer Inputs
B-208-040	(A13)	F	8/04/82	Testability Power Supplies
B-208-040	(A14)	B	8/04/82	Testability Card File Tabulations
B-208-040	(A15)	D	8/16/82	Testability
B-208-040	(A16)	D	8/16/82	Transient Test
B-208-049	(410)	E	10/31/80	Process Computer System Safety-Related Analog Cabinet A Binary Inputs
B-208-049	(411)	E	10/31/80	Process Computer System Safety-Related Analog Cabinet B Binary Inputs
B-208-050		A	10/26/79	D17 Plant Radiation Monitoring Index
B-208-050	(1)	C	1/23/81	Containment Isolation Valves F079A & F089A
B-208-050	(3)	E	9/28/81	Containment Isolation Valves F079B & F089B Subsystem
B-208-050	(5)	D	2/15/80	Containment Evacuation Alarm
B-208-050	(6)	C	10/26/79	Drywell Evacuation Alarm
B-208-050	(7)	A	10/26/79	Fuel Handling Building Evacuation Alarm
B-208-050	(8)	C	6/09/81	Containment Isolation Valve F071A
B-208-050	(9)	D	6/09/81	Containment Isolation Valve F081A
B-208-050	(10)	C	6/09/81	Containment Isolation Valve F071B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-050	(11)	D	6/09/81	Containment Isolation Valve F081B
B-208-050	(200)	A	10/23/76	Index
B-208-050	(201)	A	10/26/79	General Notes
B-208-050	(268)	C	10/26/79	Auxiliary Remote Alarms & Interlocks
B-208-052		A	7/19/82	D23 Containment Atmosphere Monitoring Index
B-208-052	(1)	B	7/16/81	Containment Isolation Valves F010A, F020A, F030A, F040A
B-208-052	(2)	C	7/16/81	Containment Isolation Valves F010B, F020B, F030B, F040B
B-208-052	(200)	A	1/05/79	C.V. Train "A" Pressure Process Instrumentation
B-208-052	(201)	A	1/05/79	C.V. Train "B" Pressure Process Instrumentation
B-208-052	(202)	A	1/05/79	Containment/Drywell Train "A" Differential Pressure Process Instrumentation
B-208-052	(203)	A	1/05/79	Containment/Drywell Train "B" Differential Pressure Process Instrumentation
B-208-052	(204)	A	1/05/79	Drywell Train "A" Pressure Process Instrumentation
B-208-052	(205)	A	1/05/79	Drywell Train "B" Pressure Process Instrumentation
B-208-052	(206)	E	10/31/80	Suppression Pool Train "A" Temperature Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-052 (207)	F	2/25/81	Suppression Pool Train "B" Temperature Process Instrumentation
B-208-052 (208)	E	10/31/80	Suppression Pool Train "A" Temperature Process Instrumentation
B-208-052 (209)	G	2/25/81	Suppression Pool Train "B" Temperature Process Instrumentation
B-208-052 (210)	D	1/10/80	Drywell Train "A" Temperature Process Instrumentation
B-208-052 (211)	F	2/25/81	Drywell Train "B" Temperature Process Instrumentation
B-208-052 (212)	D	1/10/80	Drywell Train "A" Temperature Process Instrumentation
B-208-052 (213)	E	1/10/80	Drywell Train "B" Temperature Process Instrumentation
B-208-052 (214)	E	2/25/81	Upper Containment Train "A" Temperature Process Instrumentation
B-208-052 (215)	F	2/25/81	Upper Containment Train "B" Temperature Process Instrumentation
B-208-052 (216)	E	2/25/81	Lower Containment Train "A" Temperature Process Instrumentation
B-208-052 (217)	E	2/25/81	Lower Containment Train "B" Temperature Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-052	(218)	A	8/30/77	Train "A" Suppression Pool Temperature & Containment Pressure Recorders
B-208-052	(219)	A	8/30/77	Train "B" Suppression Pool Temperature & Containment Pressure Recorders
B-208-052	(220)	A	8/30/77	Train "A" Containment/ Drywell Temperature Recorder
B-208-052	(221)	A	8/30/77	Train "B" Containment/ Drywell Temperature Recorder
B-208-052	(222)	E	10/31/80	Suppression Pool Train "A" Temperature Process Instrumentation
B-208-052	(223)	E	10/31/80	Suppression Pool Train "B" Temperature Process Instrumentation
B-208-052	(224)	F	10/31/80	Suppression Pool Train "A" Temperature Process Instrumentation
B-208-052	(225)	F	10/31/80	Suppression Pool Train "B" Temperature Process Instrumentation
B-208-053		A	10/23/78	D51 Environs Monitoring Index
B-208-053	(201)	A	2/25/81	Peak Shock Annunciator Inputs Horizontal N/S
B-208-053	(202)	A	2/25/81	Peak Shock Annunciator Inputs Vertical
B-208-053	(203)	B	2/25/81	Peak Shock Annunciator Inputs Horizontal E/W
B-208-053	(204)	A	10/23/78	Seismic Instrument Panel

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-053	(205)	A	10/23/78	Seismic Instrument Panel
B-208-053	(206)	A	10/23/78	Seismic Instrument Panel
B-208-054		-	7/28/77	D17A Process Radiation Monitoring Index
B-208-054	(A01)	E	8/09/82	Switch Developments, Notes, References, Tables, and Legend
B-208-054	(A02)	E	2/03/82	AC Power Distribution
B-208-054	(A03)	C	2/03/82	DC Power Distribution
B-208-054	(A04)	E	2/03/82	Protection System Outputs
B-208-054	(A05)	D	2/03/82	Relay Tabulations
B-208-054	(A06)	C	2/03/82	Main Steam Line Radiation Monitoring Subsystem
B-208-054	(A10)	F	2/03/82	Containment Vent Plenum (Analog Section)
B-208-054	(A11)	E	2/03/82	Containment Vent Plenum (Analog Section)
B-208-055		A	9/21/78	E12 Residual Heat Removal Index
B-208-055	(A01)	D	4/05/82	Valve & Control Tabulation
B-208-055	(A02)	B	4/05/82	Switch Developments
B-208-055	(A03)	E	4/05/82	Relay Tabulation
B-208-055	(A04)	F	8/02/82	Power Distribution for Testability
B-208-055	(A05)	E	4/05/82	Power Distribution
B-208-055	(A06)	C	4/05/82	Power Distribution
B-208-055	(A07)	E	4/05/82	Relay Logic Bus “A”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-055	(A08)	E	4/05/82	Relay Logic Bus “B”
B-208-055	(A09)	D	4/05/82	Relay Logic Bus “B”
B-208-055	(A10)	E	4/05/82	Process Instrumentation
B-208-055	(A11)	D	4/05/82	Process Instrumentation
B-208-055	(A12)	E	8/09/82	MCC Power Loss
B-208-055	(A13)	D	4/05/82	Computer Inputs & Status Lights
B-208-055	(A14)	C	4/05/82	Testability (A)
B-208-055	(A15)	D	4/05/82	Testability (B)
B-208-055	(A16)	D	4/05/82	Testable Check Valves F041A, F041B and F041C
B-208-055	(A17)	D	4/05/82	RHR Pump C002A
B-208-055	(A18)	C	4/05/82	RHR Pump C002B
B-208-055	(A19)	C	4/05/82	RHR Pump C002C
B-208-055	(A20)	C	4/05/82	Water Leg Pump C003
B-208-055	(A21)	D	4/05/82	RHR Pump C002A Suction MOV F004A
B-208-055	(A22)	D	4/05/82	RHR Pump C002B Suction MOV F004B
B-208-055	(A23)	D	4/05/82	RHR Pump C002C Suction MOV F105
B-208-055	(A24)	E	9/16/82	Containment Spray Valve F028A
B-208-055	(A25)	E	9/16/82	Containment Spray Valve F028B
B-208-055	(A26)	E	9/16/82	Containment Spray Valve F537A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-055	(A27)	E	9/16/82	Containment Spray Valve F537B
B-208-055	(A28)	E	4/05/82	Shutdown Cooling Valve F006A
B-208-055	(A29)	E	4/05/82	Shutdown Cooling MOV F006B
B-208-055	(A30)	C	4/05/82	RHR “A” Injection MOV F027A
B-208-055	(A31)	C	4/05/82	RHR “B” Injection Valve F027B
B-208-055	(A32)	F	9/16/82	RHR Injection MOV F042A
B-208-055	(A33)	E	9/16/82	RHR Injection Valve F042B
B-208-055	(A34)	D	4/05/82	RHR Injection Valve F042C
B-208-055	(A35)	D	4/05/82	RHR “A” Test Return MOV F024A
B-208-055	(A36)	D	4/05/82	RHR “B” Test Return MOV F024B
B-208-055	(A37)	D	4/05/82	RHR “C” Test Return MOV F021 (Throttle Valve)
B-208-055	(A38)	E	4/05/82	RHR Pump Minimum Flow MOV F064A
B-208-055	(A39)	D	4/05/82	RHR Pump Minimum Flow Valve F064B
B-208-055	(A40)	D	4/05/82	RHR Pump Minimum Flow MOV F064C
B-208-055	(A41)	B	4/05/82	Steam Pressure Reducing Valve F051A Condensate Discharge to Suppression Pool or RCIC Valve F065A
B-208-055	(A42)	C	4/05/82	Steam Pressure Reducing Valve F051B Condensate Discharge to Suppression Pool or RCIC Valve F065B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-055	(A43)	D	8/23/82	Shutdown Manual Suction Valve F010 RHR A, B & C Manual Injection Valves F039A, B & C
B-208-055	(A44)	E	9/16/82	Shutdown Cooling Upper Pool MOV F037A (Throttle Type)
B-208-055	(A45)	E	9/16/82	Shutdown Cooling Upper Pool MOV F037B (Throttle Type)
B-208-055	(A46)	E	4/05/82	Shutdown Cooling Injection MOV F053A (Throttle Type)
B-208-055	(A47)	E	8/23/82	Shutdown Cooling Injection MOV F053B (Throttle Valve)
B-208-055	(A48)	D	4/05/82	Heat Exchanger-Shell Side Inlet MOV F047A
B-208-055	(A49)	D	4/05/82	Heat Exchanger-Shell Side Inlet MOV F047B
B-208-055	(A50)	E	4/05/82	Heat Exchanger Shell Side Outlet MOV F003A (Throttle Valve)
B-208-055	(A51)	D	4/05/82	Heat Exchanger Shell Side Outlet MOV F003B (Throttle Valve)
B-208-055	(A52)	D	4/05/82	RHR Heat Exchanger-Flow to RCIC MOV F026A
B-208-055	(A53)	D	4/05/82	RHR Heat Exchanger-Flow to RCIC MOV F026B
B-208-055	(A54)	C	4/05/82	RHR Heat Exchanger Flow to Suppression Pool MOV F011A (Throttle Type)
B-208-055	(A55)	C	4/05/82	RHR Heat Exchanger Flow to Suppression Pool MOV F011B (Throttle Type)

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-055 (A56)	E	4/05/82	Heat Exchanger-Shell Side Bypass MOV F048A (Throttle Valve)
B-208-055 (A57)	D	4/05/82	Heat Exchanger-Shell Side Bypass MOV F048B (Throttle Valve)
B-208-055 (A58)	D	4/05/82	Heat Exchanger Vent MOV F073A (Throttle Valve)
B-208-055 (A59)	D	4/05/82	Heat Exchanger Vent MOV F073B (Throttle Valve)
B-208-055 (A60)	D	4/05/82	Heat Exchanger Vent MOV F074A (Throttle Valve)
B-208-055 (A61)	D	4/05/82	Heat Exchanger Vent MOV F074B (Throttle Valve)
B-208-055 (A62)	D	4/05/82	Steam Line Isolation MOV F087A (Throttle Valve)
B-208-055 (A63)	D	4/05/82	Steam Line Isolation Valve F087B (Throttle Valve)
B-208-055 (A64)	D	4/05/82	Steam Line Isolation MOV F052A (Throttle Valve)
B-208-055 (A65)	D	4/05/82	Steam Line Isolation Valve F052B (Throttle Valve)
B-208-055 (A100)	C	11/17/80	LOCA Signal
B-208-055 (A101)	D	7/16/81	Combined LOCA Signal
B-208-055 (A102)	D	7/16/81	LOCA Signal
B-208-055 (A103)	D	7/16/81	Combined LOCA Signal
B-208-055 (A105)	A	7/31/79	Inop & Bypass
B-208-055 (A106)	C	7/23/81	Inop & Bypass

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-060	-	6/30/77	E21 Low Pressure Core Spray Index
B-208-060 (A01)	G	8/23/82	Notes & Control Tabulation
B-208-060 (A02)	D	11/07/80	Switch Development & Power Distribution
B-208-060 (A03)	D	5/27/82	Power Distribution for Testability
B-208-060 (A04)	E	8/09/82	Relay Logic & Testable Check Valve F006
B-208-060 (A05)	E	4/19/82	Status Lights & Computer Inputs
B-208-060 (A06)	D	5/27/82	Testability Circuits
B-208-060 (A07)	G	8/23/82	Power Distribution for Testability
B-208-060 (A08)	E	5/27/82	LPCS Pump C001
B-208-060 (A09)	D	12/05/79	Water Leg Pump C002 & Valve F007
B-208-060 (A10)	D	12/05/79	Minimum Flow to Suppression Pool MOV F011
B-208-060 (A11)	C	12/05/79	Core Spray Injection MOV F005
B-208-060 (A12)	C	12/05/79	Core Spray Pump Suction MOV F001
B-208-060 (A13)	E	5/27/82	Test Bypass to Suppression Pool MOV F012 (Throttle Type)
B-208-065 (A00)	A	3/16/82	E22 High Pressure Core Spray System Index
B-208-065 (A01)	D	3/16/82	Notes, References & Tabulations

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-065	(A02)	D	8/02/82	Switch Developments, Relay Tabulations & Power Distribution
B-208-065	(A03)	F	8/02/82	Relay Logic
B-208-065	(A04)	D	8/02/82	Testability
B-208-065	(A05)	D	3/16/82	Process Instrumentation, Computer Inputs & SOV 1E22-F005
B-208-065	(A06)	D	3/16/82	Status Lights
B-208-065	(A07)	E	3/16/82	Testability Card File Tabulations & Power Supplies
B-208-065	(A08)	D	3/16/82	Test Return to Condensate Storage Tank MOV F011 (Throttling)
B-208-065	(A09)	D	3/16/82	Minimum Flow Bypass to Suppression Pool MOV F012
B-208-065	(A10)	D	3/16/82	Pump Suction from Suppression Pool MOV F015
B-208-065	(A11)	D	3/16/82	Test Bypass to Suppression Pool MOV F023 (Throttling)
B-208-065	(A12)	D	3/16/82	Pump Suction from Condensate Storage Tank MOV F001
B-208-065	(A13)	E	3/16/82	Standby Water Leg Pump C003 & Valve F036
B-208-065	(A14)	D	3/16/82	Pump Injection Shutoff MOV F004
B-208-065	(A15)	D	3/16/82	Test Bypass to Condensate Storage Tank MOV F010 (Throttling)
B-208-065	(A100)	E	4/06/82	LOCA Signal

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-065	(A101)	C	4/06/82	Inop & Bypass
B-208-066		A	7/15/77	E22B High Pressure Core Spray Power Index
B-208-066	(B001)	D	3/16/82	High Pressure Core Spray Power Supply System Pump C001
B-208-066	(B002)	C	3/16/82	Pump C001 Manual Override Control
B-208-066	(B100)	D	4/06/82	HPCS Diesel Driven Generator Division 3 Control 1E22-S001
B-208-066	(B101)	F	4/06/82	HPCS Diesel Generator Control 1E22-S001
B-208-066	(B102)	F	4/14/82	HPCS Diesel Generator Control 1E22-S001 1E22-C005, 1E22-C006
B-208-066	(B103)	B	11/17/80	HPCS Diesel Generator Excitation 1E22-S001
B-208-066	(B104)	C	8/26/81	HPCS Diesel Generator AC Metering & Sync. 1E22-S001
B-208-066	(B105)	C	4/06/82	Starting Air Compressor 1E22-C004B
B-208-066	(B106)	C	4/06/82	Jacket Water Keep Warm Heater 1E22-D010
B-208-066	(B107)	C	4/06/82	Circulating Oil Pump 1E22-C007
B-208-066	(B108)	C	4/06/82	Space Heater 1E22-D011
B-208-066	(B109)	A	7/16/81	High Pressure Core Spray System Diesel Generator Standby Air Compressor 1E22-C004A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-066 (B114)	G	4/06/82	HPCS Diesel Generator Annunciator 1E22-S001
B-208-070	B	3/19/82	E31 Leak Detection System Index
B-208-070 (A01)	B	3/19/82	Notes, References, Switch Developments, Tabulations
B-208-070 (A02)	C	3/19/82	Power Distribution
B-208-070 (A03)	D	3/19/82	Power Distribution
B-208-070 (A05)	C	8/09/82	Valve Logic
B-208-070 (A06)	C	8/09/82	Valve Logic
B-208-070 (A07)	D	3/19/82	RWCU Flow Circuit Computer Input
B-208-070 (A08)	B	3/19/82	Leakage Flow Monitors
B-208-070 (A10)	G	3/19/82	Temperature Elements
B-208-070 (A11)	E	3/19/82	Temperature Elements
B-208-070 (A12)	F	3/19/82	Temperature Elements
B-208-070 (A13)	E	3/19/82	Temperature Recorders
B-208-070 (A14)	B	3/19/82	Computer Inputs
B-208-070 (A15)	B	3/19/82	Sensitive Sump Monitor
B-208-070 (A200)	C	10/31/80	Turbine Power Complex Temperature Process Instrumentation
B-208-070 (A201)	C	10/31/80	Turbine Building Temperature Process Instrumentation
B-208-070 (A202)	C	7/13/79	Power Distribution and Logic

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-071		-	7/28/77	E32 MSIV Leakage Control Index
B-208-071	(A01)	D	3/19/82	Notes, References, Control Tabs
B-208-071	(A02)	D	3/19/82	Relay Tabulations
B-208-071	(A05)	D	3/19/82	Instrument Loop
B-208-071	(A06)	E	3/19/82	Power Distribution
B-208-071	(A07)	D	3/30/81	Logic (Outboard)
B-208-071	(A08)	C	3/19/82	Logic (Inboard)
B-208-071	(A09)	C	3/19/82	Logic (Inboard)
B-208-071	(A10)	D	3/19/82	Outboard Bleed Valve F006
B-208-071	(A11)	D	3/19/82	Outboard Bleed Valve F007
B-208-071	(A12)	C	12/05/79	Outboard Depress Valve F008
B-208-071	(A13)	C	12/05/79	Outboard Depress Valve F009
B-208-071	(A14)	B	8/15/79	Outboard Air Blower C002B
B-208-071	(A15)	B	8/15/79	Outboard Air Blower C002F
B-208-071	(A16)	C	12/05/79	Inboard Valve F001A
B-208-071	(A17)	C	12/05/79	Inboard Valve F002A
B-208-071	(A18)	C	12/05/79	Inboard Valve F001E
B-208-071	(A19)	C	12/05/79	Inboard Valve F002E
B-208-071	(A20)	C	12/05/79	Inboard Valve F001J
B-208-071	(A21)	C	12/05/79	Inboard Valve F002J
B-208-071	(A22)	C	12/05/79	Inboard Valve F001N
B-208-071	(A23)	C	12/05/79	Inboard Valve F002N

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-071	(A24)	C	12/05/79	Flow Transmitter Bypass Valve F003A
B-208-071	(A25)	C	12/05/79	Flow Transmitter Bypass Valve F003E
B-208-071	(A26)	C	12/05/79	Flow Transmitter Bypass Valve F003J
B-208-071	(A27)	D	3/30/81	Flow Transmitter Bypass Valve F003N
B-208-071	(A28)	C	1/15/82	Pipe Heater B001A
B-208-071	(A29)	C	1/15/82	Pipe Heater B001E
B-208-071	(A30)	C	1/15/82	Pipe Heater B001J
B-208-071	(A31)	C	1/15/82	Pipe Heater B001N
B-208-071	(A32)	B	8/15/79	Inboard Air Blower C001
B-208-075		-	7/28/77	E51 Reactor Core Isolation Cooling Index
B-208-075	(A01)	D	8/02/82	Notes, References & Relay Tabulations
B-208-075	(A02)	C	8/02/82	Relay and Switch Tabulations
B-208-075	(A03)	D	6/08/82	Power Distribution
B-208-075	(A04)	D	8/02/82	Logic Circuit A
B-208-075	(A05)	E	8/02/82	Logic Circuits A and B
B-208-075	(A06)	C	6/08/82	Process Instrumentation Equipment
B-208-075	(A07)	F	5/14/82	SOV’s F026, F005, F054, F025, F004
B-208-075	(A08)	F	5/14/82	Status Lights and SOV F066
B-208-075	(A09)	D	8/02/82	Testability Circuit

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-075	(A10)	C	5/14/82	Testability Circuits
B-208-075	(A11)	D	8/02/82	Testability Circuits
B-208-075	(A12)	E	5/14/82	Transient Test Points
B-208-075	(A13)	D	5/14/82	Steam Supply Isolation Valve F064 (Outboard) Throttling, Open Only
B-208-075	(A14)	D	5/14/82	RCIC Injection Shutoff MOV F013
B-208-075	(A15)	C	5/14/82	Pump Suction from Condensate Storage Tank MOV F010
B-208-075	(A16)	C	5/14/82	Minimum Flow to Suppression Pool MOV F019
B-208-075	(A17)	E	10/01/80	RCIC Turbine Steam Supply MOV F045
B-208-075	(A18)	C	5/14/82	Turbine Lube Oil Cooling Water Supply MOV F046
B-208-075	(A19)	E	5/14/82	Test Bypass to Condensate Storage Tank F022 (Throttling) MOV
B-208-075	(A20)	C	9/16/82	Bypass MOV F076 Throttling, Open Only
B-208-075	(A21)	G	8/23/82	Pump Suction from Suppression Pool MOV F031
B-208-075	(A22)	D	5/14/82	Turbine Exhaust to Suppression Pool MOV F068
B-208-075	(A23)	C	5/14/82	Test Bypass to Condensate Storage Tank F059
B-208-075	(A24)	C	12/05/79	Steam Supply Line Isolation (Inboard) to RHR Condensate Heat Exchanger F063

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-075	(A25)	D	5/14/82	Turbine Trip & Throttling Valve F510
B-208-075	(A26)	C	11/26/79	Water Leg Pump C003
B-208-075	(A27)	D	5/14/82	Vacuum Breaker Isolation MOV F077 (Outboard)
B-208-075	(A28)	D	5/14/82	Vacuum Breaker Isolation MOV F078 (Inboard)
B-208-075	(A29)	E	5/14/82	Gland Seal Air Compressor Pump C004
B-208-089		-	6/30/77	G43 Suppression Pool Makeup Index
B-208-089	(1)	D	6/10/81	Suppression Pool Makeup Valve F030A
B-208-089	(2)	C	6/10/81	Suppression Pool Makeup Valve F030B
B-208-089	(3)	D	6/10/81	Suppression Pool Makeup Valve F040A
B-208-089	(4)	C	6/10/81	Suppression Pool Makeup Valve F040B
B-208-089	(5)	C	6/11/81	Misc. Valves & Indication
B-208-089	(200)	E	10/31/80	Train “A” Temperature Process Instrumentation
B-208-089	(201)	E	10/31/80	Train “B” Temperature Process Instrumentation
B-208-089	(202)	D	1/02/80	Upper Pool Train “A” Level Process Instrumentation
B-208-089	(203)	D	1/02/80	Upper Pool Train “B” Level Process Instrumentation
B-208-089	(204)	B	1/02/80	Upper Pool Train “A” Level Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-089	(205)	-	6/27/77	Upper Pool Train "A" Level Process Instrumentation
B-208-089	(206)	B	1/02/80	Upper Pool Train "A" Level Process Instrumentation
B-208-089	(207)	-	6/27/77	Upper Pool Train "A" Level Process Instrumentation
B-208-089	(208)	C	10/31/80	Upper Pool Train "B" Level Process Instrumentation
B-208-089	(209)	-	6/27/77	Upper Pool Train "B" Level Process Instrumentation
B-208-089	(210)	B	1/02/80	Upper Pool Train "B" Level Process Instrumentation
B-208-089	(211)	-	6/27/77	Upper Pool Train "B" Level Process Instrumentation
B-208-094		B	11/17/80	G42 Suppression Pool Cleanup Index
B-208-094	(2)	C	11/17/80	Pump Suction Valve F010
B-208-094	(3)	C	11/17/80	Pump Suction Valve F020
B-208-094	(6)	B	11/17/80	Demineralizer Effluent to RHR Valve F080
B-208-095		B	7/06/79	G41 Fuel Pool Cooling & Cleanup Index
B-208-095	(1)	B	11/17/80	Fuel Pool Circulating Pump G41-C003A
B-208-095	(2)	B	11/17/80	Fuel Pool Circulating Pump G41-C003B
B-208-095	(7)	A	7/06/79	Filter Demineralizer Bypass Isolation Valve G41-F280
B-208-095	(8)	A	7/06/79	Filter Demineralizer Bypass Isolation Valve G41-F285

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-095	(9)	A	7/06/79	Filter Demineralizer Bypass Isolation Valve G41-F290
B-208-095	(10)	A	7/06/79	Filter Demineralizer Bypass Isolation Valve G41-F295
B-208-095	(11)	A	7/06/79	Filter Demineralizer Bypass Valve G41-F360
B-208-095	(12)	A	7/06/79	Filter Demineralizer to Fuel Pool Control Valve G41-F085
B-208-095	(14)	B	7/16/81	Filter Demineralizer to C.V. Pool Isolation Valve 1G41-F100
B-208-095	(15)	C	7/16/81	C.V. Pool to Surge Tank Inboard Isolation Valve 1G41-F140
B-208-095	(16)	B	7/16/81	C.V. Pool to Surge Tank Outboard Isolation Valve 1G41-F145
B-208-095	(17)	B	7/16/81	Filter Demineralizer to C.V. Pool Control Valve 1G41-F090
B-208-096		A	5/10/79	G50 Liquid Radwaste Index
B-208-096	(A)	B	8/07/78	G50 Liquid Radwaste System Index (Continued)
B-208-096	(B)	B	5/10/79	G50 Liquid Radwaste System Index (Continued)
B-208-096	(C)	B	5/10/79	G50 Liquid Radwaste System Index (Continued)
B-208-096	(1)	A	7/13/78	RWCU Backwash Outboard Isolation MCV F272
B-208-096	(2)	B	7/19/82	RWCU Backwash Outboard Isolation MCV F277

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-099		C	9/20/79	G61 Liquid Radwaste Sumps Index
B-208-099	(1)	C	11/17/80	Drywell Equipment Drain Line Inboard Isolation MOV 1G61-F030
B-208-099	(2)	C	11/17/80	Drywell Equipment Drain Line Outboard Isolation MOV 1G61-F035
B-208-099	(5)	C	11/17/80	Containment Equipment Drain Line Inboard Isolation MOV 1G61-F075
B-208-099	(6)	C	11/17/80	Containment Equipment Drain Line Outboard Isolation MOV 1G61-F080
B-208-099	(7)	C	11/17/80	Drywell Floor Drain Line Inboard Isolation MOV 1G61-F150
B-208-099	(8)	C	11/17/80	Drywell Floor Drain Line Outboard Isolation MOV 1G61-F155
B-208-099	(9)	C	11/17/80	Containment Floor Drain Line Inboard Isolation MOV 1G61-F165
B-208-099	(10)	C	11/17/80	Containment Floor Drain Line Outboard Isolation MOV 1G61-F170
B-208-108		A	5/05/80	M14 Containment Vessel & Drywell Purge Index
B-208-108	(8)	B	5/05/80	Containment Vessel Supply Outboard Isolation Valve F040 & Inboard Isolation Valve F045
B-208-108	(9)	B	5/05/80	Drywell Supply Outboard Isolation Valves 1M14-F055A & 1M14-F060A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-108	(10)	B	5/05/80	Drywell Supply Outboard Isolation Valves 1M14-F055B & 1M14-F060B
B-208-108	(11)	C	12/01/80	Drywell Outboard Isolation Valve F065 & Containment Vessel Exhaust Inboard Isolation Valve F085
B-208-108	(12)	C	5/05/80	Drywell Outboard Isolation Valve F070 & Containment Vessel Exhaust Outboard Isolation Valve F090
B-208-108	(13)	C	6/10/81	Containment Vessel Supply Inboard Isolation Valve F190 & Containment Vessel Exhaust Inboard Isolation Valve F200
B-208-109		-	8/25/77	M15 Annulus Exhaust Gas Treatment Index
B-208-109	(1)	C	6/09/81	Annulus Exhaust Fan C001A
B-208-109	(2)	D	6/09/81	Annulus Exhaust Fan C001B
B-208-109	(3)	C	6/10/81	Heating Coil “A” 1M15-D001A
B-208-109	(4)	C	6/10/81	Heating Coil “B” 1M15-D001B
B-208-109	(5)	B	7/22/81	Inop and Bypass
B-208-109	(6)	B	7/22/81	Inop and Bypass
B-208-109	(7)	B	7/22/81	Heating Coil “A” (Continued) 1M15-D001A
B-208-109	(8)	B	7/22/81	Heating Coil “B” (Continued) 1M15-D001B
B-208-109	(201)	C	2/25/81	Train “A” Pressure Control Process Instrumentation
B-208-109	(202)	C	2/25/81	Train “B” Pressure Control Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-109	(203)	B	10/31/80	Trains “A” & “B” Pressure Process Instrumentation
B-208-109	(204)	-	8/25/77	Train “A” Pressure Process Instrumentation
B-208-109	(205)	-	8/25/77	Train “B” Pressure Process Instrumentation
B-208-109	(206)	A	1/05/79	Train “A” Temperature Process Instrumentation
B-208-109	(207)	A	1/05/79	Train “B” Temperature Process Instrumentation
B-208-110		A	11/09/81	M16 Drywell Vacuum Relief Index
B-208-110	(1)	C	6/10/81	Train “A” Vacuum Relief Isolation MOV F010A
B-208-110	(2)	D	11/10/81	Train “B” Vacuum Relief Isolation MOV F010B
B-208-110	(3)	C	7/22/81	Testable Check Valves F020A & B
B-208-111		-	6/30/77	M17 Containment Vacuum Relief Index
B-208-111	(1)	A	9/12/78	Containment Vacuum Relief Isolation MOV F025
B-208-111	(2)	A	9/12/78	Containment Vacuum Relief Isolation MOV F045
B-208-111	(3)	A	9/12/78	Containment Vacuum Relief Isolation MOV F015
B-208-111	(4)	A	9/12/78	Containment Vacuum Relief Isolation MOV F035
B-208-111	(5)	B	11/10/81	Check Valves - F010, F020, F030, F040

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-111	(6)	B	7/23/81	Instrument Line Sensing Isolation Valves F055 & F065
B-208-111	(7)	B	11/24/80	Inop & Bypass
B-208-111	(8)	B	11/24/80	Inop & Bypass
B-208-111	(200)	A	1/05/79	Train "A" Pressure Process Instrumentation
B-208-111	(201)	A	1/05/79	Train "B" Pressure Process Instrumentation
B-208-115		A	1/31/79	M23 MCC, Switchgear & Miscellaneous Electrical HVAC Index
B-208-115	(1)	B	9/08/81	Supply Fan "A" C001A
B-208-115	(2)	B	9/08/81	Equipment Area HVAC Supply Fan C001B
B-208-115	(3)	C	11/10/81	Equipment Area HVAC Recirculation Fan C001A
B-208-115	(4)	D	11/10/81	Equipment Area HVAC Recirculation Fan C002B
B-208-115	(5)	D	11/10/81	Equipment Area HVAC Miscellaneous Solenoid Dampers "A"
B-208-115	(6)	D	11/10/81	Equipment Area HVAC Miscellaneous Solenoid Dampers "B"
B-208-115	(7)	C	11/10/81	Relay Isolation Logic
B-208-115	(207)	D	2/25/81	Train "A" Chilled Water Temperature Control Process Instrumentation
B-208-115	(208)	D	2/25/81	Train "B" Chilled Water Temperature Control Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-116		-	8/25/77	M24 Battery Room Exhaust Index
B-208-116	(1)	D	6/09/81	Exhaust Fan C001A
B-208-116	(2)	E	11/10/81	Exhaust Fan C001B
B-208-116	(3)	D	11/10/81	Trip Logic, Auto Switchover “A” & Alarms
B-208-116	(4)	D	11/10/81	Trip Logic, Auto Switchover “B” & Alarms
B-208-117		C	11/10/81	M25 Control Room HVAC Index
B-208-117	(1)	E	7/06/82	Supply Fan “A” M25-C001A
B-208-117	(2)	E	7/06/82	Supply Fan “B” M25-C001B
B-208-117	(3)	E	7/06/82	Return Fan “A” M25-C002A
B-208-117	(4)	F	8/04/82	Return Fan “B” M25-C002B
B-208-117	(5)	F	7/06/82	Trip Logic “A”
B-208-117	(6)	G	7/06/82	Trip Logic “B”
B-208-117	(7)	H	8/04/82	Control Room Dampers “A” F220A, F250A, E, F255A, F260A, F010A, F020A, F130A, F110A, M26-F040A
B-208-117	(8)	K	8/04/82	Control Room Dampers “B” F220B, F250B, G, F255B, F260B, F010B, F020B, F130B, F110B, M26-F040B
B-208-117	(201)	D	2/25/81	Train “A” Chilled Water Temperature Control Process Instrumentation
B-208-117	(202)	D	2/25/81	Train “B” Chilled Water Temperature Control Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-118		-	8/25/77	M26 Control Room Emergency Recirculation Index
B-208-118	(1)	D	7/06/82	Emergency Recirculation Fan "A" M26-C001A
B-208-118	(2)	D	7/06/82	Emergency Recirculation Fan "B" M26-C001B
B-208-118	(3)	C	7/06/82	Electric Heating Coil D001A
B-208-118	(4)	B	7/06/82	Electric Heating Coil D001B
B-208-118	(5)	C	11/10/81	Inop & Bypass
B-208-118	(6)	C	11/10/81	Inop & Bypass
B-208-118	(7)	A	12/14/78	Electric Heating Coil (Continued) D001A
B-208-118	(8)	A	12/14/78	Electric Heating Coil (Continued) D001B
B-208-120		-	8/25/77	M28 Emergency Pump Area Cooling Index
B-208-120	(1)	B	10/29/79	Ventilation Fan "A" B001A
B-208-120	(2)	B	10/29/79	Ventilation Fan "B" B001B
B-208-124		-	10/28/77	M32 Emergency Service Water Pumphouse Ventilation Index
B-208-124	(1)	C	6/09/81	Ventilation Unit 1M32-C001A
B-208-124	(2)	C	5/01/81	Ventilation Unit 1M32-C001B
B-208-124	(3)	B	7/23/81	Pumphouse Louvers F070A & F070B
B-208-124	(201)	C	2/25/81	Train "A" Temperature Control Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-124	(202)	C	2/25/81	Train “B” Temperature Control Process Instrumentation
B-208-128		-	11/30/77	M36 Offgas Building Exhaust Index
B-208-128	(1)	C	11/09/81	Exhaust Fan A 1M36-C001A
B-208-128	(2)	B	11/09/81	Exhaust Fan B 1M36-C001B
B-208-131		-	7/28/77	M39 ECCS Pump Room Cooler Index
B-208-131	(1)	C	7/13/81	RCIC Pump Room Cooler 1M39-B004
B-208-131	(2)	C	7/23/81	LPCS Pump Room Cooler 1M39-B006
B-208-131	(3)	C	7/23/81	HPCS Pump Room Cooler 1M39-B003
B-208-131	(4)	C	7/23/81	RHR Pump “A” & Heat Exchanger Cooler 1M39-B001A
B-208-131	(5)	C	7/23/81	RHR Pump “B” & Heat Exchanger Cooler 1M39-B001B
B-208-131	(6)	C	7/23/81	RHR Pump “C” Room Cooler 1M39-B002
B-208-132		A	12/14/78	M40 Fuel Handling Building Ventilation Index
B-208-132	(1)	B	10/29/79	Supply Fan A M40-C001A
B-208-132	(2)	C	5/01/81	Supply Fan B M40-C001B
B-208-132	(3)	B	6/09/81	Exhaust Fan A M40-C002A
B-208-132	(4)	B	6/09/81	Exhaust Fan B M40-C002B
B-208-132	(5)	C	2/25/81	Heating Coil D001A
B-208-132	(6)	C	2/25/81	Heating Coil D001B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-132	(7)	A	12/14/78	Heating Coil (Continued) D001A
B-208-132	(8)	A	12/14/78	Heating Coil (Continued) D001B
B-208-135		A	5/03/79	M43 Diesel Generator Building Ventilation Index
B-208-135	(1)	G	8/11/82	Ventilation Fan C001B
B-208-135	(2)	H	8/11/82	Ventilation Fan C001C
B-208-135	(3)	G	8/11/82	Ventilation Fan C001A
B-208-135	(4)	G	8/11/82	Ventilation Fan C002B
B-208-135	(5)	H	8/11/82	Ventilation Fan C002C
B-208-135	(6)	G	8/11/82	Ventilation Fan C002A
B-208-135	(7)	C	2/06/81	Louvers F070A, B, F080A
B-208-135	(8)	C	2/06/81	Louvers F070C, D, F080B
B-208-135	(9)	C	2/06/81	Louvers F070E, F, F080C
B-208-135	(202)	F	1/12/81	Room “A” Temperature Control Process Instrumentation
B-208-135	(203)	E	1/12/81	Room “A” Temperature Control Process Instrumentation
B-208-135	(204)	E	1/12/81	Room “B” Temperature Control Process Instrumentation
B-208-135	(205)	E	1/12/81	Room “B” Temperature Control Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-135	(206)	E	1/12/81	Room “C” Temperature Control Process Instrumentation
B-208-135	(207)	F	1/12/81	Room “C” Temperature Control Process Instrumentation
B-208-140		B	11/09/81	M51 Combustible Gas Control Index
B-208-140	(1)	C	5/28/81	Hydrogen Mixing Compressor C001A
B-208-140	(2)	C	5/29/81	Hydrogen Mixing Compressor C001B
B-208-140	(3)	D	6/02/81	Hydrogen Mixing Compressor Isolation MOV F010A
B-208-140	(4)	D	6/02/81	Hydrogen Mixing Compressor Isolation MOV F010B
B-208-140	(7)	D	6/02/81	Compressor Cooling Water Isolation Valve F020A
B-208-140	(8)	D	6/02/81	Compressor Cooling Water Isolation Valve F020B
B-208-140	(9)	B	6/10/81	Hydrogen Recombiner P.S. 1M51-S001
B-208-140	(10)	B	6/10/81	Hydrogen Recombiner P.S. 1M51-S002
B-208-140	(13)	C	11/09/81	Backup Hydrogen Purge Inboard Isolation MOV F090
B-208-140	(14)	C	11/09/81	Backup Hydrogen Purge Inboard Isolation MOV F110
B-208-140	(15)	B	11/24/80	Hydrogen Analyzer Isolation Valves

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-140	(16)	C	11/24/80	Hydrogen Analyzer Isolation Valves
B-208-140	(17)	C	11/09/81	Inop & Bypass
B-208-140	(18)	B	4/10/80	Inop & Bypass
B-208-140	(22)	A	11/24/80	Compressor & Auxiliary Oil Pump C001A
B-208-140	(32)	A	11/24/80	Compressor & Auxiliary Oil Pump C001B
B-208-140	(200)	D	10/31/80	Hydrogen Analyzer
B-208-140	(201)	D	10/31/80	Hydrogen Analyzer
B-208-142		-	9/26/77	N11 Main & Reheat Steam Index
B-208-142	(6)	E	12/03/80	Main Steam Stop Valve F020A
B-208-142	(7)	E	12/03/80	Main Steam Stop Valve F020B
B-208-142	(8)	E	12/03/80	Main Steam Stop Valve F020C
B-208-142	(9)	E	12/03/80	Main Steam Stop Valve F020D
B-208-158		A	7/15/80	N41 Generator Index
B-208-158	(7)	B	12/15/80	Synchronizing Main Generator and Div. 1 Diesel
B-208-158	(8)	B	7/09/79	Synchronizing Div. 2 and Div. 3 Diesel
B-208-166		-	6/30/77	P11 Condensate Transfer and Storage Index
B-208-166	(6)	C	7/31/81	Influent Outboard Isolation Valve F060
B-208-166	(7)	C	7/31/81	Effluent Inboard Isolation Valve F090

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-166	(8)	C	7/31/81	Effluent Outboard Isolation Valve F080
B-208-169		-	6/30/77	P22 Demineralized Water Index
B-208-169	(1)	A	7/06/79	Containment Supply Outboard Isolation MOV F010
B-208-169	(2)	C	9/25/80	Drywell Isolation Valve F015
B-208-172		-	8/25/77	P41 Service Water Index
B-208-172	(8)	D	6/09/81	Cooling Tower Makeup Isolation Valve P41-F420
B-208-172	(9)	D	6/09/81	Cooling Tower Makeup Isolation Valve F41-F430
B-208-173		A	11/13/81	P42 Emergency Closed Cooling Index
B-208-173	(1)	C	1/06/81	Pump A C001A
B-208-173	(2)	C	4/08/81	Pump B C001B
B-208-173	(3)	A	5/16/79	Chiller Bypass MOV F42-F150A
B-208-173	(4)	A	5/16/79	Chiller Bypass MOV P42-F150A
B-208-173	(5)	A	5/16/79	Fuel Pool Heat Exchanger Bypass MOV P42-F255A
B-208-173	(6)	B	4/10/80	Fuel Pool Heat Exchanger Bypass MOV F255B
B-208-173	(7)	A	5/16/79	Fuel Pool Heat Exchanger Emergency Supply MOV P42-F260A
B-208-173	(8)	B	4/10/80	Fuel Pool Heat Exchanger Emergency Supply MOV F260B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-173 (9)	A	5/16/79	Fuel Pool Heat Exchanger Emergency Return MOV F265A
B-208-173 (10)	B	4/10/80	Fuel Pool Heat Exchanger Emergency Return MOV F265B
B-208-173 (11)	B	4/10/80	Control Room Chiller Cross Tie Isolation MOV F295A
B-208-173 (12)	A	5/16/79	Control Room Chiller Cross Tie Isolation MOV F295B
B-208-173 (13)	A	5/16/79	Control Room Chillers Emergency Supply MOV F300A
B-208-173 (14)	A	5/16/79	Control Room Chillers Emergency Supply MOV F300B
B-208-173 (15)	A	5/16/79	Control Room Chiller Cross Tie Isolation MOV F325A
B-208-173 (16)	A	5/16/79	Control Room Chiller Cross Tie Isolation MOV F325B
B-208-173 (17)	A	5/16/79	Control Room Chillers Emergency Return MOV F330A
B-208-173 (18)	A	5/16/79	Control Room Chillers Emergency Return MOV F330B
B-208-173 (19)	B	6/15/81	Fuel Pool Heat Exchanger Cross Tie Isolation MOV F380A
B-208-173 (20)	A	5/16/79	Fuel Pool Heat Exchanger Cross Tie Isolation MOV F380B
B-208-173 (21)	B	6/15/81	Fuel Pool Heat Exchanger Cross Tie Isolation MOV F390A
B-208-173 (22)	B	4/10/80	Fuel Pool Heat Exchanger Cross Tie Isolation MOV F390B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-173	(24)	-	6/15/81	Control Complex Chillers Normal Return MOV F290
B-208-173	(25)	-	6/15/81	Control Complex Chillers Normal Return MOV F320
B-208-173	(26)	-	6/15/81	Fuel Pool Heat Exchanger Normal Supply MOV F440
B-208-173	(27)	-	6/15/81	Fuel Pool Heat Exchanger Normal Return MOV F445
B-208-173	(200)	C	1/09/81	Emergency Closed Cooling Pump A Pressure & Flow Process Instrumentation
B-208-173	(201)	B	1/09/81	Emergency Closed Cooling Pump B Pressure & Flow Process Instrumentation
B-208-173	(202)	E	1/10/80	Emergency Closed Cooling Temperature Process Instrumentation
B-208-173	(203)	F	2/25/81	Control Complex Chiller “A” Temperature Process Instrumentation
B-208-173	(204)	G	2/25/81	Control Complex Chiller “B” Temperature Process Instrumentation
B-208-174		-	6/30/77	P43 Nuclear Closed Cooling Index
B-208-174	(7)	B	10/07/80	Nuclear Closed Cooling Containment Supply Outboard Isolation MOV F055
B-208-174	(8)	B	10/07/80	Nuclear Closed Cooling Containment Return Outboard Isolation MOV F140
B-208-174	(9)	C	1/06/81	Nuclear Closed Cooling Containment Return Inboard MOV F215

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-174	(10)	C	1/06/81	Nuclear Closed Cooling Drywell Supply Outboard Isolation MOV F355
B-208-174	(11)	C	1/06/81	Nuclear Closed Cooling Drywell Return Outboard Isolation MOV F410
B-208-174	(12)	C	1/06/81	Nuclear Closed Cooling Drywell Return Inboard Isolation MOV F400
B-208-176		B	4/27/79	P45 Emergency Service Water Index
B-208-176	(1)	D	7/20/82	"A" Emergency Service Water Pump C001A
B-208-176	(2)	C	10/29/79	"B" Emergency Service Water Pump C001B
B-208-176	(3)	C	1/07/81	HPCS Emergency Service Water Pump C002
B-208-176	(4)	C	6/09/81	"A" Emergency Service Water Pump Discharge Valve F130A
B-208-176	(5)	C	6/09/81	"B" Emergency Service Water Pump Discharge Valve F130B
B-208-176	(6)	A	11/22/78	HPCS Emergency Service Water Pump Discharge Valve F140
B-208-176	(7)	B	6/09/81	Inlet Isolation Valve to RHR Heat Exchanger A (1P45-F014A)
B-208-176	(8)	A	11/22/78	Inlet Isolation Valve to RHR Heat Exchanger B (1P45-F014B)
B-208-176	(9)	B	6/09/81	Outlet Isolation Valve from RHR Heat Exchanger A (1P45-F068A)

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-176	(10)	A	11/22/78	Outlet Isolation Valve from RHR Heat Exchanger B (1P45-F068B)
B-208-176	(13)	C	1/07/81	Sluice Gate P45-D004A
B-208-176	(14)	C	1/07/81	Sluice Gate P45-D004B
B-208-176	(200)	C	1/10/80	Loop A Pressure & Temperature Process Instrumentation
B-208-176	(201)	C	1/10/80	Loop B Pressure & Temperature Process Instrumentation
B-208-176	(202)	B	2/15/81	Loop C Pressure & Temperature Process Instrumentation
B-208-176	(203)	C	1/05/79	Loop A Flow Process Instrumentation
B-208-176	(204)	C	1/05/79	Loop B Flow Process Instrumentation
B-208-176	(205)	B	1/05/79	Loop C Flow Process Instrumentation
B-208-178		A	11/09/81	P47 Control Complex Chilled Water Index
B-208-178	(1)	F	9/20/82	Control Complex Chiller “A” B001A
B-208-178	(2)	D	11/05/81	Control Complex Chiller “B” B001B
B-208-178	(4)	C	11/09/81	Chilled Water Pump “A” C001A
B-208-178	(5)	D	11/09/81	Chilled Water Pump “B” C001B
B-208-178	(8)	B	6/09/81	Nonsafety Coil Outlet Isolation Valve “A” F290A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-178	(9)	B	6/09/81	Nonsafety Coil Outlet Isolation Valve “B” F290B
B-208-178	(10)	A	4/27/79	Nonsafety Coil Outlet Isolation Valve “A” F295A
B-208-178	(11)	A	4/27/79	Nonsafety Coil Outlet Isolation Valve “B” F295B
B-208-178	(12)	F	11/05/81	Chiller “A” Controls B001A
B-208-178	(13)	E	11/09/81	Chiller “A” Controls (Continued) B001A
B-208-178	(14)	F	11/05/81	Chiller “B” Controls B001B
B-208-178	(15)	E	11/09/81	Chiller “B” Controls (Continued) B001B
B-208-178	(16)	B	10/17/80	Chiller “A” Oil Pump B001A
B-208-178	(17)	B	10/17/80	Chiller “B” Oil Pump B001B
B-208-180		-	10/27/78	P49 Emergency Service Water Screen Wash Index
B-208-180	(1)	D	11/10/81	Screen Control P49-D001A
B-208-180	(2)	A	11/10/81	Strainer P49-D003A and Pump P49-C002A
B-208-180	(4)	D	11/10/81	Screen Control P49-D001B
B-208-180	(5)	A	11/10/81	Strainer P49-D003B and Pump P49-C002B
B-208-181		-	6/30/77	P50 Containment Vessel Chilled Water Index
B-208-181	(7)	D	1/23/81	Containment Vessel Chilled Water Isolation Valve 1P50-F140
B-208-181	(9)	C	1/23/81	Containment Vessel Chilled Water Isolation Valve 1P50-F150

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-181	(10)	B	10/07/80	Containment Vessel Chilled Water Isolation Valve 1P50-F060
B-208-182		B	4/10/80	P51 Service Air Index
B-208-182	(2)	B	10/07/80	Service Air Receiver Tank Drain Valve 1P51-F080 & Outboard Isolation Valve 1P51-F150
B-208-182	(9)	C	1/23/81	Drywell Isolation Valve F652
B-208-183		A	4/10/80	P52 Instrument Air Index
B-208-183	(3)	C	11/10/81	Outboard Isolation Valve F200
B-208-183	(10)	B	3/17/81	Outboard Isolation Valves 1P52-F160 & F170
B-208-183	(11)	C	10/08/81	B21 Accumulator Isolation Valve F646
B-208-184		A	1/23/81	P53 Penetration Pressurization Index
B-208-184	(1)	C	6/03/81	Outboard Isolation Valves 1P53-F030, F035, F040, F045
B-208-184	(2)	C	6/03/81	Inboard Isolation Valves 1P53-F050, F055, F060, F065
B-208-184	(3)	-	10/02/79	Personnel Air Locks Alarm Units
B-208-184	(4)	B	1/21/80	Personnel & Drywell Air Locks Miscellaneous Relaying
B-208-185		A	3/10/80	P54 Fire Protection Index

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-185	(3)	B	10/29/79	Charcoal Filter Deluge Valves F3180, F3270, F3290, F3000, F3250, M26-F080A, M40-F150A & C, 1M36-F110A, 1M15-F100A
B-208-185	(4)	B	10/29/79	Charcoal Filter Deluge Valves F3200, F3280, F3010, F3260, M26-F080B, M40-F150B, 1M36-F110B, 1M15-F100B
B-208-185	(5)	B	2/25/81	CO_2 Outboard Isolation Valve 1P54-F340
B-208-185	(6)	B	2/25/81	CO_2 Outboard Isolation Valve 1P54-F395
B-208-185	(42)	A	2/25/81	Fire Protection System CO_2 Discharge Valves IP 543410, IP 543430, IP 543420
B-208-194	(3)	E	12/15/80	Condenser Hogging Pump Isolation Valves - F130A & B, Steam Jet Air Ejector Isolation Valves – F140A & B
B-208-198		-	7/28/77	P86 Nitrogen Supply Index
B-208-198	(1)	B	7/6/82	Nitrogen Supply Isolation Valve (1P86-F002)
B-208-199		A	1/23/81	P57 Safety-Related Instrument Air Index
B-208-199	(1)	A	7/06/79	Containment Isolation Valve 1P57-F015A
B-208-199	(2)	B	4/29/80	Containment Isolation Valve 1P57-F015B
B-208-199	(3)	B	1/23/81	Drywell Isolation Valve 1P57-F020A
B-208-199	(4)	C	1/23/81	Drywell Isolation Valve 1P57-F020B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-206		A	7/23/79	R22 Metal Clad Switchgear Index
B-208-206	(1)	E	8/07/81	Unit 2 Tie Breaker L1001
B-208-206	(2)	E	8/07/81	13.8 kV Bus L10 Startup Supply Breaker L1003
B-208-206	(4)	D	3/18/81	13.8 kV Bus L12 Startup Supply Breaker L1009
B-208-206	(5)	E	10/29/81	Interbus Transformer LH-1-A Supply Breaker L1010
B-208-206	(6)	D	8/07/81	13.8 kV Bus L10 Startup Alternate Supply Breaker L1004
B-208-206	(15)	C	3/18/81	Interbus Transformer LH-1-C Supply Breaker L1206
B-208-206	(23)	G	7/20/82	4.16 kV Unit 2 Tie Breaker EH1101
B-208-206	(24)	G	7/20/82	4.16 kV Bus EH11 Diesel Breaker EH1102
B-208-206	(25)	D	7/20/82	Transformer EHF-1-A Supply Breaker EH1104
B-208-206	(26)	D	7/20/82	Transformer EHF-1-B Supply Breaker EH1113
B-208-206	(27)	F	7/20/82	4.16 kV Bus EH11 Preferred Supply Breaker EH1114
B-208-206	(28)	E	7/20/82	4.16 kV Bus EH11 Alternate Preferred Supply Breaker EH1115
B-208-206	(29)	C	7/20/82	Bus EH11 Stub Bus Tie Breaker EH1116
B-208-206	(30)	E	8/07/81	4.16 kV Bus EH12 Diesel Breaker EH1201

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-206	(31)	C	8/07/81	Transformer EHF-1-C Supply Breaker EH1204
B-208-206	(32)	C	8/28/81	Transformer EHF-1-D Supply Breaker EH1209
B-208-206	(33)	E	8/28/81	4.16 kV Bus EH12 Preferred Supply Breaker EH1212
B-208-206	(34)	D	8/28/81	4.16 kV Bus EH12 Alternate Preferred Source Breaker EH1213
B-208-206	(35)	D	8/28/81	4.16 kV Bus EH12 Stub Bus Tie Breaker EH1214
B-208-206	(37)	E	8/28/81	4.16 kV Bus EH13 Diesel Breaker EH1301
B-208-206	(38)	D	3/19/81	4.16 kV Bus EH13 Alternate Preferred Supply Breaker EH1302
B-208-206	(39)	F	8/28/81	4.16 kV Bus EH13 Preferred Supply Breaker EH1303
B-208-206	(40)	C	8/28/81	Transformer EHF-1-E Supply Breaker EH1305
B-208-206	(41)	D	8/28/81	13.8 kV Bus L10 Undervoltage and Potential Circuits
B-208-206	(46)	D	7/20/82	4.16 kV Bus EH11 Undervoltage and Potential Circuits
B-208-206	(47)	F	8/28/81	4.16 kV Bus EH12 Undervoltage and Potential Circuits
B-208-206	(48)	D	7/20/82	4.16 kV Bus XH11 & XH12 Undervoltage and Potential Circuits

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-206	(49)	E	8/28/81	4.16 kV Bus EH13 Undervoltage and Potential Circuits
B-208-206	(50)	E	8/28/81	4.16 kV Bus Differential Lock-Out Relays
B-208-206	(52)	E	10/29/81	Interbus Transformer LH-1-B & LH-1-C Lock-Out Relays
B-208-206	(53)	E	9/29/81	4.16 kV Standby Diesel Breaker EH1102 Protective Relaying
B-208-206	(54)	E	9/29/81	4.16 kV Standby Diesel Breaker EH1201 Protective Relaying
B-208-206	(55)	E	8/28/81	4.16 kV Standby Diesel Breaker EH1301 Protective Relaying
B-208-206	(200)	E	10/11/81	Metal-Clad Switchgear (15 kV & 5 kV) Diesel Generator Circuit Breaker Contacts to Transient Test Panel
B-208-206	(201)	D	4/30/79	Auxiliary and Startup Transformer Contacts to Transient Test Panel
B-208-207		A	7/11/79	R23 Load Centers Index
B-208-207	(8)	B	8/28/81	480V Load Center EF1A Main Supply Breaker & Tie Breaker EF1A03, EF1A13
B-208-207	(9)	B	8/28/81	480V Load Center EF1B Main Supply Breaker EF1B03
B-208-207	(10)	B	8/28/81	480V Load Center EF1C Main Supply Breaker & Tie Breaker EF1C03, EF1C13

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-207	(11)	B	8/28/81	480V Load Center EF1D Main Supply Breaker EF1D03
B-208-207	(12)	B	9/29/81	Potential Ground Circuit Typical
B-208-207	(20)	B	8/28/81	480V Load Center EF1A Spare Breaker EF1A11
B-208-207	(21)	B	8/28/81	480V Load Centers EF1C Spare Breaker EF1C11
B-208-209		B	3/17/81	R25 Distribution Panels Index
B-208-209	(2)	-	11/30/77	Unit Control Console 1H13-P680
B-208-209	(3)	-	11/30/77	Unit Control Console 1H13-P680
B-208-209	(5)	A	2/27/79	HVAC Panel 1H13-P800
B-208-209	(6)	B	5/28/80	HVAC Panel 1H13-P800
B-208-209	(7)	A	2/27/79	HVAC Panel 1H13-P800
B-208-209	(10)	A	2/20/79	Long Response Benchboard 1H13-P870
B-208-209	(11)	A	2/27/79	Long Response Benchboard 1H13-P870
B-208-209	(13)	A	2/27/79	Common HVAC Panel H13-P904
B-208-209	(14)	A	2/27/79	Common HVAC Panel H13-P904
B-208-209	(16)	A	2/27/79	Common Long Response Benchboard H13-P970
B-208-209	(17)	A	2/27/79	Common Long Response Benchboard H13-P970
B-208-209	(20)	A	2/27/79	Auxiliary Relay Panel 1H13-P873

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-209	(23)	C	10/17/80	Auxiliary Relay Panel 1H13-P871
B-208-209	(25)	B	7/10/79	Auxiliary Relay Panel 1H13-P872
B-208-209	(27)	A	2/27/79	Common Auxiliary Relay Panel H13-P969
B-208-209	(28)	A	2/27/79	Common Auxiliary Relay Panel H13-P969
B-208-209	(31)	A	2/27/79	ECCS Benchboard 1H13-P601
B-208-209	(32)	A	2/27/79	ECCS Benchboard 1H13-P601
B-208-209	(40)	C	10/06/81	Space Heater Distribution Panel 1R25-S037
B-208-209	(41)	A	10/06/81	Space Heater Distribution Panel 1R25-S037 (Continued)
B-208-209	(42)	D	3/17/81	Space Heater Distribution Panel 1R25-S041
B-208-209	(44)	B	10/29/79	Space Heater Distribution Panel 1R25-S043
B-208-209	(45)	B	10/30/79	Space Heater Distribution Panel 1R25-S043 (Continued)
B-208-209	(46)	B	10/31/79	Space Heater Distribution Panel 1R25-S047
B-208-209	(48)	A	2/27/79	Space Heater Distribution Panel 1R25-S039
B-208-209	(50)	A	2/27/79	Space Heater Distribution Panel 1R25-S045
B-208-209	(52)	A	2/27/79	Space Heater Distribution Panel 1R25-S099

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-209	(62)	A	2/27/79	Distribution Panels Containment and Drywell solation Valve Control Panel 1H13-P881
B-208-209	(90)	A	7/10/79	Elapsed Time Meters Bus EH11
B-208-209	(91)	A	7/10/79	Elapsed Time Meters Bus EH12
B-208-209	(92)	A	7/10/79	Elapsed Time Meters Bus EH13
B-208-214		A	4/30/79	R41 Instruments Power Distribution Index
B-208-214	(208)	E	6/09/81	Safety-Related Analog Loop Cabinet "A" Power Distribution
B-208-214	(209)	C	3/12/82	Analog Loop Division 1 Instrumentation Panel Power Distribution
B-208-214	(210)	D	6/09/81	Safety-Related Analog Loop Cabinet "B" Power Distribution
B-208-214	(211)	C	3/12/82	Analog Loop Division 2 Instrumentation Panel Power Distribution
B-208-214	(215)	C	2/11/81	Division "3" Analog Loop Cabinet Power Distribution
B-208-214	(220)	B	3/12/82	1H13-P868, 1H13-P869, and 1H13-P873 Signal Resistor Units and Indicator Amplifiers Tabulation
B-208-214	(221)	C	10/31/80	1H13-P869 Signal Resistor Units and Indicator Amplifiers Tabulation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-214	(222)	A	10/31/80	Div. 1 Analog Isolator Power Distribution
B-208-214	(223)	A	10/31/80	Div. 2 Analog Isolator Power Distribution
B-208-214	(227)	B	10/31/79	Common Analog Loop (Div. 1) Instrumentation Panel Power Distribution
B-208-214	(228)	B	10/31/79	Common Analog Loop (Div. 2) Instrumentation Panel Power Distribution
B-208-214	(231)	B	6/09/81	Div. 1 Analog Isolator (1H13-P601) Power Distribution
B-208-214	(232)	B	6/09/81	Div. 2 Analog Isolator (1H13-P601) Power Distribution
B-208-214	(233)	C	6/09/81	Div. 3 Analog Isolator (1H13-P601) Power Distribution
B-208-214	(237)	C	4/05/78	1H13-P601 Division 1 Power Distribution
B-208-214	(238)	C	4/05/78	1H13-P601 Division 2 Power Distribution
B-208-214	(239)	E	1/05/79	1H13-P601 Division 3 Power Distribution
B-208-215		B	7/11/79	R42 DC System Index
B-208-215	(1)	A	2/05/79	DC Miscellaneous DC Volt Meters
B-208-215	(4)	D	7/20/82	Distribution Panel 1R42-S012 Switchgear Feeds
B-208-215	(5)	D	5/18/81	Distribution Panel 1R42-S013 Switchgear Feeds

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-215	(6)	C	3/18/81	Distribution Panel 1E22-P002 Switchgear Feed
B-208-215	(14)	D	8/12/81	Metering and Relaying Bus ED-1-A
B-208-215	(15)	D	8/12/81	Metering and Relaying Bus ED-1-B
B-208-215	(16)	D	7/13/81	Battery Chargers-1R42-S005, S006, S007, S008, S009, S019, & S026
B-208-215	(17)	B	3/18/81	Battery Charger 1R42-S011
B-208-215	(25)	A	2/05/79	Auxiliary Relay Panel 1H13-P872 Power Distribution
B-208-215	(27)	C	10/06/81	Common Analog/Auxiliary Relay Panel H13-P969 Power Distribution
B-208-215	(30)	A	2/05/79	Auxiliary Relay Panel 1H13-P871 Power Distribution
B-208-215	(32)	C	10/06/81	Common Analog/Auxiliary Relay Panel H13-P969 Power Distribution
B-208-216		A	1/24/79	R43 Standby Diesel Generator Index
B-208-216	(1)	C	6/03/81	Emergency Diesel Driven Generator Division 1 Control 1R43-5001A
B-208-216	(2)	C	6/03/81	Emergency Diesel Driven Generator Division 2 Control 1R43-5001B
B-208-216	(3)	A	6/10/81	Legend Division 1 Control 1R43-C001A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-216	(4)	A	6/10/81	Legend Division 2 Control 1R43-C001B
B-208-216	(5)	D	9/28/81	Diesel Control Panel 1H51-P054A Division 1 1R43-C001A
B-208-216	(6)	D	9/28/81	Diesel Control Panel 1H51-P054B Division 2 1R43-C001B
B-208-216	(7)	D	9/28/81	Diesel Control Panel 1H51-P054A Division 1 1R43-C001A
B-208-216	(8)	D	9/28/81	Diesel Control Panel 1H51-P055B Division 2 1R43-C001B
B-208-216	(9)	B	10/06/81	Diesel Control Panel 1H51-P054A Division 1 1R43-C001A
B-208-216	(10)	B	10/06/81	Diesel Control Panel 1H51-P054B Division 2 1R43-C001B
B-208-216	(11)	D	7/17/81	Inop & Bypass
B-208-216	(12)	C	10/06/81	Inop & Bypass
B-208-216	(13)	C	6/10/81	Inop & Bypass
B-208-216	(14)	D	10/06/81	Inop & Bypass
B-208-216	(15)	B	10/06/81	Diesel/Generator Control Interconnection Diagram Division 1
B-208-216	(16)	A	8/09/79	Diesel/Generator Control Interconnection Diagram Division 2
B-208-216	(17)	B	4/10/80	Diesel Control Panel 1H51-P054A Division 1 1R43-C001A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-216 (18)	B	4/10/80	Diesel Control Panel 1H51-P054B Division 2 1R43-C001B
B-208-216 (19)	A	8/09/79	Diesel Annunciator Schematic Division 1 1R43-C001A
B-208-216 (20)	A	8/09/79	Diesel Annunciator Schematic Division 2 1R43-C001B
B-208-216 (21)	A	8/09/79	Diesel Annunciator Schematic Division 1 1R43-C001A
B-208-216 (22)	A	8/09/79	Diesel Annunciator Schematic Division 2 1R43-C001B
B-208-216 (23)	-	1/24/79	Crankcase Fans
B-208-216 (24)	-	1/24/79	Generator Control Switch Developments Division 1 1R43-S001A
B-208-216 (25)	-	1/24/79	Generator Control Switch Developments Division 2 1R43-S001B
B-208-216 (26)	B	5/01/80	Generator Control Switch Developments Division 1 1R43-S001A
B-208-216 (27)	B	5/01/80	Generator Control Switch Developments Division 2 1R43-S001B
B-208-216 (28)	A	4/10/80	Generator Control Interconnection Diagram Division 1 1R43-S001A
B-208-216 (29)	A	4/10/80	Generator Control Interconnection Diagram Division 2 1R43-S001B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-216	(30)	C	10/06/81	Generator Control Panel 1H51-P055A Division 1 1R43-S001A
B-208-216	(31)	C	10/06/81	Generator Control Panel 1H51-P055B Division 2 1R43-S001B
B-208-216	(32)	C	10/06/81	Generator Control Panel 1H51-P055A Division 1 1R43-S001A
B-208-216	(33)	C	10/06/81	Generator Control Panel 1H51-P055B Division 2 1R43-S001B
B-208-216	(34)	A	5/02/80	Generator Control Panel Schematic Division 1 1R43-S002A
B-208-216	(35)	A	5/02/80	Generator Control Panel Schematic Division 2 1R43-S002B
B-208-216	(36)	A	10/06/81	Generator Control Power Chassis Schematic Division 1 1H51-P055A
B-208-216	(37)	A	10/06/81	Generator Control Power Chassis Schematic Division 2 1H51-P055B
B-208-216	(38)	-	1/24/79	Generator Control Panel 1H51-P055A Division 1 1R43-S001A
B-208-216	(39)	-	1/24/79	Generator Control Panel 1H51-P055B Division 2 1R43-S001B
B-208-216	(40)	-	1/24/79	Generator Control Rectifier Chassis Schematic Division 1 1H51-P055A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-216 (41)	-	1/24/79	Generator Control Rectifier Chassis Schematic Division 2 1H51-P055B
B-208-216 (42)	A	4/10/80	Engine & Skid Wiring Division 1 1R43-C001A
B-208-216 (43)	A	4/10/80	Engine & Skid Wiring Division 2 1R43-C001B
B-208-216 (44)	-	1/24/79	Potentiometer Division 1 1R43-S001A
B-208-216 (45)	-	1/24/79	Potentiometer Division 2 1R43-S001B
B-208-217	A	2/05/79	R44 Standby Diesel Generator Starting Air Index
B-208-217 (1)	A	2/05/79	Diesel Generator Starting Air Starting Air Compressor 1R44-C001A
B-208-217 (2)	A	2/05/79	Diesel Generator Starting Air Starting Air Compressor 1R44-C001B
B-208-217 (3)	A	2/05/79	Diesel Generator Starting Air Starting Air Compressor 1R44-C002A
B-208-217 (4)	A	2/05/79	Diesel Generator Starting Air Starting Air Compressor 1R44-C002B
B-208-217 (5)	B	12/17/80	Starting Air Dryer Division 1 1R44-D001A
B-208-217 (6)	B	12/17/80	Starting Air Dryer Division 2 1R44-D001B
B-208-217 (7)	A	12/17/80	Air After Cooler 1R44-B001A
B-208-217 (8)	A	12/17/80	Air After Cooler 1R44-B001B

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-217	(9)	A	12/17/80	Air After Cooler 1R44-B002A
B-208-217	(10)	B	12/17/80	Starting Air Dryer Division 1 1R44-D002A
B-208-217	(11)	A	12/17/80	Air After Cooler 1R44-B002B
B-208-217	(12)	B	12/17/80	Starting Air Dryer Division 2 1R44-D002B
B-208-218		-	11/30/77	R45 Standby Diesel Generator Fuel Oil Index
B-208-218	(1)	D	6/15/81	Fuel Oil Transfer Pump R45-C001C
B-208-218	(2)	D	6/15/81	Fuel Oil Transfer Pump R45-C001D
B-208-218	(3)	D	6/15/81	Backup Fuel Oil Transfer Pump R45-C001A
B-208-218	(4)	D	6/15/81	Backup Fuel Oil Transfer Pump R45-C001B
B-208-218	(5)	C	6/04/81	Fuel Oil Transfer Pump R45-C002A
B-208-218	(6)	D	6/04/81	Backup Fuel Oil Transfer Pump R45-C002B
B-208-218	(200)	G	10/31/80	Diesel Generator Fuel Oil Day Tank “A” & Storage Tank “A” Level Process Instrumentation
B-208-218	(201)	G	10/31/80	Diesel Generator Fuel Oil Day Tank “B” & Storage Tank “B” Level Process Instrumentation
B-208-218	(202)	E	10/31/80	Diesel Generator Fuel Oil High Pressure Core Spray Diesel Day Tank & Storage Tank Level Process Instrumentation

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-220		B	10/31/79	S11 Power Transformers Index
B-208-220	(200)	C	1/02/80	Main Transformers "A", "B" & "C" Temperature Process Instrumentation
B-208-222		-	8/24/77	R61 Control Room Annunciator System Index
B-208-222	(4)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(5)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(6)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(7)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(8)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(9)	A	10/24/77	Unit Control Console (1H13-P680) Section 1A
B-208-222	(54)	L	6/09/81	Unit Control Console (1H13-P680) Section 3A
B-208-222	(55)	B	6/09/78	Unit Control Console (1H13-P680) Section 3A
B-208-222	(100)	B	3/17/78	Unit Control Console (1H13-P680) Section 5A
B-208-222	(101)	B	6/09/81	Unit Control Console (1H13-P680) Section 5A
B-208-222	(102)	B	6/09/78	Unit Control Console (1H13-P680) Section 5A
B-208-222	(103)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(104)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(105)	C	6/09/81	Unit Control Console (1H13-P680) Section 5A
B-208-222	(106)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(107)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(108)	B	6/09/81	Unit Control Console (1H13-P680) Section 5A
B-208-222	(109)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(110)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(111)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(112)	A	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(113)	B	11/09/77	Unit Control Console (1H13-P680) Section 5A
B-208-222	(114)	C	6/09/81	Unit Control Console (1H13-P680) Section 5A
B-208-222	(115)	B	12/31/80	Unit Control Console (1H13-P680) Section 5A
B-208-222	(116)	B	12/31/80	Unit Control Console (1H13-P680) Section 5A
B-208-222	(117)	E	10/01/81	Unit Control Console (1H13-P680) Section 5A
B-208-222	(138)	E	6/09/81	Unit Control Console (1H13-P680) Section 7A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(140)	D	1/07/82	Unit Control Console (1H13-P680) Section 7A
B-208-222	(153)	-	8/24/77	Unit Control Console (1H13-P680) Section 8A
B-208-222	(154)	D	6/09/81	Unit Control Console (1H13-P680) Section 8A
B-208-222	(200)	G	12/31/80	Control Room Annunciator Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(201)	H	10/31/80	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(202)	H	10/31/80	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(203)	H	6/09/81	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(204)	J	10/31/80	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(205)	B	10/15/81	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(206)	C	4/01/80	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(207)	C	4/01/80	Diesel Generator BB (1H13-P877) Section 1A
B-208-222	(208)	A	1/07/82	Control Room Annunciator DC Alarms
B-208-222	(225)	G	6/09/81	Control Room Annunciator Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(226)	G	10/31/80	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(227)	G	10/31/80	Diesel Generator BB (1H13-P877) Section 2A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(228)	H	12/31/80	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(229)	H	10/31/80	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(230)	A	10/15/81	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(231)	D	4/01/80	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(232)	D	4/01/80	Diesel Generator BB (1H13-P877) Section 2A
B-208-222	(233)	A	1/07/82	Control Room Annunciator DC Alarms
B-208-222	(250)	A	11/09/77	Diesel Generator BB (1H13-P877) Section 3A
B-208-222	(251)	B	12/31/80	Diesel Generator BB (1H13-P877) Section 3A
B-208-222	(252)	C	12/31/80	Diesel Generator BB (1H13-P877) Section 3A
B-208-222	(275)	F	10/12/81	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(276)	B	1/02/80	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(277)	A	9/30/77	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(278)	E	10/31/80	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(279)	E	10/15/81	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(280)	F	10/31/80	ECCS Benchboard (1H13-P601) Section 16A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(281)	G	6/09/81	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(282)	F	9/02/81	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(283)	-	9/02/81	Div. 3 Diesel DC System Trouble
B-208-222	(284)	-	8/24/77	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(285)	C	10/31/80	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(300)	H	9/01/81	ECCS Benchboard (1H13-P601) Section 16A
B-208-222	(301)	B	1/05/79	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(302)	A	2/28/78	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(303)	A	2/28/78	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(304)	-	8/24/77	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(305)	D	12/31/80	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(307)	C	3/17/78	ECSS Benchboard (1H13-P601) Section 17A
B-208-222	(308)	B	10/24/77	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(309)	E	6/09/81	ECCS Benchboard (1H13-P601) Section 17A
B-208-222	(310)	F	6/09/81	ECCS Benchboard (1H13-P601) Section 17A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (311)	E	6/09/81	ECCS Benchboard (1H13-P601) Section 17A
B-208-222 (312)	F	6/09/81	ECCS Benchboard (1H13-P601) Section 17A
B-208-222 (313)	A	11/14/77	ECCS Benchboard (1H13-P601) Section 17A
B-208-222 (328)	B	10/02/78	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (331)	A	11/14/77	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (332)	F	12/31/80	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (333)	D	12/31/80	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (334)	C	1/05/79	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (335)	B	3/17/78	ECCS Benchboard (1H13-P601) Section 18A
B-208-222 (350)	E	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (351)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (352)	E	9/01/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (354)	A	9/01/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (355)	F	12/31/80	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (356)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (357)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (358)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (359)	A	11/14/77	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (360)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (361)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (362)	A	10/24/77	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (363)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (364)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (365)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (366)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (367)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (368)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (369)	C	1/02/80	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (370)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (371)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 19A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (372)	C	6/09/81	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (373)	A	11/09/77	ECCS Benchboard (1H13-P601) Section 19A
B-208-222 (375)	G	9/01/81	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (376)	C	6/12/80	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (377)	A	2/28/78	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (378)	A	2/28/78	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (379)	-	8/24/77	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (380)	D	12/31/80	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (381)	A	10/24/77	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (382)	C	1/05/79	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (383)	F	12/31/80	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (384)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (385)	B	6/09/81	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (386)	B	11/14/77	ECCS Benchboard (1H13-P601) Section 20A
B-208-222 (400)	F	9/28/81	ECCS Benchboard (1H13-P601) Section 21A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (401)	C	3/17/78	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (402)	G	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (403)	F	12/31/80	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (404)	F	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (405)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (406)	E	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (407)	D	4/30/79	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (408)	E	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (409)	F	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (410)	D	6/09/81	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (411)	B	10/24/77	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (412)	A	10/24/77	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (413)	B	10/24/77	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (414)	F	12/03/80	ECCS Benchboard (1H13-P601) Section 21A
B-208-222 (426)	H	10/01/81	ECCS Benchboard (1H13-P601) Section 22A

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(465)	D	6/09/81	Control Room Annunciator Long Response BB (1H13-P870) Section 1A
B-208-222	(466)	A	10/31/80	Control Room Annunciator Long Response BB (1H13-P870) Section 1A
B-208-222	(478)	C	12/31/80	Long Response BB (1H13-P870) Section 2A
B-208-222	(601)	D	12/31/80	Common HVAC Panel (1H13-P904)
B-208-222	(602)	E	12/31/80	Common HVAC Panel (1H13-P904)
B-208-222	(603)	D	12/31/80	Common HVAC Panel (1H13-P904)
B-208-222	(604)	F	10/31/79	Common HVAC Panel (1H13-P904)
B-208-222	(605)	C	10/31/79	Common HVAC Panel (1H13-P904)
B-208-222	(606)	A	10/02/78	Common HVAC Panel (1H13-P904)
B-208-222	(608)	C	10/31/79	Common HVAC Panel (1H13-P904)
B-208-222	(609)	D	12/31/80	Common HVAC Panel (1H13-P904)
B-208-222	(610)	C	12/09/81	Common HVAC Panel (1H13-P904)
B-208-222	(611)	E	12/09/81	Common HVAC Panel (1H13-P904)
B-208-222	(612)	C	12/09/81	Common HVAC Panel (1H13-P904)
B-208-222	(613)	D	12/31/80	Common HVAC Panel (1H13-P904)

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-208-222	(614)	E	12/31/80	Common HVAC Panel (1H13-P904)
B-208-222	(615)	B	6/30/79	Common HVAC Panel (H13-P904)
B-208-222	(616)	D	10/31/79	Common HVAC Panel (H13-P904)
B-208-222	(617)	E	10/01/81	Common HVAC Panel (H13-P904)
B-208-222	(618)	C	12/31/80	Common HVAC Panel (H13-P904)
B-208-222	(619)	D	12/31/80	Common HVAC Panel (H13-P904)
B-208-222	(620)	D	12/31/80	Common HVAC Panel (H13-P904)
B-208-222	(621)	E	12/31/80	Common HVAC Panel (H13-P904)
B-208-222	(626)	C	12/31/80	Common Long Response BB (H13-P970)
B-208-222	(628)	D	12/31/80	Common Long Response BB (H13-P970)
B-208-222	(629)	D	12/31/80	Common Long Response BB (H13-P970)
B-208-222	(725)	A	9/30/77	Unit HVAC Panel (1H13-P800)
B-208-222	(726)	F	12/31/80	Unit HVAC Panel (1H13-P800)
B-208-222	(727)	C	10/31/79	Unit HVAC Panel (1H13-P800)
B-208-222	(728)	C	2/12/82	Unit HVAC Panel (1H13-P800)
B-208-222	(729)	B	6/09/81	Unit HVAC Panel (1H13-P800)
B-208-222	(730)	C	6/09/81	Unit HVAC Panel (1H13-P800)
B-208-222	(732)	C	12/31/80	Unit HVAC Panel (1H13-P800)

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (733)	-	8/24/77	Unit HVAC Panel (1H13-P800)
B-208-222 (734)	B	1/02/80	Unit HVAC Panel (1H13-P800)
B-208-222 (735)	C	12/31/80	Unit HVAC Panel (1H13-P800)
B-208-222 (736)	A	9/30/77	Unit HVAC Panel (1H13-P800)
B-208-222 (737)	C	12/31/80	Unit HVAC Panel (1H13-P800)
B-208-222 (738)	C	6/09/81	Unit HVAC Panel (1H13-P800)
B-208-222 (739)	-	8/24/77	Unit HVAC Panel (1H13-P800)
B-208-222 (740)	E	12/31/80	Unit HVAC Panel (1H13-P800)
B-208-222 (925) Panel	D	10/31/80	Div. 2 Auxiliary Relay (1H13-P868) Isolator Power Supply
B-208-222 (926)	D	10/31/80	Div. 1 Analog Loop Cab (1H13-P869) Isolator Power Supply
B-208-222 (927)	F	10/15/81	Div. 2 Auxiliary Relay Panel (1H13-P871) Isolator Power Supply
B-208-222 (928)	F	10/15/81	Div. 1 Auxiliary Relay Panel (1H13-P872) Isolator Power Supply
B-208-222 (929)	D	10/15/81	Div. 3 Auxiliary Relay Panel (1H13-P873) Isolator Power Supply
B-208-222 (930)	D	10/31/80	Common HVAC Panel (1H13-P904) Isolator Power Supply
B-208-222 (931)	A	8/07/78	Common HVAC Panel (1H13-P969) Isolator Power Supply

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-208-222 (932)	A	4/30/79	Control Room Annunciator Isolator Power Supply Distribution
B-208-229	-	10/30/77	Diesel Generator Jacket Water Index
B-208-229 (1)	B	10/17/80	Circulating Pump 1R46-C005A
B-208-229 (2)	B	10/17/80	Circulating Pump 1R46-C005B
B-208-229 (3)	C	10/17/80	Jacket Water Heater 1R46-D006A
B-208-229 (4)	C	10/17/80	Jacket Water Heater 1R46-D006B
B-208-230	-	11/30/77	Diesel Generator Lube Oil Index
B-208-230 (1)	B	10/17/80	Lube Oil Circulating Pump 1R47-C002A
B-208-230 (2)	B	10/17/80	Lube Oil Circulating Pump 1R47-C002B
B-208-230 (3)	B	10/17/80	Lube Oil Heater 1R47-C004A
B-208-230 (4)	B	10/17/80	Lube Oil Heater 1R47-C004B
D-214-001	P	4/02/81	Legend, Notes, References & Standard Details
D-214-002	H	3/04/82	Details
D-214-004	J	7/12/82	Conduit & Tray Separation Criteria
D-214-005	J	12/17/80	Conduit & Tray Separation Criteria
D-214-111	D	12/28/78	Control Complex - East – Elev. 574’-10” Unit 1 & 2
D-214-112	D	11/16/78	Control Complex - West – Elev. 574’-10” Unit 1 & 2

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-214-121	P	7/14/82	Control Complex - East – Elev. 599’-0” Unit 1 & 2
D-214-122	L	7/14/82	Control Complex - West – Elev. 599’-0” Unit 1 & 2
D-214-131	N	4/27/82	Control Complex - East – Elev. 620’-6” Unit 1 & 2
D-214-132	Q	3/05/82	Control Complex - West – Elev. 620’-6” Unit 1 & 2
D-214-141	G	3/04/82	Control Complex - East – Elev. 638’-6” Unit 1 & 2
D-214-142	J	3/05/82	Control Complex - West – Elev. 638’-6” Unit 1 & 2
D-214-143	K	4/23/82	Control Complex-Cable Chase – Elev. 638’-6” & 654’-6” Unit 1 & 2
D-214-144	H	4/27/82	Control Complex – East - Auxiliary Plan – Elev. 638’-6” Units 1 & 2
D-214-145	D	8/18/82	Control Complex - West - Auxiliary Plan – Elev. 638’-6” Units 1 & 2
D-214-161	J	4/21/81	Control Complex – East – Elev. 679’-6” Unit 1 & 2
D-214-162	F	8/11/80	Control Complex - East – Elev. 679’-6” Unit 1 & 2
D-214-221	J	9/15/82	Auxiliary Building – East – Elev. 599’-0”
D-214-222	Q	9/15/82	Auxiliary Building – West – Elev. 599’-0”
D-214-232	P	9/15/82	Auxiliary Building – West – Elev. 620’-6”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-214-331	J	10/29/80	Reactor Building – East - Elev. 620′-6″
D-214-332	G	4/13/82	Reactor Building - West - Elev. 620′-6″
D-214-341	F	9/20/82	Reactor Building – East - Elev. 642′-0″
D-214-342	L	7/12/82	Reactor Building – West - Elev. 642′-0″
D-214-351	G	9/20/82	Reactor Building - East - Elev. 652′-2″
D-214-361	J	8/20/81	Reactor Building – East - Elev. 664′-7″
D-214-362	H	4/13/82	Reactor Building - West - Elev. 664′-7″
D-214-411	F	4/15/81	Intermediate Building - North - Elev. 574′-10″ Units 1 & 2
D-214-412	D	4/15/81	Intermediate Building - South - Elev. 574′-10″ Units 1 & 2
D-214-421	M	9/20/82	Intermediate Building - North - Elev. 599′-0″ Units 1 & 2
D-214-422	N	9/20/82	Intermediate Building - South - Elev. 599′-0″ Units 1 & 2
D-214-423	C	7/08/80	Fuel Handling Area - East - Elev. 599′-0″ Units 1 & 2
D-214-424	A	7/25/77	Fuel Handling Area - West - Elev. 599′-0″ Units 1 & 2
D-214-431	T	9/20/82	5/1/81 Units 1 & 2

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
D-214-432		R	9/20/82	Intermediate Building - South - Elev. 620’-6” Units 1 & 2
D-214-434		J	9/20/82	Fuel Handling Area – West - Elev. 620’-6” Units 1 & 2
D-214-441		K	9/02/82	Intermediate Building - North - Elev. 639’-6” Units 1 & 2
D-214-442		G	9/02/82	Intermediate Building - South - Elev. 639’-6” Units 1 & 2
D-214-451		G	12/04/80	Intermediate Building - North - Elev. 654’-6” Units 1 & 2
D-214-452		F	1/29/80	Intermediate Building - South - Elev. 654’-6” Units 1 & 2
D-214-471		J	8/11/80	Intermediate Building - North - Elev. 682’-6” Units 1 & 2
D-214-472		H	4/16/81	Intermediate Building - South - Elev. 682’-6” Units 1 & 2
D-214-611		G	3/05/82	Diesel Generator Building - Elev. 620’-6” Unit 1
D-214-612		H	4/29/82	Diesel Generator Building - Elev. 620’-6” Unit 2
D-214-651		N	3/04/82	Sections & Details
D-214-652		F	3/04/82	Sections & Details
D-215-001	(1)	N	8/09/82	Legend and Notes
D-215-002	(1)	L	7/13/82	Details
D-215-002	(2)	K	5/26/82	Details

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
D-215-004	(601)	J	5/26/82	Details & References
D-215-004	(602)	B	1/25/82	Details & References
D-215-004	(603)	B	1/25/82	Details & References
D-215-002	(3)	A	5/26/82	Details
D-215-021		L	9/02/82	Turbine Building Lube Oil Area Elev. 593'-6"
D-215-022		M	3/18/82	Turbine Building - East - Elev. 605'-6"
D-215-031		M	9/02/82	Turbine Building Lube Oil Area Elev. 620'-6"
D-215-032		M	9/10/82	Turbine Building - East - Elev. 624'-6"
D-215-033		E	1/06/82	Turbine Building Elev. 624'-6"
D-215-034		K	9/02/82	Turbine Building Elev. 624'-6"
D-215-042		K	9/02/82	Turbine Building - East - Elev. 647'-6"
D-215-043		E	9/02/82	Turbine Building - Elev. 647'-6"
D-215-044		G	11/09/81	Turbine Building - West - Elev. 647'-6"
D-215-067		K	8/26/82	Offgas Building - Elev. 602'-6"
D-215-074		M	2/03/82	Condensate Demineralizer Area - East Elev. 593'-6"
D-215-081		S	8/26/82	Heater Bay - East - Elev. 620'-6"
D-215-084		R	6/21/82	Turbine Power Complex - East - Elev. 620'-6"

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-086	N	8/26/82	Offgas Building – Elev. 620′-6″
D-215-087	J	3/24/82	Offgas Building – Elev. 635′-0″, 660′-0″ & 668′-3″
D-215-111	T	9/10/82	Control Complex – East - Elev. 574′-10″ Units 1 & 2
D-215-112	R	9/10/82	Control Complex – West - Elev. 574′-10″ Units 1 & 2
D-215-121	U	5/27/82	Control Complex – East - Elev. 599′-0″ Units 1 & 2
D-215-122	M	4/28/82	Control Complex – West - Elev. 599′-0″ Units 1 & 2
D-215-131	X	4/27/82	Control Complex – East - Elev. 620′-6″ Units 1 & 2
D-215-132	Q	6/21/82	Control Complex – West - Elev. 620′-6″ Units 1 & 2
D-215-133	-	5/06/77	Embedded Conduits Control Complex Elev. 620′-6″
D-215-134	R	6/21/82	Conduit Layout Control Complex – West – Elev. 620′-6″
D-215-141	P	6/21/82	Control Complex – East - Elev. 638′-6″ Units 1 & 2
D-215-142	T	6/21/82	Control Complex – West - Elev. 638′-6″ Units 1 & 2
D-215-143	P	6/21/82	Control Complex - Cable Chase - Elev. 638′-6″ and 654′-6″
D-215-144	P	9/16/82	Control Complex – East - Elev. 638′-6″ Auxiliary Plans Unit 1

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-145	M	9/16/82	Electrical Conduit Layout Control Complex – Elev. 638’-6”
D-215-146	H	7/13/82	Electrical Conduit Layout Control Complex – Elev. 638’-6”
D-215-147	D	2/03/82	Electrical Conduit Layout Control Complex – Elev. 638’-6”
D-215-151	H	9/10/82	Control Complex – East - Elev. 654’-6” Units 1 & 2
D-215-152	H	9/10/82	Control Complex - West – Elev. 654’-6”
D-215-161	M	3/08/82	Control Complex – East - Elev. 679’-6” Units 1 & 2
D-215-162	J	3/31/82	Control Complex – West - Elev. 679’-6” Units 1 & 2
D-215-163	C	3/31/82	Control Complex Auxiliary Plans
D-215-211	M	6/21/82	Auxiliary Building – East - Elev. 574’-10”
D-215-212	N	8/05/82	Auxiliary Building – West - Elev. 574’-10”
D-215-221	K	6/21/82	Auxiliary Building – East – Elev. 599’-0”
D-215-222	L	8/05/82	Auxiliary Building - West – Elev. 599’-0”
D-215-231	F	3/02/82	Auxiliary Building – East – Elev. 620’-6”
D-215-232	L	8/05/82	Auxiliary Building – West – Elev. 620’-6”

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-233	G	6/21/82	Conduit Layout Auxiliary Building- Steam Tunnel- Elev. 614’-6” & 620’-6”
D-215-311	M	9/28/81	Reactor Building – East - Elev. 574’-10”
D-215-312	P	7/13/82	Reactor Building – West - Elev. 574’-10”
D-215-313	F	2/28/81	Auxiliary Plan - Under Reactor Pressure Vessel Elev. 594’-5-3/16”
D-215-314	D	8/31/81	Sections & Details-Under Reactor Pressure Vessel Elev. 594’-5-3/16”
D-215-321	Q	7/13/82	Reactor Building – East - Elev. 599’-9”
D-215-322	P	7/13/82	Reactor Building – West - Elev. 599’-9”
D-215-331	Q	7/13/82	Reactor Building – East - Elev. 620’-6”
D-215-332	M	7/13/82	Reactor Building – West - Elev. 620’-6”
D-215-341	G	3/22/82	Reactor Building – East - Elev. 642’-0”
D-215-342	J	6/31/82	Reactor Building – West - Elev. 642’-0”
D-215-351	G	6/21/82	Reactor Building – East - Elev. 652’-2”
D-215-352	H	6/21/82	Reactor Building – West - Elev. 652’-2”
D-215-361	E	3/22/82	Reactor Building – East - Elev. 664’-7”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-362	F	11/30/81	Reactor Building – West - Elev. 664’-7”
D-215-371	D	11/30/81	Reactor Building – East - Elev. 689’-6”
D-215-372	C	5/13/81	Reactor Building – West - Elev. 689’-6”
D-215-373	G	8/31/81	Reactor Building – Fuel Pool Area Elev. 689’-6” – Embedded Conduit
D-215-374	D	8/04/81	Embedded Conduit Details Reactor Building-Fuel Pool Area - Elev. 689’-6”
D-215-411	P	9/10/82	Intermediate Building – North - Elev. 574’-10”
D-215-412	J	4/27/82	Intermediate Building – South - Elev. 574’-10”
D-215-413	J	5/07/82	Fuel Handling Area – East – Elev. 574’-10” Units 1 & 2
D-215-414	L	7/13/82	Fuel Handling Area – West - Elev. 574’-10” Units 1 & 2
D-215-421	N	6/07/82	Intermediate Building – North - Elev. 599’-0”
D-215-422	H	6/07/82	Intermediate Building – South - Elev. 599’-0”
D-215-423	F	9/18/81	Fuel Handling Area – East - Elev. 599’-0” Units 1 & 2
D-215-424	J	7/13/82	Fuel Handling Area – West - Elev. 599’-0” Units 1 & 2
D-215-431	N	7/13/82	Intermediate Building – North - Elev. 620’-6”
D-215-432	K	6/21/82	Intermediate Building – South - Elev. 620’-6”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-433	G	2/03/82	Fuel Handling Building – East - Elev. 620’-6” Units 1 & 2
D-215-434	K	6/07/82	Fuel Handling Area – West - Elev. 620’-6” Units 1 & 2
D-215-435	E	9/18/81	Fuel Handling Area - Elev. 620’-6” Units 1 & 2 – Embedded Conduit
D-215-436	E	5/20/81	Fuel Handling Area Elev. 620’-6” Embedded Conduit - Details
D-215-441	P	9/10/82	Intermediate Building – North - Elev. 639’-6”
D-215-442	L	7/13/82	Intermediate Building – South - Elev. 639’-6”
D-215-443	J	1/20/82	Conduit Layout Penetration Access Area - Sections & Details
D-215-444	G	7/13/82	Conduit Layout Penetration Access Area - Sections & Details
D-215-445	Q	6/21/82	Conduit Layout Penetration Access Area - Sections & Details
D-215-451	J	6/21/82	Intermediate Building – North - Elev. 654’-6” & 665’-0”
D-215-452	G	3/31/82	Intermediate Building – South - Elev. 654’-6” & 665’-0”
D-215-471	E	2/03/82	Intermediate Building – North - Elev. 682’-6”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-472	E	2/03/82	Intermediate Building – South - Elev. 682’-6”, 721’-6” & 707’-6”
D-215-522	K	4/02/82	Radwaste Building – Elev. 602’-0”
D-215-523	J	6/21/82	Radwaste Building - West – Elev. 602’-0”
D-215-611	K	7/13/82	Diesel Generator Building – Div. 2 - Elev. 620’-6” Unit 1
D-215-612	J	8/05/82	Diesel Generator Building – Div. 3 - Elev. 620’-6” Unit 1
D-215-613	G	2/06/80	Embedded Conduit - Diesel Generator Building Elev. 620’-6” Unit 1
D-215-614	G	2/06/80	Embedded Conduit - Diesel Generator Building Elev. 620’-6” Unit 2
D-215-615	K	6/21/82	Diesel Generator Building Div. 1 - Elev. 620’-6” Unit 1
D-215-616	H	8/26/82	Diesel Generator Building Div. 2 - Elev. 620’-6” Unit 2
D-215-617	G	6/21/82	Diesel Generator Building Div. 3 - Elev. 620’-6” Unit 2
D-215-618	G	8/05/82	Diesel Generator Building Div. 1 - Elev. 620’-6” Unit 2
D-215-621	C	9/10/82	Diesel Generator Building - Elev. 646’-6” - Unit 1

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-622	C	9/10/82	Diesel Generator Building - Elev. 646'-6" - Unit 2
D-215-651	K	6/21/82	Sections & Details
D-215-652	M	8/05/82	Sections & Details
D-215-653	M	5/07/82	Sections & Details
D-215-654	B	11/15/80	Sections & Details
D-215-655	J	8/05/82	Sections & Details
D-215-656	P	7/13/82	Sections & Details
D-215-657	N	9/16/82	Sections & Details
D-215-658	P	9/16/82	Sections & Details
D-215-659	E	3/02/82	Sections & Details
D-215-660	G	7/13/82	Sections & Details
D-215-661	N	9/16/82	Control Complex – Sections and Details
D-215-662	E	8/06/81	Control Complex - Sections and Details
D-215-663	J	6/21/82	Control Complex - Sections and Details
D-215-664	H	6/21/82	Control Complex - Sections and Details
D-215-665	D	7/13/82	Sections & Details
D-215-666	B	2/28/81	Sections & Details
D-215-667 (501) <Figure 8.3-19>	K	1/06/98	Sections & Details
D-215-668	F	1/06/82	Sections & Details
D-215-669	F	1/06/82	Sections & Details

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-215-670	D	7/13/82	Sections & Details
D-215-671	E	3/31/82	Intermediate Building Sections and Details
D-215-711	F	3/17/82	Manholes & Underground Duct Runs - Plan Units 1 & 2
D-215-712	C	12/23/81	Underground Duct Runs Sections 1-1 thru 31-31 Units 1 & 2
D-215-713	F	3/17/82	Underground Duct Runs Sections 51-51 thru 80-80 Units 1 & 2
D-215-716	-	8/03/79	Conduit Layout Electrical Manhole No. 1 Cable Racking Details
D-215-717	-	8/03/79	Conduit Layout Electrical Manhole No. 2 Cable Racking Details
D-215-718	-	8/03/79	Conduit Layout Electrical Manhole No. 3 Cable Racking Details
D-215-719	-	8/03/79	Conduit Layout Electrical Manhole No. 4 Cable Racking Details
D-215-720	-	8/03/79	Conduit Layout Electrical Manhole No. 18 Cable Racking Details
D-216-001	L	6/29/82	Underground Duct Runs Notes, Legend, References
D-216-002	F	6/29/82	Underground Duct Runs Units 1 & 2 Miscellaneous Sections & Details
D-216-011	P	6/29/82	Plan Units 1 & 2

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-216-012	D	3/12/81	Underground Duct Runs Unit 1 Between Radwaste Building & Turbine Power Complex
D-216-013	D	6/29/82	Underground Duct Runs Units 1 & 2 Diesel Generator Building
D-216-014	G	6/29/82	Underground Duct Runs Unit 2 Between Service Building & Turbine Power Complex
D-216-016	D	4/26/80	Underground Duct Runs Div. 1 to Emergency Service Water Pumphouse East of Plant
D-216-017	C	11/15/78	Underground Duct Runs Div. 2 & Div. 3 to Emergency Service Water Pumphouse West of Plant
D-216-020	B	6/29/82	Underground Duct Runs Service to Under Drain Manholes Nos. 3, 9, 10, 11 & 23
D-216-028	D	6/29/82	Underground Duct Runs Units 1 & 2 Miscellaneous Duct Runs East Side of Plant
D-216-029	F	11/21/80	Underground Duct Runs Miscellaneous Duct Runs
D-218-004 <Figure 9.5-23>	K	2/27/01	Maintenance and Calibration System Device List
D-218-106 <Figure 9.5-6 (1)>	J	6/18/92	Schematic Diagram Offsite Communications
D-218-111 <Figure 9.5-6 (2)>	C	1/17/91	Communication System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)		Revision	Date	Title
B-224-301		L	11/06/81	Containment Vessel Penetration Schedule Unit 1
D-226-511		S	9/16/82	Conduit Layout – Pumphouse - Elev. 586’-6”
D-226-512		K	8/05/82	Conduit Layout Emergency Service Water Pumphouse Sections
D-226-532		B	7/02/80	Embedded Conduit – Elev. 586’-6”
D-230-002		E	8/19/82	Conduit Layout Yard Area
B-258-132		-	8/25/77	M40 Fuel Handling Building Ventilation Index
B-258-132	(1)	-	8/25/77	Exhaust Fan C M40-C002C
B-258-132	(2)	B	2/25/81	Heating Coil D001C
B-258-132	(3)	A	12/14/78	Heating Coil D001C (Continued)
B-258-173		A	11/13/81	P42 Emergency Closed Cooling Index
B-258-178		A	11/09/81	P47 Control Complex Chilled Water Index
B-258-178	(1)	E	7/20/82	Control Complex Chiller “C” B001C
B-258-178	(2)	B	1/07/81	Chilled Water Pump “C” C001C
B-258-178	(3)	F	11/05/81	Chiller “C” Controls B001C
B-258-178	(4)	C	12/31/80	Chiller “C” Controls B001C (Continued)

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
B-258-178 (5)	C	11/09/81	Chiller "C" Oil Pump B001C
D-264-001	K	4/22/81	Electrical Cable Tray Layout, Legend Notes, References, and Standard Details Unit 2
D-264-002	H	3/04/82	Electrical Cable Tray Layout Details
D-264-004	F	7/12/82	Electrical Conduit & Tray Separation Criteria
D-264-222	K	9/15/82	Electrical Cable & Tray Layout Auxiliary Building – West – Elev. 599'-0"
D-264-232	M	9/15/82	Electrical Cable Tray Layout Auxiliary Building – West - Elev. 620'-6"
D-264-331	E	2/02/82	Reactor Building - East – Elev. 620'-6"
D-264-332	E	2/02/82	Electrical Cable Tray Layout Reactor Building - West - Elev. 620'-6"
D-264-341	C	4/07/81	Reactor Building - East – Elev. 642'-0"
D-264-342	D	9/15/82	Electrical Cable Tray Layout Reactor Building - West - Elev. 642'-0"
D-264-352	D	9/15/82	Electrical Cable Tray Layout Reactor Building - West - Elev. 652'-2"
D-264-361	D	10/28/81	Electrical Cable Tray Layout Reactor Building - East - Elev. 664'-7"

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-264-362	E	9/15/82	Electrical Cable Tray Layout Reactor Building - West - Elev. 664’-7”
D-265-031	D	9/10/82	Turbine Building Lube Oil Area - Elev. 620’-6”
D-265-042	E	9/10/82	Turbine Building – Elev. 647’-6”
D-265-044	C	6/21/82	Turbine Building - West – Elev. 647’-6”
D-265-074	K	2/03/82	Condensate Demineralizer Area - East - Elev. 593’-0”
D-265-081	G	6/21/82	Heater Bay - East – Elev. 620’-6”
D-265-084	K	6/21/82	Turbine Power Complex – East - Elev. 620’-6”
D-265-211	E	3/18/82	Auxiliary Building - East – Elev. 574’-10”
D-265-212	F	3/18/82	Electrical Conduit Layout Auxiliary Building – West - Elev. 574’-10”
D-265-221	F	3/18/82	Auxiliary Building - East – Elev. 599’-0”
D-265-222	E	3/18/82	Electrical Conduit Layout Auxiliary Building – West - Elev. 599’-0”
D-265-231	C	3/18/82	Auxiliary Building - East – Elev. 620’-6”
D-265-232	D	3/18/82	Auxiliary Building - West – Elev. 620’-6”
D-265-233	B	3/18/82	Auxiliary Building - Steam Tunnel - Elev. 614’-6” and Elev. 620’-6”

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-265-373	B	5/20/81	Reactor Building - Fuel Pool Area - Elev. 689’-6”
D-265-374	B	8/04/81	Reactor Building - Fuel Pool Area - Elev. 689’-6”
D-265-651	G	3/18/82	Sections and Details
D-265-655	E	3/18/82	Sections and Details
D-265-656	E	3/18/82	Electrical Conduit Layout Sections & Details
B-274-301	M	11/06/81	Containment Vessel Penetration Schedule Unit 2
D-806-001 <Figure 11.3-3>	F	3/22/01	Plant Radiation Monitoring (Airborne)
D-806-006 <Figure 11.5-1 (1)>	F	3/02/87	Plant Radiation Monitoring Subsystems K660, K690A, K690B
D-806-007 <Figure 11.5-1 (2)>	H	3/22/01	Plant Radiation Monitoring Subsystems K680, K780, K790
D-806-008 <Figure 11.5-1 (3)>	G	3/22/01	Plant Radiation Monitoring Subsystems K800, K830, K840
D-806-009 <Figure 11.5-1 (4)>	K	3/01/94	Liquid System Radiation Monitoring
D-806-010 <Figure 11.5-1 (5)>	G	5/02/00	Liquid System Radiation Monitoring
D-806-017 <Figure 11.5-1 (6)>	F	3/24/87	Under Drain Monitors K820A, K820B
D-806-018 <Figure 11.5-1 (7)>	G	10/21/91	Offgas Pretreatment Radiation Monitor Subsystem
D-806-019 <Figure 11.5-1 (8)>	F	10/21/91	Offgas Post Treatment Radiation Monitor Subsystem
D-806-022 <Figure 11.5-1 (9)>	J	6/30/94	Automatic Isokinetic Sampling System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-806-023 <Figure 11.5-1 (10)>	H	10/28/91	Isokinetic Sampling System
D-806-024 <Figure 11.5-1 (11)>	H	7/19/94	Containment Ventilation Exhaust and MSL Radiation Monitoring Subsystem
D-806-025 <Figure 11.5-1 (12)>	G	10/21/91	Carbon Bed Vault Radiation Monitoring Subsystem
D-808-303 (1) <Figure 7.3-3 (1)>	D	10/14/99	Nuclear Boiler System
D-808-303 (2) <Figure 7.3-3 (2)>	B	10/14/99	Nuclear Boiler System
D-808-303 (3) <Figure 7.3-3 (3)>	A	12/15/89	Nuclear Boiler System
D-808-303 (4) <Figure 7.3-3 (4)>	B	8/10/94	Nuclear Boiler System
D-808-303 (5) <Figure 7.3-3 (5)>	B	9/22/99	Nuclear Boiler System
D-808-303 (6) <Figure 7.3-3 (6)>	A	7/16/92	Nuclear Boiler System
D-808-303 (7) <Figure 7.3-3 (7)>	B	9/22/99	Nuclear Boiler System
D-808-304 (1) <Figure 7.7-4 (1)>	A	6/08/92	Reactor Recirculation System
D-808-304 (2) <Figure 7.7-4 (2)>	-	6/14/88	Reactor Recirculation System
D-808-304 (3) <Figure 7.7-4 (3)>	-	6/14/88	Reactor Recirculation System
D-808-304 (4) <Figure 7.7-4 (4)>	-	6/14/88	Reactor Recirculation System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-808-304 (5) <Figure 7.7-4 (5)>	-	6/14/88	Reactor Recirculation System
D-808-304 (6) <Figure 7.7-4 (6)>	-	6/14/88	Reactor Recirculation System
D-808-304 (7) <Figure 7.7-4 (7)>	A	12/15/89	Reactor Recirculation System
D-808-305 (1) <Figure 7.7-1 (1)>	A	3/31/92	Control Rod Drive Hydraulic System
D-808-305 (2) <Figure 7.7-1 (2)>	-	6/13/88	Control Rod Drive Hydraulic System
D-808-305 (3) <Figure 7.7-1 (3)>	-	6/14/88	Control Rod Drive Hydraulic System
D-808-305 (4) <Figure 7.7-1 (4)>	-	6/14/88	Control Rod Drive Hydraulic System
D-808-305 (5) <Figure 7.7-1 (5)>	-	6/14/88	Control Rod Drive Hydraulic System
D-808-305 (6) <Figure 7.7-1 (6)>	E	8/15/96	Control Rod Drive Hydraulic System
D-808-305 (7) <Figure 7.7-1 (7)>	-	6/14/88	Control Rod Drive Hydraulic System
D-808-306 (1) <Figure 7.4-2 (1)>	A	12/15/89	Standby Liquid Control System
D-808-306 (2) <Figure 7.4-2 (2)>	-	6/14/88	Standby Liquid Control System
D-808-307 (1) <Figure 7.6-2 (1)>	B	3/16/01	Neutron Monitoring System
D-808-307 (2) <Figure 7.6-2 (2)>	A	9/10/91	Neutron Monitoring System
D-808-307 (3) <Figure 7.6-2 (3)>	A	9/10/91	Neutron Monitoring System

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-808-307 (4) <Figure 7.6-2 (4)>	B	10/14/99	Neutron Monitoring System
D-808-307 (5) <Figure 7.6-2 (5)>	C	8/15/96	Neutron Monitoring System
D-808-307 (6) <Figure 7.6-2 (6)>	A	9/10/91	Neutron Monitoring System
D-808-307 (7) <Figure 7.6-2 (7)>	C	8/15/96	Neutron Monitoring System
D-808-309 (1) <Figure 7.3-5 (1)>	B	4/05/00	Residual Heat Removal System
D-808-309 (2) <Figure 7.3-5 (2)>	E	11/29/00	Residual Heat Removal System
D-808-309 (3) <Figure 7.3-5 (3)>	C	7/05/94	Residual Heat Removal System
D-808-309 (4) <Figure 7.3-5 (4)>	E	10/27/99	Residual Heat Removal System
D-808-309 (5) <Figure 7.3-5 (5)>	E	5/02/00	Residual Heat Removal System
D-808-310 (1) <Figure 7.3-4 (1)>	A	12/15/89	Low Pressure Core Spray System
D-808-310 (2) <Figure 7.3-4 (2)>	A	8/10/94	Low Pressure Core Spray System
D-808-311 (1) <Figure 7.3-1 (1)>	-	6/14/88	High Pressure Core Spray System
D-808-311 (2) <Figure 7.3-1 (2)>	-	6/14/88	High Pressure Core Spray System
D-808-311 (3) <Figure 7.3-1 (3)>	-	6/14/88	High Pressure Core Spray System
D-808-311 (4) <Figure 8.3-8>	D	7/13/94	HPCS Diesel, Diesel Breakers Alternate Preferred Supply Breakers Logic Diagram, Division 3

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-808-313 (1)	C	8/26/99	MSIV Leakage Control System
D-808-313 (2)	C	8/26/99	MSIV Leakage Control System
D-808-313 (3)	A	8/26/99	MSIV Leakage Control System
D-808-313 (4)	C	8/26/99	MSIV Leakage Control System
D-808-314 (1) <Figure 7.4-1 (1)>	F	7/23/96	Reactor Core Isolation Cooling System
D-808-314 (2) <Figure 7.4-1 (2)>	B	10/04/93	Reactor Core Isolation Cooling System
D-808-314 (3) <Figure 7.4-1 (3)>	C	11/13/97	Reactor Core Isolation Cooling System
D-808-314 (4) <Figure 7.4-1 (4)>	A	7/16/92	Reactor Core Isolation Cooling System
D-808-314 (5) <Figure 7.4-1 (5)>	B	10/04/93	Reactor Core Isolation Cooling System
D-808-315 <Figure 7.3-6>	C	6/07/94	Reactor Water Cleanup System
D-808-317 (1) <Figure 8.3-6 (1)>	D	11/29/99	Diesel Logic Diagrams Division 1
D-808-317 (2) <Figure 8.3-6 (2)>	D	1/13/00	Diesel Logic Diagrams Division 2
D-808-317 (3) <Figure 8.3-9>	D	7/13/94	Diesel Breakers, Preferred, Alternate Preferred and Stub Bus Logic Diagram Division 1, (Division 2)
D-809-023	D	5/17/78	Diesel Generator Benchboard 1H13-P877 Sections 1 & 2 - Front View

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-809-041	G	8/04/78	Heating, Ventilation & Air Conditioning Panel 1H13-P800 - Front View
D-809-042	E	3/22/78	Heating, Ventilation & Air Conditioning Panel 1H13-P800 - Front View
D-809-043	D	5/14/77	Heating, Ventilation & Air Conditioning Panel 1H13-P800 - Front View
D-809-044	C	12/20/76	Heating, Ventilation & Air Conditioning Panel 1H13-P800 - Front View
D-809-045	D	5/14/77	Heating, Ventilation & Air Conditioning Panel 1H13-P800 - Front View
D-809-051	B	5/13/77	Containment & Drywell Isolation Valve Status Lights
D-809-052	C	11/21/76	Process Radiation Monitoring Panel 1H13-P604 Front View
D-809-053	D	5/13/77	Airborne Radiation Monitoring Panel 1H13-P804 Front View
D-809-054	B	7/01/76	Airborne Radiation Monitoring Panel 1H13-P804 Front View
D-809-055	A	7/01/76	Airborne Radiation Monitoring Panel 1H13-P804 Front View
D-809-056	A	7/01/76	Area Radiation Monitoring Panel 1H13-P803 Front View
D-809-057	-	1/02/76	Fire & Security Console 1H13-P802 Isometric View & Details

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-809-059	-	1/02/76	Division 3 Auxiliary Relay Panel 1H13-P873 Front View
D-809-064	F	9/22/78	Analog Loop Div. 2 Instrument Panel 1H13-P868 - Front View
D-809-065	F	9/22/78	Analog Loop Div. 2 Instrument Panel 1H13-P868 - Front View
D-809-066	F	9/22/78	Analog Loop Div. 1 Instrument Panel 1H13-P869 - Front View
D-809-067	F	9/22/78	Analog Loop Div. 1 Instrument Panel 1H13-P869 - Front View
D-809-068	C	11/03/78	Div. 1 Containment & Drywell Isolation Valve Control Panel 1H13-P881 - Front View
D-809-069	C	11/03/78	Div. 2 Containment & Drywell Isolation Valve Control Panel 1H13-P882 - Front View
D-809-070	C	2/28/78	Postaccident Monitoring Recorder Panel 1H13-P883 - Front View
B-809-071	B	7/24/78	Local Control Panel Details
B-809-073	A	3/01/78	Pump Room Cooling HVAC Control Panel 1H51-P037
B-809-076	B	3/22/79	HVAC Control Panel H51-P177A - Front View
B-809-081	B	3/22/79	HVAC Control Panel H51-P177B - Front View

TABLE 1.7-1 (Continued)

Dwg. No. (Sh. No.) (CEI No.)	Revision	Date	Title
D-809-095	C	8/07/78	Common Analog Loop Instrument & Auxiliary Relay Panel H13-P969 Front View
D-809-096	C	8/07/78	Common Analog Loop Instrument & Auxiliary Relay Panel H13-P969 Front View
D-814-663 <Figure 3.7-17>	K	10/10/01	Seismic Instrumentation Installation Details

TABLE 1.7-2

PIPING AND INSTRUMENTATION DIAGRAMS USED IN THE USAR

Drawing No. (GE No.)	Title	USAR Reference
D-302-871 (767E673CA)	Control Rod Drive Hydraulic System	<Figure 4.6-5 (1)>
D-302-872 (767E673CA)	Control Rod Drive Hydraulic System	<Figure 4.6-5 (2)>
D-302-605 (769E305CA-2)	Nuclear Boiler System	<Figure 5.1-3 (1)>
D-302-606 (769E305CA-3)	Nuclear Boiler System	<Figure 5.1-3 (2)>
D-302-607 (769E305CA-5)	Nuclear Boiler System	<Figure 5.1-3 (3)>
D-302-608 (769E305CA-6)	Nuclear Boiler System	<Figure 5.1-3 (4)>
768E324	Nuclear Boiler System	<Figure 5.2-11 (1)>
768E324	Nuclear Boiler System	<Figure 5.2-11 (2)>
D-302-601 (796E369)	Reactor Water Recirculation System	<Figure 5.4-2 (1)>
D-302-602 (796E369)	Reactor Water Recirculation System	<Figure 5.4-2 (2)>
D-302-603 (796E369A)	Reactor Water Recirculation System	<Figure 5.4-2 (3)>
D-302-604 (796E369A)	Reactor Water Recirculation System	<Figure 5.4-2 (4)>
D-302-631 (112D3192-1)	Reactor Core Isolation Cooling System	<Figure 5.4-9 (1)>
D-302-632 (112D3192-2)	Reactor Core Isolation Cooling System	<Figure 5.4-9 (2)>
762E421CA	Reactor Core Isolation Coolant System	<Figure 5.4-10>
D-302-641 (762E424CA-1)	Residual Heat Removal System	<Figure 5.4-13 (1)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-642 (762E424CA-2)	Residual Heat Removal System	<Figure 5.4-13 (2)>
D-302-643 (762E424CA-3)	Residual Heat Removal System	<Figure 5.4-13 (3)>
762E425CA1	Residual Heat Removal System	<Figure 5.4-14 (1)>
762E425CA2	Residual Heat Removal System	<Figure 5.4-14 (2)>
762E425CA3	Residual Heat Removal System	<Figure 5.4-14 (3)>
D-302-671 (105D5594-1)	Reactor Water Cleanup System	<Figure 5.4-16 (1)>
D-302-672 (105D5594-2)	Reactor Water Cleanup System	<Figure 5.4-16 (2)>
D-302-675 (794E766)	Filter/Demineralizer System (Reactor Water Cleanup System)	<Figure 5.4-19>
	RWCU Main Flow Piping Inside Containment and Drywell	<Figure 6.2-27>
D-304-646	Residual Heat Removal System Plan and Section - West	<Figure 6.2-55>
D-304-647	Residual Heat Removal System Plan and Section - East	<Figure 6.2-56>
D-300-761	Containment and Drywell Isolation	<Figure 6.2-60 (1)>
D-300-762	Containment and Drywell Isolation	<Figure 6.2-60 (2)>
D-300-763	Containment and Drywell Isolation	<Figure 6.2-60 (3)>
D-300-764	Containment and Drywell Isolation	<Figure 6.2-60 (4)>
D-302-831	Combustible Gas Control System	<Figure 6.2-62>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-811	Containment Integrated Leak Rate Testing System	<Figure 6.2-65>
D-302-686	Suppression Pool Makeup System	<Figure 6.2-67>
D-302-574	ECCS Suction Strainer	<Figure 6.2-83>
762E455CA	High Pressure Core Spray System Process Diagram	<Figure 6.3-1>
762E467CA	Low Pressure Core Spray System Process Diagram	<Figure 6.3-2>
762E425CA	Residual Heat Removal System Process Diagram	<Figure 6.3-3 (1)>
762E425CA	Residual Heat Removal System Process Diagram	<Figure 6.3-3 (2)>
762E425CA	Residual Heat Removal System Process Diagram	<Figure 6.3-3 (3)>
D-302-701 (795E873)	High Pressure Core Spray System	<Figure 6.3-7>
D-302-705 (105D5593)	Low Pressure Core Spray System	<Figure 6.3-8>
D-912-610	Control Room HVAC and Emergency Recirculation Systems	<Figure 6.4-1 (1)>
D-912-611	Notes and Operating Data for <Figure 6.4-1> and <Figure 9.4-1>	<Figure 6.4-1 (2)>
D-912-605	Annulus Exhaust Gas Treatment	<Figure 6.5-1>
D-302-661	Containment Spray System	<Figure 6.5-3>
D-302-271	Safety-Related Instrument Air System	<Figure 6.8-1>
D-302-971	Feedwater Leakage Control System	<Figure 6.9-1>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-832	Hydrogen Analysis System	<Figure 7.3-8>
D-912-606	Drywell and Containment Vacuum Relief System	<Figure 7.3-10>
D-302-961 (762E293CA)	Leak Detection System	<Figure 7.6-1 (1)>
D-302-962 (762E293CA)	Leak Detection System	<Figure 7.6-1 (2)>
D-302-963 (762E293CA)	Leak Detection System	<Figure 7.6-1 (3)>
D-302-964 (762E293CA)	Leak Detection System	<Figure 7.6-1 (4)>
D-302-881	Containment Atmosphere Monitoring System	<Figure 7.6-7>
D-302-651	Fuel Pool Cooling and Cleanup System	<Figure 9.1-9 (1)>
D-302-653	Fuel Pool Filter Demineralizer System	<Figure 9.1-9 (2)>
D-302-654	Fuel Pool Transfer Tank Drain Tank System	<Figure 9.1-9 (3)>
D-302-655	Fuel Pool Storage and Transfer System	<Figure 9.1-9 (4)>
D-302-791	Emergency Service Water System	<Figure 9.2-1 (1)>
D-302-792	Emergency Service Water System	<Figure 9.2-1 (2)>
D-302-793	Emergency Service Water System	<Figure 9.2-1 (3)>
E-304-791	Emergency Service Water Plan, Yard Area, Units 1 & 2	<Figure 9.2-2 (1)>
E-304-792	Emergency Service Water Profile and Sections, Yard Area, Units 1 & 2	<Figure 9.2-2 (2)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-621	Emergency Closed Cooling System	<Figure 9.2-3 (1)>
D-302-622	Emergency Closed Cooling System	<Figure 9.2-3 (2)>
D-302-623	Emergency Closed Cooling System	<Figure 9.2-3 (3)>
D-352-621	Emergency Closed Cooling System	<Figure 9.2-3 (4)>
D-302-611	Nuclear Closed Cooling System	<Figure 9.2-4 (1)>
D-302-612	Nuclear Closed Cooling System	<Figure 9.2-4 (2)>
D-352-612	Nuclear Closed Cooling System	<Figure 9.2-4 (3)>
D-302-613	Nuclear Closed Cooling System	<Figure 9.2-4 (4)>
D-352-613	Nuclear Closed Cooling System	<Figure 9.2-4 (5)>
D-302-713	Mixed Bed Demineralizer Distribution System	<Figure 9.2-5>
D-302-711	Two Bed Demineralizer and Distribution System Storage and North Zone Distribution	<Figure 9.2-6>
D-302-172	Two Bed Demineralizer and Distribution System Regeneration Facilities	<Figure 9.2-7>
D-302-171	Two Bed Demineralizer and Distribution System Cation and Anion Exchangers	<Figure 9.2-8>
D-302-161	Makeup Water System - Pretreatment	<Figure 9.2-9 (1)>
D-302-162	Makeup Water System - Pretreatment	<Figure 9.2-9 (2)>
D-300-060 (2)	Ultimate Heat Sink, Unit 1 Prior to Unit 2 Operation	<Figure 9.2-10>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-102	Condensate Transfer and Storage System	<Figure 9.2-13>
D-302-212	Service Water System Unit	<Figure 9.2-14>
D-302-221	Turbine Building Closed Cooling System	<Figure 9.2-15 (1)>
D-302-222	Turbine Building Closed Cooling System	<Figure 9.2-15 (2)>
D-302-223	Turbine Building Closed Cooling System	<Figure 9.2-15 (3)>
D-302-241	Service and Instrument Air Supply	<Figure 9.3-1 (1)>
D-352-241	Service and Instrument Air Supply	<Figure 9.3-1 (2)>
D-911-005	Lube Oil Area and Turbine Location Drains	<Figure 9.3-5>
D-911-021	Turbine Power Complex Turbine Building, Heater Bay and Offgas Drains, Unit 1	<Figure 9.3-6>
D-911-022	Turbine Power Complex	<Figure 9.3-7>
D-911-023	Turbine Power Complex	<Figure 9.3-8>
D-911-024	Heater Bay Building Drains	<Figure 9.3-9>
D-911-601	Reactor Building Drains	<Figure 9.3-10>
D-911-617	Auxiliary Building Dirty Radwaste Drains	<Figure 9.3-11>
D-911-627	Intermediate Building Clean Radwaste Drains	<Figure 9.3-12>
D-911-628	Intermediate Building Dirty Radwaste Floor and Equipment Drains, Units 1 & 2	<Figure 9.3-13>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-911-629	Intermediate Building Dirty Radwaste Floor and Equipment Drains, Units 1 & 2	<Figure 9.3-14>
D-911-651	Radwaste Building Dirty Radwaste Floor and Equipment Drains	<Figure 9.3-15>
D-911-652	Radwaste Building Clean and Dirty Equipment Drains	<Figure 9.3-16>
D-911-671	Control Complex Dirty Radwaste Floor & Equipment Drain	<Figure 9.3-17>
D-911-691	Diesel Generator Building Drain	<Figure 9.3-18>
D-302-691 (762E433CA)	Standby Liquid Control System	<Figure 9.3-19 (1)>
D-302-692	Standby Liquid Control System	<Figure 9.3-19 (2)>
D-302-180	Turbine Plant Sampling System	<Figure 9.3-21>
D-302-181	Turbine Plant Sampling System	<Figure 9.3-22>
D-302-182	Turbine Plant Sampling System	<Figure 9.3-23>
D-302-183	Turbine Plant Sampling System	<Figure 9.3-24>
D-302-184	Turbine Plant Sampling System	<Figure 9.3-25>
D-302-185	Turbine Plant Sampling System	<Figure 9.3-26>
D-302-186	Turbine Plant Sampling System	<Figure 9.3-26a>
D-302-771	Nuclear Sampling System	<Figure 9.3-27>
D-302-772 (769E336-3)	Reactor Plant Sampling	<Figure 9.3-28>
D-302-242	Service Air Distribution	<Figure 9.3-29>
D-302-243	Instrument Air	<Figure 9.3-31 (1)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-244	Parallel Instrument Air Distribution System	<Figure 9.3-31 (2)>
D-302-431	Postaccident Sampling System	<Figure 9.3-33>
D-302-077	Hydrogen Water Chemistry System	<Figure 9.3-35>
D-912-609	MCC Switchgear and Miscellaneous Equipment Areas HVAC System	<Figure 9.4-1 (1)>
D-912-611	Notes and Operating Data for <Figure 6.4-1> and <Figure 9.4-1>	<Figure 9.4-1 (2)>
D-912-608	Controlled Access and Miscellaneous Equipment Areas HVAC System	<Figure 9.4-2>
D-912-607	Computer Rooms HVAC System	<Figure 9.4-3>
D-912-617	Fuel Handling Ventilation System	<Figure 9.4-4>
D-912-615	Auxiliary Building Ventilation System	<Figure 9.4-5>
D-912-625	Steam Tunnel Cooling System	<Figure 9.4-6>
D-912-612	Radwaste Building Ventilation System	<Figure 9.4-7>
D-912-614	Turbine Building Ventilation System	<Figure 9.4-8>
D-912-621	Heater Bay Ventilation System	<Figure 9.4-9>
D-912-622	Offgas Building Exhaust	<Figure 9.4-10>
D-912-630	Emergency Service Water Pumphouse Ventilation System	<Figure 9.4-11>
D-912-623	Emergency Closed Cooling Pump Area Cooling System	<Figure 9.4-12>
D-912-616	ECCS Pump Rooms Cooling Systems	<Figure 9.4-13>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-912-619	Diesel Generator Building Ventilation System	<Figure 9.4-14>
D-912-603	Drywell Cooling System	<Figure 9.4-15>
D-912-602	Containment Vessel Cooling System	<Figure 9.4-16>
D-912-604	Containment Vessel and Drywell Purge System	<Figure 9.4-17>
D-912-613	Intermediate Building Ventilation System	<Figure 9.4-18>
D-912-618	Turbine Power Complex Ventilation System	<Figure 9.4-19>
D-913-001	Control Complex Chilled Water System	<Figure 9.4-20 (1)>
D-913-002	Control Complex Chilled Water System	<Figure 9.4-20 (2)>
D-913-003	Turbine Building Chilled Water System	<Figure 9.4-21 (1)>
D-913-004	Turbine Building Chilled Water System	<Figure 9.4-21 (2)>
D-913-007	Containment Vessel Chilled Water System	<Figure 9.4-22 (1)>
D-913-008	Containment Vessel Chilled Water System	<Figure 9.4-22 (2)>
D-913-014	Hot Water Heating System, Heater Bay and Auxiliary Boiler Building	<Figure 9.4-23 (1)>
D-913-015	Hot Water Heating System, Turbine Building, Water Treatment Building and Turbine Lube Oil System, Unit 1	<Figure 9.4-23 (2)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-913-016	Hot Water Heating System, Turbine Power Complex, Auxiliary Building and Offgas Building, Unit 1	<Figure 9.4-23 (3)>
D-912-624	Offgas Charcoal Vault Refrigeration System	<Figure 9.4-24 (1)>
D-913-009	Offgas Charcoal Vault Refrigeration System Chilled Liquid Diagram	<Figure 9.4-24 (2)>
D-913-010	Offgas Charcoal Vault Refrigeration System, Brine Cooling Package Boiler Diagram	<Figure 9.4-24 (3)>
D-913-011	Offgas Charcoal Vault Refrigeration System, Brine Cooling Package Boiler Diagram	<Figure 9.4-24 (4)>
D-913-012	Offgas Charcoal Vault Refrigeration System - Brine Cooling Package Boiler Diagram	<Figure 9.4-24 (5)>
D-912-633	Smoke Venting System, Miscellaneous Electrical Areas	<Figure 9.4-25>
D-912-629	Turbine Lube Oil Diesel Drains, Fire Pump, Service Water Pumphouse, Water Treatment Building, Circulating Water Pumphouse Ventilation System	<Figure 9.4-27>
D-912-634	Radwaste Control Room HVAC System	<Figure 9.4-28>
D-913-018	Control Room and Computer Rooms Humidification System	<Figure 9.4-29>
D-914-001	Fire Service Water Yard Area	<Figure 9.5-1>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-914-002	Fire Service Water (Unit 1, Turbine Area)	<Figure 9.5-2>
D-914-003	Fire Service Water (Nuclear Plant)	<Figure 9.5-3>
D-914-004	Fire Protection Water Miscellaneous Services	<Figure 9.5-4>
D-914-005	Carbon Dioxide System	<Figure 9.5-5>
D-302-352	Standby Diesel Generator, Fuel Oil System	<Figure 9.5-8>
D-302-354	Standby Diesel Generator, Jacket Water	<Figure 9.5-9>
D-302-351	Piping System Diagram, R44, Standby Diesel Generator Starting Air	<Figure 9.5-10>
D-302-353	Standby Diesel Generator, Lube Oil	<Figure 9.5-11>
D-302-355	Standby Diesel Generator Exhaust, Intake and Crankcase	<Figure 9.5-12>
D-302-356	HPCS Diesel Generator Fuel Oil System	<Figure 9.5-15>
D-302-360	Division 3 Diesel Jacket Water Cooling System Diagram	<Figure 9.5-16>
D-302-051	Auxiliary Steam	<Figure 9.5-17>
D-302-052	Auxiliary Steam	<Figure 9.5-18>
D-302-053	Auxiliary Steam	<Figure 9.5-19>
D-302-054	Auxiliary Steam	<Figure 9.5-20>
D-304-352	Diesel Generator Fuel Oil Piping - Yard Area	<Figure 9.5-21 (1)>
D-304-353	Diesel Generator Fuel Oil Piping - Yard Area	<Figure 9.5-21 (2)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-358	Division 3 Diesel Starting Air/Air Dryer Diagram	<Figure 9.5-24>
D-302-359	Division 3 Diesel Lube Oil System Diagram	<Figure 9.5-25>
D-302-011 (769E305CA-2)	Main Steam System, Unit 1	<Figure 10.1-1 (1)>
D-302-012	Reheat Steam System, Unit 1	<Figure 10.1-1 (2)>
D-302-014	Reheater Heating Steam System	<Figure 10.1-1 (3)>
D-302-041	Extraction Steam	<Figure 10.1-2>
D-302-081	Feedwater	<Figure 10.1-3 (1)>
D-302-082 (769E305CA-4)	Feedwater	<Figure 10.1-3 (2)>
D-302-101	Condensate System	<Figure 10.1-4 (1)>
D-302-103	Condensate System	<Figure 10.1-4 (2)>
D-302-104	Condensate Filtration System	<Figure 10.1-5 (1)>
D-302-105	Condensate Filtration System	<Figure 10.1-5 (2)>
D-302-106	Condensate Filtration System	<Figure 10.1-5 (3)>
D-302-107	Condensate Demineralizer System	<Figure 10.1-6 (1)>
D-302-108	Condensate Demineralizer System	<Figure 10.1-6 (2)>
D-302-109	Condensate Demineralizer System	<Figure 10.1-6 (3)>
D-302-110	Condensate Demineralizer System	<Figure 10.1-6 (4)>
D-302-201	Circulating Water System	<Figure 10.1-7>
D-302-111	High Pressure Heater Drains and Vents	<Figure 10.1-8 (1)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-112	High Pressure Heater Drains and Vents	<Figure 10.1-8 (2)>
D-302-114	High Pressure Heater Drains and Vents	<Figure 10.1-8 (3)>
D-302-115	High Pressure Heater Drains and Vents	<Figure 10.1-8 (4)>
D-302-113	Low Pressure Heater Drains and Vents	<Figure 10.1-9>
D-302-141	Steam Seal System	<Figure 10.1-10>
D-302-131	Condenser Air Removal	<Figure 10.1-11>
D-302-301	Hydrogen Supply System	<Figure 10.2-4>
D-302-302	Generator H_2 and CO_2 Gas Control System	<Figure 10.2-5>
D-914-005	Fire Service Carbon Dioxide	<Figure 10.2-6>
D-302-021	Steam Bypass and Pressure Regulation System	<Figure 10.4-1>
D-302-739	Input Streams for the Liquid Radwaste System	<Figure 11.2-1 (1)>
D-302-740	Input Streams for the Liquid Radwaste System	<Figure 11.2-1 (2)>
D-302-741	Input Streams for the Liquid Radwaste System	<Figure 11.2-1 (3)>
D-302-731	Input Streams for the Liquid Radwaste System	<Figure 11.2-1 (4)>
D-302-751 (796E375-1)	Offgas System	<Figure 11.3-2 (1)>
D-302-752 (796E375-2)	Offgas System	<Figure 11.3-2 (2)>

TABLE 1.7-2 (Continued)

Drawing No. (GE No.)	Title	USAR Reference
D-302-753 (796E375-3)	Offgas System	<Figure 11.3-2 (3)>
D-302-754 (796E375-4)	Offgas System	<Figure 11.3-2 (4)>

1.8 NRC REGULATORY GUIDE ASSESSMENT

In 1970, the NRC (AEC) began to issue regulatory guides (Safety Guides) which describe in detail the methods acceptable to the NRC Staff for implementing specific parts of the Commission's regulations. The regulatory guides in some cases, delineate techniques used by the Staff in evaluating specific problems or postulated accidents and provide guidance to applicants concerning certain information needed by the Staff in its review of applications for permits and licenses.

<Table 1.8-1> lists each Division 1 and Division 8 Regulatory Guide addressed on the Perry Project. The appropriate revision for the Perry Project has been determined by referencing the NRC's Regulatory Requirements Review Committee (RRRC) categorization nomenclature for each of the regulatory guides. The RRRC Categories referenced in <Table 1.8-1> are defined as follows:

a. Category 1 - Clearly forward fit only.

b. Category 2 - Further Staff consideration of the need for backfitting appears to be required for certain identified items of the regulatory position. These individual issues are such that existing plants need to be evaluated to determine (a) their status with regard to these safety issues and (b) the need for backfitting.

c. Category 3 - Clearly backfit.

d. Category 4 - Regulatory guides not categorized by the RRRC.

<Table 1.8-1> provides a listing of PNPP's conformance to the recommendations of each of the non-QA related regulatory guides <Regulatory Guide 1.26> and <Regulatory Guide 1.29> for design, testing, maintenance, and operation of the Perry Nuclear Power Plant.

Conform, as used in <Table 1.8-1>, means that PNPP has implemented the regulatory guides, to the extent described in the table and in the referenced USAR sections. The level of commitment to each Regulatory Guide has been established jointly with the NRC during the acceptance review and safety review of the FSAR sections describing PNPP's implementation of the regulatory guides. Therefore, in order to obtain the specific degree of conformance to each regulatory guide, it is necessary to review <Table 1.8-1> along with the referenced sections of the USAR. The specific acceptance of this implementation by the NRC is reflected in the appropriate sections of the Safety Evaluation Report (SER). <NUREG-0887>

TABLE 1.8-1

CONFORMANCE TO NRC REGULATORY GUIDES

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.1> - (Revision 0 - 11/70; RRRC Category 1)		
Net positive suction head for emergency core cooling and containment heat removal system pumps	PNPP conforms to this guide.	<Section 5.4.7>, <Section 6.3.2>
<Regulatory Guide 1.2> - (Revision 0 - 11/70; RRRC Category 1)		
Thermal shock to reactor pressure vessels	Withdrawn by the NRC June 1991. Superseded by <10 CFR 50.61>, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."	
<Regulatory Guide 1.3> - (Revision 2 - 6/74; RRRC Category 1)		
Assumptions used for evaluating the potential radiological consequences of a loss-of-coolant accident for boiling water reactors	The original licensing basis LOCA radiological analyses, which were primarily based on <Regulatory Guide 1.3> and SRP 15.6.5, are now used only for post-LOCA equipment qualification, vital area access, PASS access, control room dose due to radiation shine, and containment purge isolation analyses. The current LOCA dose calculations are based on the alternate source terms and assumptions presented in <Regulatory Guide 1.183>, with modifications as described in the referenced USAR sections.	<Section 2.3.4>, <Section 2.3.5>, <Section 3.11.5.2.2> <Section 6.2.4.2.3> <Section 6.5.1>, <Section 9.4.2>, <Section 12.6.1>, <Section 15.0.3>, <Section 15.6.5>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.4> - (Revision 2 - 6/74)		
Assumptions used for evaluating the potential radiological consequences of a loss-of-coolant accident for pressurized water reactors	Not applicable to the PNPP design.	-
<Regulatory Guide 1.5> - (Revision 0 - 3/71; RRRC Category 1)		
Assumptions used for evaluating the potential radiological consequences of a steam line break accident for boiling water reactors	Not applicable. Replaced by <Regulatory Guide 1.183>.	<Section 2.3>, <Section 15.6.4>
<Regulatory Guide 1.6> - (Revision 0 - 3/71; RRRC Category 1)		
Independence between redundant standby (onsite) power sources and between their distribution systems	The independence among standby power sources and among their distribution systems is in accordance with this guide. The HPCS system conformance is discussed in <Section 8.3.1>.	<Section 7.1.2>, <Section 8.1>, <Section 8.3.1>
<Regulatory Guide 1.7> - (Revision 2 - 11/78; RRRC Category 1)		
Control of combustible gas concentrations in containment following a loss-of-coolant accident	PNPP conforms to this guide.	<Section 6.1.1>, <Section 6.2.5>, <Section 7.3.1>, <Section 7.3.2>, Tech. Specs.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.8> - (Revision 1 - R -5/77; RRRC Category 1)		
Personnel selection and training	PNPP commits to the regulatory position of this guide with the following clarification: <Regulatory Guide 1.8> states "The RPM should have a bachelor's degree or the equivalent in a science or engineering subject including some formal training in radiation protection and at least 5 years of professional experience in applied radiation protection." It is PNPP's position that equivalent as used in this regulatory guide for the bachelor's degree means (a) four years of post secondary schooling in science or engineering, or (b) four years of applied experience at a nuclear facility in the area for which qualification is sought, or (c) four years of operational or technical experience or training in nuclear power, or (d) any combination of the above totaling four years. The years of experience used to meet the education requirements as allowed by this exception shall not be used to also meet the experience requirements. PNPP commits to the requirements of ANSI N18.1-1971, with the exception of licensed operators. Licensed operators will comply with the requirements of <10 CFR 55>.	<Section 12.5>, <Section 13.1.1>, <Section 13.1.3>, Tech. Specs.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.9> - (Revision 0 - 3/71; RRRC Category 1)		
Selection, design and qualification of diesel-generator units used as onsite electric power systems at nuclear power plants	The standby diesel generators conform to this guide. The HPCS diesel-generator will conform to <Regulatory Guide 1.9> except that the starting transient for the single large motor load may cause the voltage or the frequency variations to exceed the maximum suggested but without impairment of the system function. Also recent test results, reported in Amendment 3 to NEDO-10905 (August 1979) showed that the voltage and frequency recovery requirements of <Regulatory Guide 1.9> were fully met.	<Section 3.11.2>, <Section 8.1>, <Section 8.3.1>, Tech. Specs.
<Regulatory Guide 1.10> - (Revision 1 - 1/73; RRRC Category 1)		
Mechanical (cadweld) splices in reinforcing bars of Seismic Category I concrete structures	PNPP design conforms to this guide with the exception that mechanical testing is based on ASME Section III, Division 2, Paragraph CB/CC 4333.	<Section 3.8.1>
<Regulatory Guide 1.11> - (Revision 0 - 2/72; RRRC Category 1)		
Instrument lines penetrating primary reactor containment	PNPP design conforms to this guide with the exception of the failure of isolation valves 1M51F0250A/B. New failure mode will be in the closed position.	<Section 6.2.4>, <Section 7.1.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.12> - (Revision 2 - 3/97; RRRC Category 4)		
Instrumentation for earthquakes	PNPP design conforms to this guide.	<Section 3.7.4>
<Regulatory Guide 1.13> - (Revision 1 - 12/75; RRRC Category 4)		
Spent fuel storage facility design basis	PNPP design conforms to this guide with the exception of paragraph C.4. The inventory of radioactive materials available for leakage are based on the assumptions given in <Regulatory Guide 1.183>.	<Section 9.1>, <Section 9.4.2>
<Regulatory Guide 1.14> - (Revision 1 - 8/75)		
Reactor coolant pump flywheel integrity	Not applicable to PNPP design.	-
<Regulatory Guide 1.15> - (Revision 1 - 12/72; RRRC Category 1)		
Testing of reinforcing bars for Seismic Category I concrete structures	PNPP design conforms to this guide.	<Section 3.8.1>, <Section 3.8.3>, <Section 3.8.4>, <Section 3.8.5>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.16> - (Revision 4 - 9/75; RRRC Category 1)		
Reporting of operating information - Appendix A Technical Specifications	PNPP conforms to this guide with the following clarification: Those sections of <Regulatory Guide 1.16> that are still applicable to reports required by the PNPP Technical Specifications may be used as guidance in preparing the respective Technical Specification reports.	Tech. Specs.
<Regulatory Guide 1.17> - (Revision 1 - 6/73; RRRC Category 1)		
Protection of nuclear power plants against industrial sabotage	PNPP conforms to this guide.	<Section 13.6>, Security Plan
<Regulatory Guide 1.18> - (Revision 1 - 12/72; RRRC Category 1)		
Structural acceptance test for concrete primary reactor containments	Not applicable to the PNPP design.	-
<Regulatory Guide 1.19> - (Revision 1 - 8/72; RRRC Category 1)		
Nondestructive examination of primary containment liner welds	PNPP conforms to this guide.	-
<Regulatory Guide 1.20> - (Revision 2 - 5/76; RRRC Category 1)		
Comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing	PNPP conforms to this guide.	<Section 1.5.1> <Section 3.9.2> <Section 14.2.12> <Section 15E.8> <Section 15F.7> <Section 15F.8>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.21> - (Revision 1 - 6/74; RRRC Category 1)		
Measuring, evaluating and reporting radioactivity in solid wastes and release of radioactivity in liquid and gaseous effluents from light-water cooled nuclear power plants	PNPP conforms to this guide with the following exceptions: 1. Meteorological data will be compiled in an annual report and will be available to the NRC upon request. 2. Liquid effluent sampling and analysis will be performed in accordance with the ODCM. All radioactive releases from liquid radwaste will be monitored by the Radwaste Discharge Radiation Monitor-ESW Discharge. The monitor alarm setpoint will eliminate the need to periodically sample the effluent during discharge. Prior to release, LRW tanks to be discharged will be mixed and samples drawn and analyzed. Based on these analyses, the radiation monitor alarm will be set to detect fluctuations in radwaste activity during release. This radiation monitor provides a control function, (i.e., if the alarm setpoint is exceeded the release will be terminated). Therefore, periodic sampling will not be necessary.	<Section 7.6.2>, <Section 11.2>, <Section 11.5>, Tech. Specs.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.21> (Continued)	3. Gaseous effluent sampling and analysis will be performed in accordance with the ODCM.	
	4. Average energy $(\overline{E})$ requirements will not be adhered to for gaseous effluent reporting since $\overline{E}$ is not used by PNPP to calculate gaseous release (rate) and dose (rate).	
	5. Periodic checks of composite samples to determine loss of radioactive material due to deposition or volatilization will not be performed since the addition of HNO3 to each sample upon collection eliminates the deposition/volatilization problem.	
	6. Periodic inservice calibrations of radiological effluent monitoring systems need not be performed since "real time" efficiencies are determined by direct correlation of measured total activity with the net monitor response. Effluent monitor set points and release rates are calculated using the efficiencies determined by the radiological monitoring systems' response to the radionuclide mix present. Effluent monitoring system calibration and testing will be performed in accordance with the ODCM.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.21> (Continued)	7. Total radionuclide release rate data will be reported in the Radioactive Effluent Release Report in the format of Table 1A-1C of the regulatory guide. This data will not be broken down by release point Table 1B and 1C because all release points are ground level and the ODCM does not contain release rate (mCi/cc) limits.	
	8. Condensation from the Turbine Building Supply Plenums that is directed to storm drains will be sampled and analyzed in accordance with the ODCM. A default value of 14,400 gallons per day, which was estimated as the value to be reached during periods of high relative humidity, will be used to calculate the dose assessment from the liquid effluent release of tritium from this point.	<Section 11.2>
	9. Gross beta radioactivity measurements, as discussed in <Regulatory Guide 1.21>, (Appendix A.3.a(1)) are not made to estimate the quantity of radioactive material released. The quantity of radioactive material released will be determined by measuring the principal gamma emitters with gamma spectroscopy equipment meeting the LLD requirements specified in <Table 11.5.7>.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.21> (Continued)	10. Alpha analysis for gaseous effluents is performed on each composite filter and not on a composite of all filters collected as discussed in <Regulatory Guide 1.21>, (Appendix A.3.a(3)).	
	11. The Effluent and Waste Disposal Report, as described in <Regulatory Guide 1.21>, (Appendix B), is prepared on an annual rather than semi-annual frequency, and is submitted in a report titled "Annual Environmental and Effluent Release Report."	
	12. Effluent concentrations are used to comply with <10 CFR 20, Appendix B> (Table 2) and not MPC as described in <Regulatory Guide 1.21>, (Section C.4).	
	13. Tritium analysis for gaseous batch releases, as described in <Regulatory Guide 1.21>, (Appendix A.4.a) will be satisfied with the analysis performed on the plant vent for the applicable area being ventilated.	
	14. Radioactive releases from the ADHR system to the service water system will be monitored by the ADHR Heat Exchanger Service Water Outlet Radiation Monitor. If a high radiation level is detected, the ADHR system shall be manually isolated.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.22> - (Revision 0 - 2/72; RRRC Category 1)		
Periodic testing of protection system actuation functions	The protective systems and components important to safety are designed to allow periodic testing in accordance with this regulatory guide, with the exception that each bypass condition (breaker operation or fuse removal) is indicated to the reactor operator in the main control room via administrative controls.	<Table 7.1-3>, <Section 7.2.2>, <Section 7.3.2>, <Section 7.4.2>, <Section 7.6.2>, <Section 8.1>, <Section 8.3.1>
<Regulatory Guide 1.23> - (Revision 0 - 2/72; RRRC Category 1)		
Onsite meteorological programs	PNPP conforms to this guide except Section C.4 Instrument Accuracy. PNPP Meteorological Monitoring Instrumentation meets system accuracy as stated in <Table 2.3-31>, which is in accordance with <Regulatory Guide 1.97>, (Revision 3 – 5/83).	<Section 2.3.3>, <Section 2.3.4>
<Regulatory Guide 1.24> - (Revision 0 - 3/72; RRRC Category 1)		
Assumptions used for evaluating the potential radiological consequences of a pressurized water reactor gas storage tank failure	Not applicable to PNPP design.	-

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.25> - (Revision 0 - 3/72; RRRC Category 1)		
Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors	Not applicable to PNPP. See <Regulatory Guide 1.183>.	
<Regulatory Guide 1.26> - (Revision 3 - 2/76; RRRC Category 1)		
Quality group classifications and standards for water-, steam- and radioactive-waste-containing components of nuclear power plants	PNPP design complies with this guide with an exception as stated in Note 39 of <Table 3.2-1>.	<Section 3.2.1>, <Table 3.2-1>, <Section 5.4>, <Section 6.2.4>, <Section 6.5>, <Section 6.9>, <Section 9.3>, <Section 9.4>, <Section 9.5>, <Section 10.3.3>, <Section 10.4>
<Regulatory Guide 1.27> - (Revision 2 - 1/76; RRRC Category 2)		
Ultimate heat sink for nuclear power plants	PNPP conforms with this guide with the following clarification: Technical Specifications do not address the loss of capability of the ultimate heat sink since there is no credible single failure which would preclude the ultimate heat sink from meeting its design criteria.	<Section 2.4>, <Section 9.2.5>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.28> - (Revision 2 - 2/79)		
Quality assurance requirements (design and construction)	PNPP conforms with this guide.	
<Regulatory Guide 1.29> - (Revision 3 - 9/78; RRRC Category 1)		
Seismic design classification	PNPP design complies with this guide, with exceptions as stated in Notes 19 and 24 of <Table 3.2-1> and with the following clarifications: Position C.1.e – The design of the main steam system incorporates a third isolation valve between the outermost MSIV and the turbine stop valve in each main steam line. The piping downstream of this MOV is nonsafety class. Position C.3 and C.4 - Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points. Thus the seismic design is not terminated at a seismic restraint, but is extended to the first	<Section 3.2.1>, <Table 3.2-1>, <Section 3.7.3>, <Section 6.2.4>, <Section 6.5>, <Section 6.7>, <Table 7.1-3>, <Table 8.1-2>, <Section 8.3.1>, <Section 9.1>, <Section 9.3.5>, <Section 9.4>, <Section 9.5>, <Section 10.3.1>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.29> (Continued)	point in the system, that can be treated as an anchor to the plant structure or to a distance sufficient such that the effects of the piping beyond the safety class boundary are insignificant. Paragraph C.4 also requires that the pertinent quality assurance requirements of <10 CFR 50, Appendix B> be applied to the safety requirements of such items. Both these requirements are considered to be adequately met by the following practice: a. Design and design control for these items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews. <u>Position C-4</u> - Design for items that would otherwise be classified as non-seismic but whose failure could reduce the functioning of items important to safety to an unacceptable safety level is performed in accordance with Seismic Category I requirements. Design control is carried out in the same manner as that for items directly important to safety.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.29> (Continued)	For piping and support of piping beyond the class break the following applies: a. Procurement of piping, inline components and their supports is performed in accordance with the item's safety classification, i.e., nonsafety. b. Installation of piping and inline components is also performed as with other nonsafety items. c. Final installation of component supports is inspected as a formal part of the Quality Assurance Program Manual.	
<Regulatory Guide 1.30> - (Revision 0 - 8/72; RRRC Category 1)		
Quality assurance requirements for the installation, inspection and testing of instrumentation and electrical equipment	See <Chapter 17.2>	<Section 3.8.2>, <Section 7.1.2>, <Table 8.1-2>, <Section 17.2>
<Regulatory Guide 1.31> - (Revision 3 - 4/78; RRRC Category 1)		
Control of ferrite content in stainless steel weld metal	Conformance evaluation was based on an extensive test program which demonstrates that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the regulatory requirements. All austenitic	<Section 3.8>, <Section 4.5.1>, <Section 4.5.2>, <Section 5.2.3>, <Section 5.3.1>, <Section 6.1.1>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.31> (Continued)	stainless steel weld filler material for PNPP is supplied with a minimum of 5% ferrite material.	
<Regulatory Guide 1.32> - (Revision 2 - 2/77; RRRC Category 1)		
Criteria for safety-related electric power systems for nuclear power plants	The design of the PNPP Class 1E power system conforms to IEEE Standard 308-1974 as modified by the positions of <Regulatory Guide 1.32>, with the exception that the battery testing intervals are controlled by the NRC-approved Technical Specifications.	<Section 7.1.2>, <Section 8.1>, <Section 8.3>
<Regulatory Guide 1.33> - (Revision 2 - 2/78; RRRC Category 1)		
Quality assurance program requirements (operations)	See <Chapter 17.2>	<Section 12.5.3>, <Section 13.4>, <Section 13.5>, <Section 17.2>
<Regulatory Guide 1.34> - (Revision 0 - 12/72; RRRC Category 1)		
Control of electroslag weld properties	Electroslag welding was not used during fabrication of ASME Boiler and Pressure Vessel Code Section III, Components.	<Section 4.5.2>, <Section 5.2.3>, <Section 5.3.1>
<Regulatory Guide 1.35> - (Revision 3 - 4/79; RRRC Category 1)		
Inservice inspection of ungrouted tendons in prestressed concrete containment structures	Not applicable to the PNPP design.	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.36> - (Revision 0 - 2/73; RRRC Category 1)		
Nonmetallic thermal insulation for austenitic stainless steel	PNPP conforms to this guide.	<Section 4.5.2>, <Section 6.1.1>
<Regulatory Guide 1.37> - (Revision 0 - 3/73; RRRC Category 1)		
Quality assurance requirements for cleaning of fluid systems and associated components of water cooled nuclear plants	See <Chapter 17.2>	<Section 4.5.1>, <Section 4.5.2>, <Section 6.1.1>, <Section 10.3.6>, <Section 17.2>
<Regulatory Guide 1.38> - (Revision 2 - 5/77; RRRC Category 1)		
Quality assurance requirements for packaging, shipping, receiving, storage, and handling of items for water cooled nuclear power plants	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.39> - (Revision 2 - 9/77; RRRC Category 1)		
Housekeeping requirements for water cooled nuclear power plants	See <Chapter 17.2>	<Section 12.5.3>, <Section 17.2>
<Regulatory Guide 1.40> - (Revision 0 - 3/73; RRRC Category 1)		
Qualification tests of continuous-duty motors installed inside containment of water-cooled nuclear power plants	Inside containment Class 1E Motors are type tested in accordance with IEEE Standard 334-1971 as modified by the regulatory positions of <Regulatory Guide 1.40>.	<Section 3.11>, <Section 7.1.2>, <Section 8.1>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.41> - (Revision 0 - 3/73; RRRC Category 1)		
Preoperational testing of redundant onsite electric power systems to verify proper load group assignments	PNPP conforms to this guide with the following clarification: Suitable preoperational tests to detect lack of independence will be performed. These tests will assure that each redundant onsite power source and its load group can function without any dependence upon any other redundant load group or portion thereof. In relation to Position C1, PNPP will isolate at startup transformer source Breakers L1003 and L1004 (L2003 and L2004).	<Section 8.1>, <Section 14.2.12>
<Regulatory Guide 1.42>		
	<Regulatory Guide 1.42> was withdrawn on 3/22/76 (Federal Register Notice 41FR11891)	-
<Regulatory Guide 1.43> - (Revision 0 - 5/73)		
Control of stainless steel weld cladding of low-alloy steel components	Safety class component specifications required that all low alloy steel be produced to fine grain practice. The requirements of this regulatory guide are not applicable to the NSSS components at PNPP.	<Section 5.3.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.44> - (Revision 0 - 5/73; RRRC Category 1)		
Control of the use of sensitized stainless steel	PNPP conforms to this guide with the following exceptions: Position C.3 The stainless steel components in the NSSS scope of supply of this regulatory guide definition were either solution heat treated or the weld joint inside surface was protected with corrosion resistant cladding or other means to minimize material susceptibility to IGSCC. Therefore, corrosion testing, as required by this position, was not performed. Position C.6 Intergranular corrosion testing was not considered necessary to qualify welding procedures because the essential variables used in welding procedures were based on recommendations made by General Electric following extensive research. Furthermore, IGSCC counter-measures (GE-22A4298) have been applied to the extent practical. Steps were taken to minimize sensitization by control of welding procedures.	<Section 4.5.1>, <Section 4.5.2>, <Section 5.2.3>, <Section 5.3.1>, <Section 6.1.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.45> - (Revision 0 - 5/73; RRRC Category 1)		
Reactor coolant pressure boundary leakage detection systems	PNPP conforms to this guide with the exception that the airborne particulate and gaseous radioactivity monitors do not meet the sensitivity level of Position C.5. Also, there is no attempt to correlate radioactivity monitoring indication to leakage flow rates as described in Position C.7. Two other methods for detecting unidentified RCPB leakage do, however, meet Positions C.5 and C.7.	<Section 5.2.5>, <Section 7.6.2>, <Section 8.3.1>, <Section 12.3.4>, Tech. Specs.
<Regulatory Guide 1.46> - (Revision 0 - 5/73, Withdrawn-3/85; RRRC Category 1)		
Protection against pipe whip inside containment	PNPP design conforms to this guide.	<Section 3.6>, <Section 6.2.5>
<Regulatory Guide 1.47> - (Revision 0 - 5/73; RRRC Category 1)		
Bypassed and inoperable status indication for nuclear power plant systems	Bypass and inoperable status indication is provided in the plant control room in accordance with <Regulatory Guide 1.47>.	<Section 6.5>, <Section 7.1.2>, <Section 7.2.2>, <Section 7.3.2>, <Section 7.4.2>, <Section 7.6.2>, <Section 8.1>, <Section 8.3>, <Section 9.4>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.48> - (Revision 0 - 5/73, Withdrawn-3/85; RRRC Category 1)		
Design limits and load combinations for Seismic Category I fluid system components	PNPP conforms to this guide, with the exception that the NRC positions are more conservative for stress allowables used for ASME Class 2 vessels and piping in faulted conditions. Refer to <Table 3.9-16> for details concerning NSSS systems. Non-NSSS systems are covered in <Section 3.9>.	<Section 3.9.1>, <Section 3.9.3>, <Section 6.2.4>, <Section 9.2.1>, <Section 9.4.6>
<Regulatory Guide 1.49> - (Revision 1 - 12/73; RRRC Category 1)		
Power levels of nuclear power plants	PNPP design conforms to this guide.	<Appendix 15B>
<Regulatory Guide 1.50> - (Revision 0 - 5/73; RRRC Category 1)		
Control of preheat temperature for welding of low-alloy steel	PNPP conforms to this guide.	<Section 5.2.3>, <Section 5.3.1>, <Section 6.1.1>, <Section 10.3.6>
<Regulatory Guide 1.51>		
	<Regulatory Guide 1.51> was withdrawn on 7/21/75. (Federal Register Notice 40FR30510)	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.52> - (Revision 2 - 3/78; RRRC Category 2)		
Design, testing and maintenance criteria for postaccident engineered-safety-feature atmosphere cleanup system air filtration and absorption units of light-water-cooled nuclear power plants	PNPP design conforms to this guide as presented in <Table 6.5-1>, <Table 6.5-2>,and <Table 6.5-3>. PNPP testing conforms to Revision 4 of this guide as presented in <Table 6.5-1>, <Table 6.5-2>, and <Table 6.5-3>.	<Section 6.4>, <Section 6.5.1>, <Section 9.1>, <Section 9.4>, <Section 12.3>, <Section 15.7>, Tech. Specs.
<Regulatory Guide 1.53> - (Revision 0 - 6/73; RRRC Category 1)		
Application of single failure criterion to nuclear power plant protection systems	Single failure criteria is applied to protection systems in accordance with <Regulatory Guide 1.53>.	<Section 6.5.3>, <Table 7.1-3>, <Section 7.2.2>, <Section 7.3.2>, <Section 7.4.2>, <Section 7.6.2>, <Section 8.1>, <Section 9.4>
<Regulatory Guide 1.54> - (Revision 0 - 6/73; RRRC Category 1)		
Quality Assurance requirements for protective coatings applied to water-cooled nuclear power plants	PNPP commits to the regulatory position of this guide with the following clarifications: 1. This regulatory guide and its associated ANSI Standard implies that a significant amount of coating work is required at the plant site. Although this is correct for construction sites, the coating work at an operating site generally	<Section 6.1.1>, <Section 6.1.2>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.54> (Continued)	consists of repair and touchup work following maintenance and repair activities or the initial coating of components such as hangers, supports, and piping during facility modifications. Therefore, in lieu of the full requirements of this regulatory guide and ANSI N101.4, PNPP imposes the following requirements:	
	a. The quality assurance requirements of Section 3 of ANSI N101.4 applicable to the coating manufacturer shall be imposed on the coating manufacturer through the procurement process.	
	b. Coating application procedures shall be developed based on the manufacturer's recommendations for application of the selected coating systems.	
	c. Coating applicators shall be qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations.	
	d. Quality control personnel shall perform inspections to verify conformance of the coating appli-	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.54> (Continued)	cation procedures. Section 6 of ANSI N101.4 shall be used as guidelines in the establishment of the inspection program.	
	e. Quality control personnel shall be qualified to the requirements of <Regulatory Guide 1.58>, (Revision 1).	
	f. Documentation demonstrating conformance to the above requirements shall be maintained.	
	2. The requirements of Position A of this guide apply to surfaces within containment with the following exceptions:	
	a. Surfaces to be insulated.	
	b. Surfaces contained within a cabinet or enclosure.	
	c. Repair/touchup areas less than 30 square inches or surface areas such as: cut ends; bolt heads, nuts and miscellaneous fasteners; and damage resulting from spot, tack or arc welding.	
	d. Small items such as small motors, handwheels, electrical cabinets, control panels, loud speakers,	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.54> (Continued)	motor operators, etc., where special painting requirements would be impracticable.	
	e. Stainless steel or galvanized surfaces.	
	f. Banding used for insulated pipe.	
	PNPP commits to the requirements of ANSI N101.4-1972 for activities comparable in nature and extent to construction phase activities.	
<Regulatory Guide 1.55> - (Revision 0 - 6/73; RRRC Category 1)		
Concrete placement in Category I structures	PNPP commits to the regulatory position of this guide for activities that are comparable in nature to construction phase activities.	<Section 3.8>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.56> - (Revision 1 - 8/78; RRRC Category 3)		
Maintenance of water purity in boiling water reactors	PNPP conforms to this guide with the exception of the resin capacity recommendations of Positions C.3 and C.4a, c and d. Sufficient resin capacity is ensured as described in <Section 5.2.3.2.2.1 (b.3)>.	<Section 5.2.3>, <Section 10.4.6>
<Regulatory Guide 1.57> - (Revision 0 - 6/73; RRRC Category 1)		
Design limits and loading combinations for metal primary reactor containment system components	PNPP design conforms to this guide as described in <Section 3.8.2.5>.	<Section 3.8.2>, <Section 3.8.3>
<Regulatory Guide 1.58> - (Revision 1 - 9/80; RRRC Category 1)		
Qualification of nuclear power plant inspection, examination and testing personnel	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.59> - (Revision 2 - 8/77; RRRC Category 2)		
Design basis floods for nuclear power plants	PNPP design conforms to this guide.	<Section 2.4.3>
<Regulatory Guide 1.60> - (Revision 1 - 12/73; RRRC Category 1)		
Design response spectra for seismic design of nuclear power plants	PNPP design conforms to this guide.	<Section 2.5.2>, <Section 3.7.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.61> - (Revision 0 - 10/73; RRRC Category 1)		
Damping values for seismic design of nuclear power plants	PNPP design conforms to this guide.	<Section 3.7.1>, <Section 3.10.1>, <Section 5.2.1>
<Regulatory Guide 1.62> - (Revision 0 - 10/73; RRRC Category 1)		
Manual initiation of protective actions	PNPP conforms to this guide.	<Table 7.1-3>, <Section 7.2.2>, <Section 7.3.2>, <Section 7.4.2>, <Section 7.6.2>, <Section 8.3.1>
<Regulatory Guide 1.63> - (Revision 2 - 7/78; RRRC Category 2)		
Electric penetration assemblies in containment structures for light-water-cooled nuclear power plants	PNPP design conforms to IEEE Standard 317-1976, as modified by <Regulatory Guide 1.63>.	<Section 3.11.2>, <Section 8.1>, <Section 8.3.1>
<Regulatory Guide 1.64> - (Revision 2 - 6/76; RRRC Category 1)		
Quality assurance requirements for the design of nuclear power plants	See <Chapter 17.2>	<Section 17.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.65> - (Revision 0 - 10/73; RRRC Category 1)		
Materials and inspection for reactor vessel closure studs	The PNPP reactor vessel order date preceded implementation of <Regulatory Guide 1.65>. The reactor vessel closure stud bolting meets the intent of <Regulatory Guide 1.65> except that the maximum tensile strength of the stud material is 174 ksi instead of 170 ksi as recommended by Position C.1.b(1) of the guide. The PNPP reactor vessel order date preceded implementation of <Regulatory Guide 1.65>. Refer to <Section 5.3.1.7> for details. PNPP conforms to this guide with the Following exceptions: The maximum tensile strength of the reactor Vessel closure stud bolting material is 174 ksi instead of 170 ksi as recommended by Position C.1.b(1) (Refer to <Section 5.3.1.7> for details). The recommendations of Position C.4 need not be met, since surface examinations are no longer necessary as a result of NRC approval of ASME Section XI Code Case N-652, and editions and addenda of Section XI that no longer require surface examination of reactor vessel studs.	<Section 5.3.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.66>		
Nondestructive examination of tubular products	This regulatory guide was withdrawn September 1977.	<Section 5.2.3>
<Regulatory Guide 1.67> - (Revision 0 - 10/73; RRRC Category 1)		
Installation of overpressure protective devices	PNPP design conforms to this guide.	<Section 3.9.3>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.68> - (Revision 2 - 8/78; RRRC Category 1)		
Initial test programs for water cooled nuclear power plants	The initial test program consists of three phases including initial checkout and run-in, preoperational testing and startup testing. PNPP conforms to this guide with the following clarifications and exceptions: 1. Section C.9, Items a and b. PNPP takes exception to items a and b as being included in the report. PNPP lists the tests performed and references <Chapter 14> for a description of test methods and objectives. For those tests which do not meet acceptance criteria, the report includes a justification for acceptance as required by C.9c, Items d and e. 2. Appendix A.1, Paragraph 2. (Page 1.68-6) PNPP takes exception to performing system expansion, vibration and restraint tests on all structures, systems and components. PNPP tests those structures, systems and components identified in <Section 3.9.2>.	<Section 8.1>, <Section 8.3.1>, <Section 9.5>, <Section 10.4.7>, <Section 14.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.68> (Continued)	3. Appendix A, Section 1.c. PNPP takes exception to time response testing requirements. PNPP time response tests the reactor protection system channels including sensors as defined in the Technical Specifications and in <Chapter 14>. 4. Appendix 9, Section 1.g (1) and (2). PNPP takes exception to the requirement to demonstrate the load-carrying capability of system cables in accordance with design criteria. PNPP demonstrated that system components and cables adequately supply system loads, not demonstrate the cable design load carrying capability. PNPP also takes exception to the requirement to demonstrate that emergency loads can start with the maximum and minimum design voltage available. PNPP verified that proper voltages are available in order to establish transformer tap settings and to verify computer modeling of the electrical system.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.68> (Continued)	5. Appendix A, Section 1.h, Paragraph 2. PNPP takes exception to the requirement to verify functioning of protective devices such as leak tight covers or housings. Leak tight requirements for covers and housings are part of the equipment specifications. 6. Appendix A, Section 1.j (15). N/A - PNPP does not use an automatic dispatcher control system. 7. Appendix A, Section 2.f. (Page 1.68-14) The flow induced vibrational measurement test following fuel load is performed after initial criticality and before nuclear heatup to allow performance of the Full Core Shutdown Margin Test (initial critical) and CRD ganged rod test before the RPV head is installed. Other flow tests are performed during the power ascension test program at rated conditions.	

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.68> (Continued)	8. Appendix A, Section 5.l. PNPP takes exception to the requirement to test RHR steam condensing mode prior to exceeding 25% power and commits to performing it prior to exceeding 32% power. The RHR steam condensing mode was eliminated from the Perry Nuclear Power Plant after the startup test program was completed.	
<Regulatory Guide 1.68.1> - (Revision 1 - 1/77; RRRC Category 1)		
Preoperational and initial startup testing of feedwater and condensate systems for boiling water reactor power plants	PNPP conforms to this guide with the exception of commitments to Position C.1 - "Preoperational Testing," and Positions C.2.f and g - "Startup Testing – Vibration and Thermal Expansion Testing" since both the condensate and portions of feedwater systems are classified as nonsafety for testability purposes. The expansion and vibration testing for both preoperational and startup phases will be performed as described in <Section 3.9.2>.	<Section 14.0>
<Regulatory Guide 1.68.2> - (Revision 1 - 7/78; RRRC Category 1)		
Initial startup test program to demonstrate remote shutdown capability for water-cooled nuclear power plants	PNPP conforms to this guide.	<Section 14.0>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.69> - (Revision 0 - 12/73; RRRC Category 1)		
Concrete radiation shields for nuclear power plants	PNPP conforms to this guide.	<Section 3.8>, <Section 12.3.2>
<Regulatory Guide 1.70> - (Revision 3 - 11/78; RRRC Category 1)		
Standard format and content of safety analysis reports for nuclear power plants	<Regulatory Guide 1.70> was utilized in the preparation of the PNPP FSAR which was docketed by the NRC on January 30, 1981. The FSAR was subsequently reviewed and accepted by the NRC through the Safety Evaluation Report (SER) and its supplements prior to its conversion to the USAR format.	USAR
<Regulatory Guide 1.71> - (Revision 0 - 12/73; RRRC Category 1)		
Welder qualification for areas of limited accessibility	During construction PNPP conforms to this guide with the exception of Position C.1: The Project has guidelines to aid in identifying developed limited access conditions. Where a potential condition is identified, a Project Organization welding engineer evaluates the actual field condition and determines what steps will be taken to assure quality.	<Section 3.8.3>, <Section 4.5.2>, <Section 5.2.3>, <Section 5.3.1>, <Section 6.1.1>, <Section 10.3.6>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.71> (Continued)	For shielded metal arc welding, the limiting conditions are as follows: Where there is an obstruction on one side only, the necessary clearance will be 8 inches; where the obstruction is on two sides, the necessary clearance will be 10 inches; where the obstruction is on three sides, the necessary clearance will be 12 inches. For gas tungsten arc welding, the limiting conditions are as follows: Where there is an obstruction on one side only, the necessary clearance will be 4 inches; where the obstruction is on two sides, the necessary clearance will be 5 inches; where the obstruction is on three sides, the necessary clearance will be 6 inches. During operations PNPP conforms to this guide with the exception of Position C.1: Performance qualifications for personnel who weld under conditions of limited access, as defined in Regulatory Position C.1, are maintained in accordance with the applicable requirements of ASME Sections III and IX.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.71> (Continued)	However, specific qualification for limited access welds will not be required. To assure that the required integrity level for a specific limited access weld is achieved, welding conducted in areas of limited access must pass the required nondestructive examination. No waiver or relaxation of examination methods or acceptance criteria because of the limited access will be permitted.	
<Regulatory Guide 1.72> - (Revision 2 - 11/78; RRRC Category 1)		
Spray pond piping made from fiberglass-reinforced thermosetting resin	Not applicable to the PNPP design.	-
<Regulatory Guide 1.73> - (Revision 0 - 1/74; RRRC Category 1)		
Qualification tests of electric valve operators installed inside containment of nuclear power plants	Qualification of electric valve operators at PNPP is in accordance with IEEE Standard 382-1972, as modified by the positions of <Regulatory Guide 1.73>.	<Section 3.11.2>, <Table 7.1-3>, <Section 7.3.2>, <Section 7.4.2>, <Section 8.1>
<Regulatory Guide 1.74> - (Revision 0 - 2/74; RRRC Category 1)		
Quality assurance terms and definitions	See <Chapter 17.2>	<Section 17.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.75> - (Revision 2 - 9/78; RRRC Category 4)		
Physical independence of electrical systems	PNPP design is in accordance with IEEE Standard 384-1974, as modified by the positions of <Regulatory Guide 1.75>, with the alternative positions as discussed in <Table 8.1-2>.	<Section 7.1.2>, <Section 7.6.1>, <Section 8.1>, <Section 8.3.1>
<Regulatory Guide 1.76> - (Revision 0 - 4/74; RRRC Category 4)		
Design basis tornado for nuclear power plants	PNPP design conforms to this guide.	<Section 2.3.1>, <Table 2.3-5>, <Section 3.3.2>, <Section 3.5.1.4>
<Regulatory Guide 1.77> - (Revision 0 - 5/74; RRRC Category 1)		
Assumptions used for evaluating a control rod ejection accident for pressurized water reactors	Not applicable to the PNPP design.	-
<Regulatory Guide 1.78> - (Revision 0 - 6/74; RRRC Category 1)		
Assumptions for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release	PNPP design conforms to this guide.	<Section 2.2.3>, <Section 6.4>, <Section 9.5.8>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.79> - (Revision 1 - 9/75; RRRC Category 1)		
Preoperational testing of emergency core cooling systems for pressurized water reactors	Not applicable to the PNPP design.	-
<Regulatory Guide 1.80> - (Revision 0 - 6/74; RRRC Category 4)		
Preoperational testing of instrument air systems	PNPP conforms to this guide for the preoperational testing of the P57 safety-related instrument air system with the following clarification: 1. Item C7 test and check requirements were accomplished during the served equipment's system preoperational phase testing. PNPP conforms to this guide for the acceptance testing of the P52 nonsafety-related instrument air system with the following clarifications: 1. Item C7 test and check requirements were accomplished during the served equipment's system preoperational phase testing.	<Section 14.0>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.80> (Continued)	2. Items C8, C9 and C10 were accomplished by placing the valves tested in their normal operating position for simulation of the instrument air pipe break on selected system branches. Similarly, valves tested were placed in their normal operating position for simulation of instrument air pipe freezing/plugging on the selected system branches. The response of valves to a loss of air pressure when placed in a position other than failed as described in Item C8a was verified, where required, on an individual component basis.	
<Regulatory Guide 1.81> - (Revision 1 - 1/75; RRRC Category 1)		
Shared emergency and shutdown electric systems for multi-unit nuclear power plants	Not applicable to PNPP.	
<Regulatory Guide 1.82> - (Revision 2 - 5/96; RRRC Category 4)		
Water sources for long-term recirculation cooling following a Loss-of-Coolant accident	PNPP conforms to this guide.	<Section 6.2.2.2>, <Section 6.3.2.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.83> - (Revision 1 - 7/75; RRRC Category 4)		
Inservice inspection of pressurized water reactor steam generator tubes	Not applicable to the PNPP design.	-
<Regulatory Guide 1.84> - (Revisions 4 through 25 - 5/88; RRRC Category 1)		
Design and fabrication code case acceptability - ASME Section III Division 1	PNPP conforms to the guide revisions which correspond to the applicable ASME code of record. Additional code cases may be endorsed by the NRC and used by PNPP prior to revision of this regulatory guide. Future application of code cases will be evaluated for conformance with the current revision level of this regulatory guide prior to their use.	<Section 5.2.1>
<Regulatory Guide 1.85> - (Revisions 4 through 25 - 5/88; RRRC Category 1)		
Materials code case acceptability - ASME Section III Division I	PNPP conforms to the guide revisions which correspond to the applicable ASME code of record. Additional code cases may be endorsed by the NRC and used by PNPP prior to revision of this regulatory guide. Future application of code cases will be evaluated for conformance with the current revision level of this regulatory guide prior to their use.	<Section 5.2.1>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.86> - (Revision 0 - 6/74; RRRC Category 1)		
Termination of operating licenses for nuclear reactors	PNPP will comply with this guide.	-
<Regulatory Guide 1.87> - (Revision 1 - 6/75; RRRC Category 1)		
Guidance for construction of Class I components in elevated-temperature reactors (supplement to ASME Section III Code Classes 1592, 1593, 1594, 1595, and 1596)	Not applicable to PNPP design.	-
<Regulatory Guide 1.88> - (Revision 2 - 10/76; RRRC Category 1)		
Collection, storage and maintenance of nuclear power quality assurance records	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.89> - (Revision 1 - 6/84; RRRC Category 4)		
Qualification of Class 1E equipment for nuclear power plants	Class 1E equipment is qualified in accordance with IEEE Standard 323-1974, as endorsed by <Regulatory Guide 1.89> with the following specific exceptions: 1. NSSS Class IE equipment located in mild environmental zones was procured and qualified to IEEE Standard 323-1971. 2. Regulatory Position C2. The basis for radiological source terms used in discussed in <Section 3.11.5.2.2>.	<Section 3.10>, <Section 3.11>, <Section 7.1.2>, <Table 8.1-2>, <Section 8.3.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.89> - (Revision 1 - 6/84; RRRC Category 4) (Continued)		
	3. Additional specific guidance for type testing of cables, field splices and terminations is provided by IEEE Standard 383-1974, <Table 8.1-2>.	
	4. Specific criteria for assessing the acceptability of the environmental qualification program for safety related electrical equipment in a harsh environmental is provided by <NUREG-0588> Category I.	
	5. The acceptance criteria for the environmental qualification of safety related equipment located in a mild environment is the following:	
	a. The documentation required to demonstrate qualifications of safety related equipment in a mild environmental is the "Design/Purchase" specifications. The specifications contain a description of the functional requirements for its specific environmental zone during normal and abnormal environmental conditions. A well supported maintenance/surveillance program in conjunction with a good preventive maintenance program will ensure that equipment that meets the specifications is qualified for the designed life.	

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.89> - (Revision 1 - 6/84; RRRC Category 4) (Continued)		
	b. The maintenance/surveillance program data and records will be reviewed periodically (not more than 24 months) to ensure that the design qualified life has not suffered thermal and cyclic degradation resulting from the accumulated stresses triggered by the abnormal environmental conditions and the normal wear due to its service condition. Engineering judgment shall be used to modify the replacement program and/or replace the equipment deemed necessary.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.90> - (Revision 1 - 8/77; RRRC Category 1)		
Inservice inspection of prestressed concrete containment structures with grouted tendons	Not applicable to PNPP design.	-
<Regulatory Guide 1.91> - (Revision 1 - 2/78; RRRC Category 2)		
Evaluations of explosions postulated to occur on transportation routes near nuclear power plants	PNPP conforms to this guide.	<Section 2.2.3>
<Regulatory Guide 1.92> - (Revision 1 - 2/76; RRRC Category 1)		
Combining model responses and spatial components in seismic response analysis	PNPP design conforms to this guide.	<Section 3.7.2>, <Section 3.7.3>, <Section 3.8.2>, <Section 3.10.1>
<Regulatory Guide 1.93> - (Revision 0 - 12/74; RRRC Category 4)		
Availability of electric power sources	The requirements of <Regulatory Guide 1.93> for Limiting Conditions for Operations are addressed in Technical Specifications.	Tech. Specs., <Table 8.1-2>
<Regulatory Guide 1.94> - (Revision 1 - 4/76; RRRC Category 1)		
Quality assurance requirements for installation, inspection and testing of structural concrete and structural steel during the construction phase of nuclear power plants	See <Chapter 17.2>	<Section 17.2>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.95> - (Revision 1 - 2/77; RRRC Category 1)		
Protection of nuclear power plant control room operators against an accidental chlorine release	PNPP design conforms to this guide with the following exceptions: 1. Automatic isolation of the control room as well as chlorine detectors are not necessary to protect against an offsite chlorine release <Section 2.2.3.1.2.1>. 2. Additionally, control room leakage was determined by using the tracer gas method per ASTM E741-83 during the preoperational test program.	<Section 2.2.3>, <Section 7.3.2>, <Section 14.0>
<Regulatory Guide 1.96> - (Revision 1 - 6/76; RRRC Category 1)		
Design of main steam isolation valve leakage control systems for boiling water reactor nuclear power plants	Not applicable to PNPP design. MSIV-LCS was eliminated/abandoned in place in refuel outage 7.	<Section 6.7>, <Section 7.3.2>
<Regulatory Guide 1.97> - (Revision 2 - 12/80; RRRC Category 3)		
Instrumentation for light-water-cooled nuclear power plants to access plant conditions during and following an accident	PNPP design conforms to this guide as stated in <Table 7.1-4>. For Meteorological Monitoring Instrumentation accuracies, refer to conformance statement of <Regulatory Guide 1.23>, (Revision 0 – 2/72).	<Table 3.2-1>, <Section 7.1.2>, <Section 12.3.4>, Tech. Specs.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.98> - (Revision 0 - 3/76; RRRC Category 1)		
Assumptions used for evaluating the potential radiological consequences of a radioactive offgas system failure in a boiling water reactor	PNPP conforms to this guide with the following exceptions: 1. Position C.2.a: The SJAE is conservatively assumed to pump for 30 minutes. 2. Position C.2.e: Condenser air in leakage is assumed to be 2 scfm. 3. Source term differences as noted in the reference section.	<Section 15.7.1>
<Regulatory Guide 1.99> - (Unit 1: Revision 2 - 5/88; RRRC Category 3)		
Effects of residual elements on predicted radiation damage to reactor vessel materials	PNPP design conforms to this guide.	<Section 4.3.2.8>, <Section 5.3.1>, <Section 5.3.2>, <Section 5.3.3> Tech. Specs.
<Regulatory Guide 1.100> - (Revision 1 - 8/77; RRRC Category 1)		
Seismic qualification of electric equipment for nuclear power plants	All Class 1E equipment is seismically qualified in accordance with IEEE 344-75,as modified by <Regulatory Guide 1.100>.	<Section 3.10>, <Section 8.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.101> - (Revision 3 - 8/92; RRRC Category 3)		
Emergency planning for nuclear power plants	PNPP conforms to this guide as described in the PNPP Emergency Plan.	PNPP Emergency Plan
<Regulatory Guide 1.102> - (Revision 1 - 9/76; RRRC Category 2)		
Flood protection for nuclear power plants	PNPP conforms to this guide.	<Section 2.4>, <Section 9.1.2>, <Section 9.5.8>
<Regulatory Guide 1.103> - (Revision 1 - 10/76; RRRC Category 1)		
Post-tensioned prestressing systems for concrete reactor vessels and containments	Not applicable to the PNPP design.	-
<Regulatory Guide 1.104>		
Overhead crane handling systems for nuclear power plants	<Regulatory Guide 1.104> was withdrawn on August 16, 1979.	-

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.105> - (Revision 1 - 11/76; RRRC Category 2)		
Instrument setpoints	PNPP conforms to this guide with the following clarifications. The trip setpoint (instrument setpoint) is contained in the Operational Requirements Manual. The allowable value (technical specification limit) or the analytical or design basis limit are contained in Technical Specifications. The setpoints and allowable values are appropriately established based on instrument accuracy, calibration capability and design drift (estimated) allowance data. The setpoints are within the instrument accuracy range. The established setpoints provide margin to satisfy both safety requirements and plant availability objectives. Securing devices per Regulatory Position C.5 are not provided on all the setpoint adjustment mechanisms. Safety-related equipment has been seismically qualified for its function per IEEE-344-1975. This qualification documentation demonstrates adequate design without the use of a securing device.	<Section 7.1.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.106> - (Revision 1 - 3/77; RRRC Category 1)		
Thermal overload protection for electric motors on motor operated valves	Thermal overload relays to protect motor operated valves are not included in the design of the Class 1E power system; therefore, the positions of this guide are not applicable to the PNPP design.	<Section 8.1>
<Regulatory Guide 1.107> - (Revision 1 - 2/77; RRRC Category 1)		
Qualifications for cement grouting for prestressing tendons in containment structures	Not applicable to the PNPP design.	-
<Regulatory Guide 1.108> - (Revision 1 - 8/77; RRRC Category 2)		
Periodic testing of diesel generator units as onsite electric power systems at nuclear power plants	The guidelines presented in <Regulatory Guide 1.108> are used in establishing preoperational and periodic test procedures for the standby (Division 1 and 2) and HPCS (Division 3) diesel generators. One exception is that "first-out" annunciation was not used. The basis for this is the use of individual trip alarms, which give the operator adequate information for correct actions. Additionally, periodic testing is performed in accordance with NRC-approved Technical Specification	<Section 8.1>, <Section 8.3.1>, Tech. Specs.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.108> (Continued)	requirements, which meet the overall intent of Regulatory Position C.2 "Testing." The term "operating error" as identified in Section C.(2).e.(2) is an error committed during an operating activity as defined in ANS 3.2/ANSI N18.7-1976.	
<Regulatory Guide 1.109> - (Revision 1 - 10/77; RRRC Category 1)		
Calculation of annual doses to man from routine releases of reactor effluents for the purpose of evaluating compliance with <10 CFR 50, Appendix I>	PNPP conforms to this guide, with the exception that the mid-point of plant operating life (t_b) is 20 years.	<Section 12.4.4>, Environmental Report - Chapter 5, Tech. Specs.
<Regulatory Guide 1.110> - (Revision 0 - 3/76; RRRC Category 1)		
Cost benefit analysis for radwaste systems for light-water-cooled nuclear power reactors	The positions of this guide are not applicable since the construction permit for PNPP was docketed on, or after, January 2, 1971, and prior to June 4, 1976, and the radwaste systems and equipment described in the USAR satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket RM-50-2.	<Section 11.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.111> - (Revision 1 - 7/77; RRRC Category 1)		
Methods of estimating atmospheric transport and dispersion of gaseous effluents in routine releases from light-water-cooled reactors	PNPP conforms to this guide.	<Section 2.3.4>, <Section 2.3.5>
<Regulatory Guide 1.112> - (Revision 0-R - 5/77; RRRC Category 1)		
Calculation of releases of radioactive materials in gaseous and liquid effluents from light-water-cooled power reactors	PNPP conforms to this guide.	<Section 11.2.3>, <Section 11.3.3>
<Regulatory Guide 1.113> - (Revision 1 - 4/77; RRRC Category 1)		
Estimating aquatic dispersion of effluents from accidental and routine reactor releases for the purpose of implementing Appendix I	PNPP conforms to this guide.	<Section 2.4.12>
<Regulatory Guide 1.114> - (Revision 1 - 11/76; RRRC Category 3)		
Guidance on being operator at the controls of a nuclear power plant	PNPP conforms to this guide with the clarification that the areas shown on <Figure 13.5-1> define the areas associated with the "Operator at the Controls" concept specified in this guide.	<Section 13.5.1>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.115> - (Revision 1 - 8/77; RRRC Category 2)		
Protection against low trajectory turbine missiles	PNPP conforms to this guide.	<Section 3.5.1>
<Regulatory Guide 1.116> - (Revision 0 - 5/77; RRRC Category 1)		
Quality assurance requirements for installation, inspection and testing of mechanical equipment and systems	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.117> - (Revision 1 - 4/78; RRRC Category 2)		
Tornado design classification	PNPP does not have a tornado design classification, however, all Seismic Category I structures housing safety class equipment and systems are protected from tornado effects, including wind pressure, pressure drop, and missiles as described in <Section 3.3>, <Section 3.5>, and <Section 3.8>. Important systems and components (those that have been identified as serving the functions listed in the Appendix to <Regulatory Guide 1.117>) which are not protected from tornado missile effects by the Seismic Category I structures are protected as described in the "Reference" sections.	<Section 3.3.2>, <Section 3.5>, <Section 3.8.1>, <Section 3.8.4>, <Section 9.1.2>, <Section 9.5.8>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.118> - (Revision 2 - 6/78; RRRC Category 1)		
Periodic testing of electric power and protection systems	PNPP conforms to this guide with the following clarification: "Lifted leads and jumpers, fuse removal and breaker operation may be utilized during the performance of tests, under the direction of approved procedures. These procedures shall require independent verification or functional testing prior to return to service." Response time testing will be performed as required by Technical Specifications.	<Section 7.1.2>, <Section 8.1>
<Regulatory Guide 1.119>		
	<Regulatory Guide 1.119> was withdrawn on 6-20-77.	-
<Regulatory Guide 1.120> - (Revision 1 - 11/77, Withdrawn – 8/2001; RRRC Category 1)		
Fire protection guidelines for nuclear power plants	The fire protection guidelines for PNPP are taken from BTP-APCSB 9.5-1 Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants docketed prior to July 1, 1976." A detailed evaluation of this BTP is provided in Section 5 of the PNPP Fire Protection Evaluation Report.	PNPP Fire Protection Evaluation Report, <Table 8.1-2>, <Section 17.2>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.121> - (Revision 0 - 8/76)		
Bases for plugging degraded PWR steam generator tubes	Not applicable to the PNPP design.	-
<Regulatory Guide 1.122> - (Revision 1 - 2/78; RRRC Category 1)		
Development of floor design response spectra for seismic design for floor-supported equipment or components	PNPP design conforms to this guide with the exception that prior to the initial issue of the guide (September 1976), the spectrum peak was broadened by ±10%.	<Section 3.7.2>, <Section 3.10.1>, <Appendix 3A>
<Regulatory Guide 1.123> - (Revision 1 - 7/77)		
Quality assurance requirements for control of procurement of items and services for nuclear power plants	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.124> - (Revision 1 - 1/78; RRRC Category 2)		
Service limits and loading combinations for Class 1 linear component supports	<Regulatory Guide 1.124> is not addressed in the PNPP USAR since the construction permit was docketed prior to January 10, 1978, as referenced in Section D of the Guide.	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.125> - (Revision 1 - 11/78; RRRC Category 1)		
Physical models for design and operation of hydraulic structures and systems for nuclear power plants	<Regulatory Guide 1.125> is not addressed in the PNPP USAR since the documentation of data and studies recommended by this guide are requested for review during the construction permit stage. Physical models used for design of the hydraulic structures are discussed in <Section 3.8>.	<Section 3.8>
<Regulatory Guide 1.126> - (Revision 1 - 4/78; RRRC Category 1)		
An acceptable model and related statistical methods for the analysis of fuel densification	General Electric's methods for treatment of fuel densification issues are addressed in the NRC-approved GESTAR II and its US supplement (latest approved revision).	<Section 4.2>
<Regulatory Guide 1.127> - (Revision 1 - 3/78; RRRC Category 3)		
Inspection of water control structures associated with nuclear power plants	PNPP conforms to this guide as it applies to the intake and discharge control structures.	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.128> - (Revision 1 - 10/78; RRRC Category 1)		
Installation design and installation of large lead storage batteries for nuclear power plants	Class 1E batteries are designed and installed in accordance with IEEE Standard 484-1975, as modified by <Regulatory Guide 1.128>, except that a hydrogen survey will not be performed. Calculations indicate that the maximum concentration in the battery area will be less than 0.003%.	<Section 8.1>
<Regulatory Guide 1.129> (Revision 2 2/2007; RRRC Category 1)		
Maintenance, Testing, and Replacement Of Vented Lead-Acid Storage Batteries For Nuclear Power Plants	PNPP conforms to <Regulatory Guide 1.129> with the following exceptions: 1. <Regulatory Guide 1.129> endorses adjusting temperature before conducting the discharge test. PNPP is taking exception and allowing the temperature to be corrected for before or after the discharge test to align with IEEE Std 450-2002. 2. <Regulatory Guide 1.129> Position 1, Subsection 2, “References,” is not applicable to the new “Battery Monitoring and Maintenance Program”	<Section 8.1>, Tech. Specs.

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.129> (Revision 2 2/2007; RRRC Category 1) (Continued)		
	3. In lieu of <Regulatory Guide 1.129>, Regulatory Position 2, Subsection 5.2, “Inspections,” the following shall be used: “Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery.”	
	4. In <Regulatory Guide 1.129>, Regulatory Position 3, Subsection 5.4.1, “State of Charge Indicator,” the following statements in paragraph (d) may be omitted: “When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage.”	
	5. In lieu of <Regulatory Guide 1.129>, Regulatory Position 7, Subsection 7.6, “Restoration,” the following may be used: “Following the test, record the float voltage of each cell of the string.”	

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.130> - (Revision 1 - 10/78; RRRC Category 2)		
Service limits and loading combinations for Class 1 plate-and-shell-type component supports	<Regulatory Guide 1.130> is not addressed in the PNPP USAR since the construction permit was issued prior to October 31, 1978, as referenced in Section D of the Guide.	-
<Regulatory Guide 1.131> - (Revision - - 8/77; RRRC Category 1)		
Qualification test of electric cables, field splices and connections for light-water-cooled nuclear power plants	Issued for comment.	-
<Regulatory Guide 1.132> - (Revision 1 - 3/79; RRRC Category 1)		
Site investigations for foundations of nuclear power plants	Most of the geological site investigations for PNPP were complete prior to original issuance of the guide in September 1977, however work was performed in conformance with the intent of the guide.	<Section 2.5.1>, <Section 2.5.4>, <Section 2.5.5>
<Regulatory Guide 1.133> - (Revision 1 - 5/81; RRRC Category 1)		
Loose part detection program for the primary system of light-water-cooled reactors	Not applicable to PNPP design. Loose Parts Monitoring System was eliminated/ abandoned in place.	<Section 4.4.6>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.134> - (Revision 2 - 4/87; RRRC Category 1)		
Medical evaluation of nuclear power plant personnel requiring operator licenses	PNPP conforms to this guide. ANSI/ ANS 3.4-1983 as endorsed by <Regulatory Guide 1.134>, Rev. 2, will be used to conduct the medical examinations.	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.135> - (Revision 0 - 9/77; RRRC Category 1)		
Normal water level and discharge at nuclear power plants	PNPP design conforms to this guide, however a conservative alternative approach to Position C-3 was used, in that the mean monthly water level of the past 100 year record was used to determine normal level.	<Section 2.4.8>, <Section 2.4.11>
<Regulatory Guide 1.136> - (Revision 1 - 10/78; RRRC Category 1)		
Material for concrete containments	Not applicable to the PNPP design.	-
<Regulatory Guide 1.137> - (Revision 1 - 10/79; RRRC Category 2)		
Fuel-oil systems for standby diesel generators	Although not required by the Implementation section of the guide, the diesel generator fuel oil storage and transfer system conforms to <Regulatory Guide 1.137> with the following exceptions: 1. <Section 9.5.4.2> describes the design characteristics of the fuel oil storage tank such that turbulence during filling is minimized. 2. The cathodic protection system, referenced in C.1.g, has no special provisions to prevent the ignition of combustible vapors of diesel	<Section 9.5.4>, Tech. Specs.

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.137> (Continued)	generator fuel oil. The No. 2-D fuel oil being used has a flashpoint of 125°F. The fuel oil is not pre-heated for use and is not expected to see a temperature greater than 100°F.	
	3. (6.1 of ANSI N195-1976) National Fire Protection Association code NFPA 37-5-3.5, 1973, requires that the capacity of unenclosed day tanks supplying engines which drive generators used for emergency purposes shall not exceed 660 gallons. The standby diesel generator fuel oil day tank capacity is sufficient to provide approximately 30 minutes of diesel operation.	
	4. (7.5 of ANSI N195-1976) Each under-ground storage tank fill line is capped at all times except during filling. The cap provides a sealed barrier to the environment and therefore a fill line isolation valve is not necessary. A strainer is provided in the fuel oil transfer pump suction line.	

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.137> (Continued)	5. (8.2.d. of ANSI N195-1976) High level alarms do not exist for the supply tanks. Abnormally high oil levels could occur only during the tank filling operation which is administratively controlled. A central oil unloading/tank fill station is provided with a roadside pulloff for the tank truck. The area surrounding the pulloff is suitably drained so that spills or overflows are drained to an oil interceptor tank.	
	6. Regulatory Position C.1.e. The fuel oil system is tested in accordance with Section XI, Division 1, edition and addenda as applicable to the current Inservice Inspection interval.	
	7. Regulatory Position C.2.a, b, and c. PNPP conforms to more current ASTM Standards and operability requirements: See the Bases for Technical Specification 3.8.3, “Diesel Fuel Oil, Lube Oil, and Starting Air” for the current testing standards, and see Technical Specifications 3.8.3 and 5.5.9 “Diesel Fuel Oil Testing Program” for operability requirements.	

TABLE 1.8-1 (Continued)

(DELETED)

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.138> - (Revision - - 4/78)		
Laboratory investigations of soils for engineering analysis and design of nuclear power plants	The PNPP Construction Permits were issued prior to December 1, 1978. Therefore, <Regulatory Guide 1.138> does not apply.	-
<Regulatory Guide 1.139> - (Revision - - 5/78)		
Guidance for residual heat removal	Issued for comment.	-
<Regulatory Guide 1.140> - (Revision 0 - 3/78; RRRC Category 1)		
Design, testing and maintenance criteria for normal ventilation exhaust system air filtration and absorption units of light-water-cooled nuclear power plants	PNPP design and testing conforms to this guide as presented in <Table 12.3-3>.	<Section 9.4>, <Section 12.3.3>
<Regulatory Guide 1.141> - (Revision 0 - 4/78)		
Containment isolation provisions for fluid systems	Issued for comment.	-
<Regulatory Guide 1.142> - (Revision 0 - 4/78)		
Safety-related concrete structures for nuclear power plants (other than reactor vessels and containments)	Issued for comment.	<Section 3.8.1>, <Section 3.8.3>, <Section 3.8.4>, <Section 3.8.5>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.143> - (Revision 1 - 10/79; RRRC Category 1)		
Design guidance for radioactive waste management systems, structures and components installed in light-water-cooled nuclear power plants.	PNPP design is in conformance with this guide, with the following exceptions and clarifications: 1. Materials in the liquid and solid radwaste systems conform to ASTM Standards, and are constructed to high industry standards. They meet the Quality Group D Criteria as identified in <Regulatory Guide 1.26> and also the requirements of Table 1 of <Regulatory Guide 1.143>. The liquid and solid radwaste systems are housed in a Seismic Category I structure. Radionuclide concentrations from the liquid radwaste system are prevented from exceeding the limits of <10 CFR 20> at the nearest potable water supply. Therefore, the components in the liquid and solid radwaste systems are classified nonsafety class (NSC) and the quality assurance requirements of <Regulatory Guide 1.143> are not applicable.	<Table 3.2-1>, <Section 11.2>, <Section 11.3>, <Section 11.4>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.143> (Continued)	2. Materials in the gaseous radwaste system are supplied to ASTM Standards. Major equipment used to treat the N64 process stream meets the material requirements of <Regulatory Guide 1.143>. Equipment not used to treat the process stream may have malleable, wrought, or cast iron materials.	
	3. A mobile radwaste solidification system is used at PNPP under contract with an approved vendor.	
	4. Hydrostatic testing was performed during construction on all process piping in the liquid and solid radwaste systems in accordance with Regulatory Position C4.4 of <Regulatory Guide 1.143>. At a minimum, piping design changes during operations will be leak tested in accordance with ANSI B31.1 rules for Initial Service Leak Testing to ensure system integrity. All piping changes during operations will be designed and constructed to the Quality Group D Criteria as identified in Table 1 of <Regulatory Guide 1.26> and also the requirements of Table 1 of <Regulatory Guide 1.143>.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.143> (Continued)	5. With respect to the Offgas System design, after the initial pressure test and helium leak test (i.e., during the Operations phase) helium leak tests shall be performed on modifications/repairs whenever practicable. All welds performed during such modifications/repairs shall be subject to non-destructive examination (e.g., radiography or liquid penetrant exam).	
<Regulatory Guide 1.144> - (Revision 1 - 9/80)		
Auditing of quality assurance programs for nuclear power plants	See <Chapter 17.2>	<Section 17.2>
<Regulatory Guide 1.145> - (For Comment - 8/79)		
Atmospheric dispersion models for potential accident consequence assessments at nuclear power plants	PNPP conforms to this guide.	<Section 2.3.4>, <Section 15.6.5>
<Regulatory Guide 1.146> - (Revision 0 - 8/80)		
Qualification of quality assurance program audit personnel for nuclear power plants	See <Chapter 17.2>	<Section 17.2>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.147>		
Inservice Inspection Code Case Acceptability, ASME XI, Division 1	PNPP conforms to the guide revisions which correspond to the applicable code of record. Additional code cases may be endorsed by the NRC and used by PNPP prior to revision of this regulatory guide. Future application of code cases will be evaluated against the current revision of this regulatory guide prior to their use. Application of specific code cases will be identified in the PNPP ISE Program or appropriate Installation Standard Specifications.	
<Regulatory Guide 1.149> - (Revision 3 - 10/01)		
Nuclear power plant simulation facilities for use in operator training and license examinations.	PNPP conforms to this guide.	-
<Regulatory Guide 1.150> - (Revision 1 - 2/83)		
Ultrasonic testing of reactor vessel welds during preservice and inservice examinations	<Regulatory Guide 1.150> was withdrawn on 2/11/2008 (Federal Register Notice 73FR7766). As of 9/22/2002, 10 CFR 50.55a(g) superseded this Regulatory Guide. PNPP conforms to 10 CFR 50.55a(g), as stated in USAR Section 5.2.4.	-
<Regulatory Guide 1.155> - (Revision 0 - 8/88)		
Station Blackout	PNPP conforms to this guide.	<Section 15.8.2>, <Appendix 15H>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.163> - (Revision 0 - 9/95)		
Performance-Based Containment Leak-Test Program	PNPP no longer complies with this guide following issuance of License Amendment 185. The Primary Containment Leakage Rate Testing Program complies with the guidelines contained in NEI Topical Report 94-01, Revision 3-A, with conditions and limitations in NEI 94-01, Revision 2-A, as modified by the following exceptions: BN-TOP-1 methodology may be used for Type A tests. The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program.	

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.183> - (Revision 0 - 7/00)		
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	PNPP conforms to this guide for evaluating design basis accidents with the following exceptions: 1. Appendix B, Section 2; water depth above reactor flange inside containment is less than 23 feet. 2. Table 6, and Appendix B Sections 4.1 and 5.3; the radioactivity that escapes from the pool is assumed to be released to the environment instantaneously. 3. Appendix I: In lieu of conformance with Appendix I, conformance with <Regulatory Guide 1.89> is maintained, with exceptions as noted in its Degree of Conformance column and its listed USAR Sections/References. 4. The original licensing basis accident source term is retained for post-LOCA equipment qualification, vital area access, post-accident sampling system (PASS) access, control room dose due to radiation shine, and containment purge isolation analyses.	<Section 3.11.5.2.2> <Section 6.2.4.2.3>, <Section 6.5>, <Section 9.1.2>, <Section 9.4.2>, <Section 12.6.1>, <Section 15.4.9>, <Section 15.6.4>, <Section 15.6.5>, <Section 15.7.4>, <Section 15.7.6>, Tech. Specs.

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.183> - (Revision 0 - 7/00) (Continued)		
	5. A <Regulatory Guide 1.183> - based alternative accident source term is used in radiological dose consequence analyses for LOCA, FHA, CRDA, and MSLB; future revisions to other design basis analyses will also utilize a <Regulatory Guide 1.183> - based alternative accident source term.	
<Regulatory Guide 1.190> – (Revision 0 – 4/01)		
Calculational and dosimetry methods for determining pressure vessel neutron fluence	Neutron fluence methodologies used by PNPP will conform to this guide.	4.1, 4.3, 5.3
<Regulatory Guide 1.192>)		
Operation and Maintenance Code Case Acceptability, ASME OM Code	PNPP conforms to the regulatory guide revisions which correspond to the applicable code of record. Application of Code Cases will be evaluated against the current revision of this regulatory guide prior to their use.	
<Regulatory Guide 1.196> - (Revision 0 - 04/03)		
Control Room Habitability at Light-Water Nuclear Power Reactors	PNPP conforms to Section C.2.7.3 of this guide, which describes compensatory measures that may be utilized as the mitigating actions described in the Technical Specifications.	

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 1.197> - (Revision 0 - 04/03)		
Demonstrating Control Room Envelope Integrity at Nuclear Power Plants	PNPP conforms to the in-leakage testing methods and frequencies in Positions C.1 and C.2 of this guide, and to the frequency for performing periodic assessments that is outlined in Positions C.1 and C.2 of this guide.	
<Regulatory Guide 8.1> - (Revision 0 - 2/73)		
Radiation symbol	PNPP conforms to this guide.	-
<Regulatory Guide 8.2> - (Revision 0 - 2/73)		
Guide for administrative practices in radiation monitoring	PNPP conforms to this guide.	<Section 12.3.4>, <Section 12.5>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 8.3> - (Revision 0 - 2/73)		
Film badge performance criteria	TLD's are the primary monitoring device at PNPP. In the event film badges are utilized for personnel monitoring, PNPP will conform to this guide.	<Section 12.5>
<Regulatory Guide 8.4> - (Revision 1 - 6/2011)		
Personnel Monitoring Device - Direct-Reading Pocket Dosimeters	PNPP conforms to this guide.	<Section 12.5>
<Regulatory Guide 8.5> - (Revision 0 - 2/73)		
Immediate evacuation signal	PNPP conforms with this guide.	<Section 12.3.4>, Emergency Plan
<Regulatory Guide 8.6> - (Revision 0 - 5/73)		
Standard test procedure for geiger muller counters	PNPP conforms with this guide.	<Section 12.5>
<Regulatory Guide 8.7> - (Revision 0 - 5/73)		
Occupational radiation exposure records systems	PNPP conforms with this guide.	<Section 12.5>

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 8.8> - (Revision 3 - 6/78)		
Information relevant to ensuring that occupational radiation exposures at nuclear power stations will be as low as reasonably achievable	PNPP conforms with this guide with exception to Section C.2.d.3. Filters used in portable auxiliary ventilation systems are certified by the manufacturer and are replaced in accordance with the manufacturer's recommendations.	<Section 11.3.1>, <Section 11.4.1>, <Section 12.1>, <Section 12.3>, <Section 12.5>
<Regulatory Guide 8.9> - (Revision 0 - 9/73)		
Acceptable concepts, models equations and assumptions for a bioassay program	PNPP conforms to this guide.	<Section 12.3>, <Section 12.5>
<Regulatory Guide 8.10> - (Revision 1-R - 5/77)		
Operating philosophy for maintaining occupational radiation exposures as low as is reasonably achievable	PNPP conforms to this guide.	<Section 12.1>, <Section 12.5>
<Regulatory Guide 8.11> - (Revision 0 - 6/74)		
Applications of bioassay for uranium	Not applicable at PNPP.	-
<Regulatory Guide 8.12> - (Revision 1 - 1/81)		
Criticality accident alarm system	Withdrawn. In lieu of complying with <10 CFR 70.24>, PNPP has chosen to comply with <10 CFR 50.68(b)>.	<Section 9.1.1>, <Section 12.3.4>, <Section 12.5>

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 8.13> - (Revision 1 - 11/75)		
Instruction concerning prenatal radiation exposure	PNPP conforms with this guide.	<Section 12.5>
<Regulatory Guide 8.14> - (Revision 1 - 8/77)		
Personnel neutron dosimeters	PNPP conforms with this guide.	<Section 12.5>
<Regulatory Guide 8.18> - (Revision 0 - FC - 12/77)		
Information relevant to ensuring that occupational radiation exposures at medical institutions will be as low as reasonably achievable	Not applicable at PNPP.	-
<Regulatory Guide 8.19> - (Revision 1 - 6/79)		
Occupational radiation dose assessment in light-water reactor power plants design stage man-rem estimates	PNPP conforms to the administrative and procedural considerations as recommended by Section D of the guide.	<Section 12.5>
<Regulatory Guide 8.20> - (Revision 1 - 9/79)		
Applications of bioassay for I-125 and I-131	Not applicable at PNPP.	-
<Regulatory Guide 8.21> - (Revision 1 - 10/79)		
Health physics surveys for byproduct material at NRC-licensed processing and manufacturing plants	Not applicable at PNPP.	-

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.; RRRC Category)	Degree of Conformance	USAR Section/ Reference
<Regulatory Guide 8.22> - (Revision 0 - FC - 7/78)		
Bioassay at uranium mills	Not applicable at PNPP.	-
<Regulatory Guide 8.23> - (Revision 0 - FC - 2/79)		
Radiation safety surveys at medical institutions	Not applicable at PNPP.	-
<Regulatory Guide 8.24> - (Revision 1 - 10/79)		
Health physics surveys during enriched Uranium-235 processing and fuel fabrication	Not applicable at PNPP.	-
<Regulatory Guide 8.26> - (Revision 0 - 9/80)		
Applications of bioassay for fission and activation products	PNPP conforms to this guide.	<Section 12.5>
<Regulatory Guide 8.28> - (Revision 0 - 8/81)		
Audible-alarm dosimeters	PNPP conforms to this guide.	-

1.9 STANDARD DESIGNS

This section is not applicable to PNPP.

1.10 EVALUATION OF UNIT 1 OPERATIONS RESULTING FROM UNIT 2 CONSTRUCTION ACTIVITIES

The information in this Section is historical. That is, information originally provided in the Final Safety Analysis Report (FSAR) to meet the requirements of <10 CFR 50.34(b)> and was accurate at the time the plant was originally licensed, but is not intended to be updated for the life of the plant.

To meet the requirements of <10 CFR 50.34(b)(6)(vii)>, an evaluation was performed of the potential hazards to the structures, systems and components related to the safety of Unit 1, resulting from construction activities on Unit 2. This evaluation examined the physical isolation of Unit 1 from Unit 2 from both a system and security barrier standpoint as well as assessed the radiological, managerial and administrative controls.

It was determined that all Unit 1 and common mechanical piping systems are needed for Unit 1 operation. For the portions of the common mechanical piping systems required only for Unit 2 (e.g., nuclear closed cooling and containment vessel chilled water), specific isolation valves or barriers have been identified and are being incorporated into the Perry design. Modifications have been implemented, for example, the fuel pool cooling mode of the emergency closed cooling system and the fire protection seismic water supplies to common area hose stations. Unit 2 systems required, in their entirety for Unit 1 operation are the Unit 2 plant vent and turbine power complex ventilation. Unit 1 and common electrical systems (Class 1E and non-Class 1E) are needed for Unit 1 operation. The Unit 2 safety-related electrical (Class 1E) is not needed. A portion of the Unit 2 nonsafety-related electrical (non-Class 1E) is needed such as electrical supply to the service building, technical support center, Unit 2 startup transformer, Non Divisional Unit 2 batteries and plant underdrain pumps.

The radiological exposures of Unit 2 construction and testing personnel in varying locations were evaluated and estimated doses are provided in <Chapter 12>. Specific security provisions during Unit 2 construction are discussed in Section 10 of the Perry Nuclear Power Plant Security Plan.

Managerial and administrative controls exist in the form of design controls, plant procedures and training. Technical specifications were taken into consideration throughout the evaluation.

<APPENDIX 1A>

<NUREG-0737> TMI ACTION PLAN REQUIREMENTS FOR

APPLICANTS FOR NEW OPERATING LICENSES

TABLE 1A-1

<NUREG-0737> TMI ACTION PLAN REQUIREMENTS FOR APPLICANTS FOR AN OPERATING LICENSE
PNPP SUMMARY

Clarification Item	Shortened Title	Description	USAR[(1)] Reference	Comment
I.A.1.1	Shift technical advisor	1. On-shift 2. Training per LL Cat B 3. Describe long term program	<Section 13.1.2.3> <Section 13.2.3.2> Tech. Spec. 5.2.2	
I.A.1.2	Shift supervisor responsibilities	Delegate nonsafety duties	<Section 13.1.2.2>	
I.A.1.3	Shift manning	1. Limit overtime 2. Minimum shift crew	<Section 13.1.2.3> Tech. Spec. 5.2.2	
I.A.2.1	Immediate upgrade of RO and SRO training and qualifications	1. SRO experience 2. SROs be ROs, 1 year 3. 3 month training on-shift 4. Modify training 5. Facility certification	<Section 13.2.2>	
I.A.2.3	Administration of training programs	Instructors complete SRO examination	<Section 13.2.2.1.8>	
I.A.3.1	Revise scope and criteria for licensing exams	1. Increase scope 2. Increase passing garde 3. Simulator exams a. Plants with simulators	<Section 13.2.2>	

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
I.B.1.2	Independent Safety Engineering (ISE)	Function of ISE.	<Appendix 1A>	
I.C.1	Short term accident and procedure review	1. SB LOCA 2. Inadequate core cooling a. Reanalyze and propose guidelines b. Revise procedures 3. Transients & accidents a. Reanalyze and propose guidelines b. Revise procedures	<Section 13.5.2.1.6> See Note[2]	
I.C.2	Shift and relief turnover procedures	Revise procedures to assure plant status known by new shift	<Section 13.5.1.3>	
I.C.3	Shift supervisor responsibility	Corporate directive establish command duties, and revise procedures	<Section 13.1.2.2>	
I.C.4	Control room access	Establish authority and limit access	<Section 13.1.2.2>	
I.C.5	Feedback of operating experience	Review and revise procedures.	<Section 13.2.2.2.1.1>	
I.C.6	Verify correct performance of operating activities	Revise performance procedures		

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
I.C.7	NSSS vendor review of procedures	1. Low-power test program 2. Lower ascension and emergency procedures	<Section 14.2.3.2>	
I.C.8	Pilot mon. of selected emergency proc. of NTOLs	Correct procedure based on NRC sample audit	See Note[2]	
I.D.1	Control room design reviews	Preliminary assessment and schedule for correcting deficiencies		
I.D.2	Plant safety parameter display console	1. Description 2. Installed 3. Fully implemented		
I.G.1	Training during low-power testing	1. Purpose tests 2. Submit analysis and procedures 3. Training & results		
II.B.1	Reactor coolant system vents	1. Design and analyses 2. Install 3. Procedures	<Section 5.2.2> <Figure 5.1-3>	
II.B.2	Plant shielding	1. Radiation and shielding review 2. Corrective actions to assure access 3. Complete modifications 4. Equipment qualification	<Section 12.6>	

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
II.B.3	Postaccident sampling	1. Design review 2. Corrective actions 3. Procedures 4. Complete actions	<Section 9.3.6> Emergency Plan	
II.B.4	Training for mitigating core damage	1. Develop training program 2. Complete training	<Section 13.2.2.1.4.2>	
II.D.1	Relief and safety-valve test requirements	1. Describe program and schedule 2. RV & SV tests	<Section 5.2.2>, <Section 3.9.3.2.3.1.2>	
II.D.3	Valve position indication	Install in control room	<Table 7.1-4> Tech. Spec. Table 3.3.3.1-1	
II.E.1.1	Auxiliary feedwater system evaluation	1. Analysis 2. Modification		Not applicable to BWR's
II.E.1.2	Auxiliary feedwater system initiation and flow	1. Initiation a. Control grade b. Safety grade 2. Flow indication a. Control grade b. Safety grade		Not applicable to BWR's
II.E.3.1	Emergency power for pressurizer heaters	Installed capability		Not applicable to BWR's

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
II.E.4.1	Dedicated hydrogen penetrations	1. Design 2. Review and revise H control procedures 3. Install	<Section 6.2.4> <Section 6.2.5>	
II.E.4.2	Containment isolation dependability	1-4 Implement diverse isolation 5. Containment press setpoint 6. Containment purge valves 7. Radiation signal on purge valves	<Section 6.2.4> <Table 6.2-32>	
II.F.1	Accident monitoring instrumentation	1. Procedures 2. Install instrumentation a. Nobel gas monitor b. Iodine/particulate sampling c. Containment high range monitor d. Containment pressure e. Containment water level f. Containment hydrogen	<Section 6.2.5> <Section 7.3> <Section 7.6> <Section 11.5> Emergency Plan	
II.F.2	Instrumentation for detection of inadequate core cooling	1. Procedures instruments 2. Subcooling meter 3. Describe other instrumentation 4. Install additional instrumentation		Not applicable to BWR’s Not applicable to BWR’s

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
II.G.1	Power supplied for pressurizer relief valves, block valves, and level indicators	Power supply from emergency buses		Not applicable to BWR's
II.K.1	IE Bulletins	5. Review ESF valves	<Table 7.1-4>	
		10. Operability status	See Note[2]	
		17. Trip per low-level B/S		Not applicable to BWR's
		20. Prompt manual reactor trip		Not applicable to BWR's
		21. Auto SG anticipatory reactor trip		Not applicable to BWR's
		22. Aux. heat removal system procedures	<Section 7.3>	
		23. RV level, procedures	<Section 7.2.1> <Table 7.1-4>	
II.K.2	Orders on B&W plants	2. Procedures to control AFW ind of ICS		Not applicable to BWR's
		9. FMEA on ICS system		
		10. Safety-grade trip anticipatory		
		13. Thermal mechanical report		
		14. Lift frequency of PORV & SVs		
		15. Effects of slug flow of OTSGS		
		16. RCP seal damage		

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
II.K.2 (Continued)		17. Voiding in RCS		
		19. Benchmark analysis seq AFW flow		
II.K.3	Final recommendations, B&O task force	1. Auto PORV isolation		Not applicable to BWR's
		2. Report on PORV failure		Not applicable to BWR's
		3. Reporting SV & RV failures & challenges	See Note[2]	
		5. Auto trip of RCPs		Not applicable to BWR's
		a. Propose mods		Not applicable to BWR's
		b. Modify		Not applicable to BWR's
		7. Evaluation of PORV opening probability		Not applicable to BWR's
		9. PID controller		Not applicable to BWR's
		10. Applicant's proposed anticipatory trip at high power		Not applicable to BWR's
		11. Justification use of certain PORVs		Not applicable to BWR's
		12. Confirm anticipatory trip		
		a. Propose modifications		Not applicable to BWR's
		b. Modify		Not applicable to BWR's
		13. HPCI & RCIC unit levels	<Section 7.4.1.1>	
		a. Analysis		
		b. Modify		
		15. Isolation of HPCI and RCIC	<Section 7.6.1.3.2.b>	
		16. Challenges to and failure of relief valves		
		a. Study		
		b. Modify		

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
II.K.3 (Continued)		17. ECCS outages	See Note[2]	
		18. ADS actuation	<Section 7.3.1.1.2>	
		a. Study	<Section 15.6.4.2.1>	
		b. Proposed mods		
		21. Restart of LPCS & LCPI		
		a. Design		
		b. Modification		
		22. RCIC suction	<Section 5.4.6>	
		a. Procedures	<Section 7.4.1.1>	
		b. Modification		
		24. Space cooling for HPCI/ RCIC, modifications	<Section 9.4.5>	
		25. Power on pump seals		
		a. Propose mods		
		b. Modifications		
		27. Common reference level		
		28. Qual of ADS accumulators	<Section 6.8>	
		30. SB LOCA methods		
		a. Schedule outline		
		b. Model		
		c. New analyses		
		31. Plant-specific analysis	<Section 6.3.3>	
		44. Evaluate transients		
		45. Manual depressurization		
		46. Michelson concerns		
III.A.1.1	Emergency preparedness, short term	1. Comply with <10 CFR 50, Appendix E>	Emergency Plan	
		2. Comply with <NUREG-0654>		
		3. Conduct exercise		
		4. Meteorological data		

TABLE 1A-1 (Continued)

Clarification Item	Shortened Title	Description	USAR[1] Reference	Comment
III.A.1.2	Upgrade emergency support facilities	1. Establish TSC, OSC, EOF (interim basis)	Emergency Plan	
III.D.1.1	Primary coolant outside containment	Measure leak rates and establish program to keep leakage ALARA	Tech. Spec. 5.5.2	
III.D.3.3	Inplant I2 radiation monitoring	1. Provide means to determine presence 2. Modifications to accurately measure radioiodine	<Section 12.3.4> <Table 12.3-10> <Section 12.5> <Table 12.5-4>	
III.D.3.4	Control-room habitability	1. Identify and evaluate potential hazards 2. Schedule for modifications 3. Modifications	<Section 2.2.3> <Section 6.4>	

NOTES:

(1) Column provides suggested USAR section for each item. Detailed responses are provided on the pages which follow this table.
(2) Item is not specifically referenced in the USAR; details are provided in plant procedures.

RESPONSE TO REQUIREMENTS OF <NUREG-0737>

This document contains a response for each TMI-related requirement identified in <NUREG-0737> and applicable to Perry Nuclear Power Plant.

Item No. I.A.1.1

Shift Technical Advisor

REQUIREMENT

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The Shift Technical Advisor (STA) may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

PNPP has committed to provide a Shift Technical Advisor who offers shift technical support to the shift supervisor and who advises the shift supervisor on the safety status of the plant, diagnoses plant accidents, and recommends actions to mitigate the consequences of accidents. An STA at PNPP must have a bachelor degree in Engineering or related sciences or a High School diploma and sixty semester hours of college-level education in mathematics, reactor physics, chemistry, materials, reactor thermodynamics, fluid mechanics, heat transfer, electrical, and reactor control theory. In addition, an STA must have one year of professional level nuclear power plant experience. The STA's will complete additional instruction at PNPP including pertinent portions of onsite training dealing with FSAR accident analyses, technical specifications, normal and off-normal operating procedures, and Perry system operating modes and construction.

In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on shift. The STA position may be filled by an on-shift Shift Supervisor or Senior Reactor Operator provided the individual meets the Commission Policy Statement on Engineering Expertise on shift.

Item No. I.A.1.2

Shift Supervisor Administrative Duties

REQUIREMENT

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

RESPONSE

Administrative procedures will be clearly written to define the shift supervisor’s command and control responsibilities and authorities and to emphasize his responsibility for safe operation of the plant. Those functions which clearly detract from the shift supervisor’s responsibility for assuring safe operation of the plant will be assigned to other personnel not directly responsible for reactor operations.

Item No. I.A.1.3

Shift Manning

REQUIREMENT

This position defines shift manning requirements for normal operation. The letter of July 31, 1980, from D. G. Eisenhut to all power reactor licensees and applicants sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980, letter.

RESPONSE

The shift staffing described in the USAR <Section 13.1.2.3> for the Perry Plant control room provides qualified personnel staffing levels that meet the <NUREG-0737> interim guidance and <10 CFR 50.54(m)> criteria.

Working hours of unit staff who perform safety related functions are limited by administrative procedures in accordance with License Amendment 98.

The STA position may be filled by an on-shift Shift Supervisor or Senior Reactor Operator provided the individual meets the Commission Policy Statement on Engineering Expertise on shift.

Item I.A.2.1

Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications

REQUIREMENT[1]

A. Training programs shall be modified as necessary to provide
   (1) Training in heat transfer, fluid flow and thermodynamics
   (2) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged
   (3) Increased emphasis on reactor and plant transients.

B. Instructors shall be enrolled in appropriate requalification program.

C. Certifications shall be signed by the highest level of corporate management for plant operation.

D. Requalification programs shall be modified as above, grading criteria shall be modified to be consistent with licensing and additional control manipulations shall be required.

NOTE:

[1] The above “REQUIREMENT” is taken from H. R. Denton’s letter of March 28, 1980. Since the requirements in this letter extend over seven pages, they are presented here in a summary form and as applicable to precritical applicants.

RESPONSE

PNPP has committed to provide onsite training for licensed operators which includes the topics of reactor fundamentals, radiation protection, heat transfer, fluid flow, thermodynamics, plant transients, and plant systems as well as updating license candidates on procedures, plant design changes, technical specifications, and regulations with which the operator or senior operator must comply. The operator training program will be developed to insure that plant operators, appropriate staff engineers and management personnel possess the knowledge and skills necessary to recognize potentially severe accident conditions that have resulted or could result in core damage and to mitigate the consequences of such accidents.

Training instructors who teach systems, integrated responses, transients, and simulator courses have successfully completed SRO certification through approved General Electric Control Room Simulator Program or are monitored by such personnel. Subsequent to initial fuel load, these instructors shall be required to possess valid NRC SRO licenses, instructor certifications or be technically competent in the specific area and be monitored by an NRC SRO or instructor licensed individual.

The PNPP Operator Requalification Program will include various topic areas such as heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

Certification of training completed pursuant to <10 CFR 55.10a (6)>, <10 CFR 55.33a (4)>, <10 CFR 55.33a (5)> and <10 CFR 55>, shall be signed by the Site Vice President, Perry.

As Perry is a facility not in operation, the requirements for on-shift training and SRO experience as an RO for one year are not applicable. However, steps are taken to achieve an equivalency. The majority of all license applicants will have previous experience as an NRC licensed operator at another site or an RO/EWS/EOOW in the Naval Nuclear Power Program. Additionally, candidates assigned to the Operations Section will perform shift duties during the testing phase at Perry prior to initial fuel load. Whenever possible, candidates will also be sent to other operating plants to gain additional experience. Finally, each operating shift will have assigned to it a person with commercial BWR startup experience during the period from fuel load until 100% power is attained or for one year, whichever occurs later.

Item No. I.A.2.3

Administration of Training Program

REQUIREMENT

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate Senior Reactor Operator (SRO) qualifications and be enrolled in appropriate requalification programs.

RESPONSE

Personnel selected to act as instructors are individuals with previous technical training experience, civilian or military, and/or above average performance who have demonstrated the potential to effectively communicate in an instructional situation. Instructors responsible for instruction in systems, integrated responses, transients, and simulator courses complete the same training and requalification programs as NRC SRO license candidate or are monitored by such personnel. Prior to instructing, these instructors shall have successfully certified at the SRO level through approved General Electric Control Room Simulator Programs. Subsequent to initial fuel load, these instructors shall be required to possess valid NRC SRO licenses, instructor certifications or be technically competent in the specific area and be monitored by an NRC SRO or instructor licensed individual. Additionally, instructors are enrolled in a continuing program which teaches instructional skills. CEI-developed programs to develop instructional abilities are supplemented by university or vendor programs. Finally, all instructors are frequently monitored and evaluated by supervisory staff to ensure continued competency.

Item No. I.A.3.1

Revise Scope and Criteria for Licensing
Examinations--Simulator Exams (Item 3)

REQUIREMENT

Simulator examinations will be included as part of the licensing examinations.

RESPONSE

All Reactor Operator and Senior Reactor Operator license applicants will prepare to take the new licensing examinations as required by the NRC prior to fuel load. Persons seeking operator and senior operator licenses receive extensive classroom, simulator and on-the-job training.

NRC Operator License candidates utilize the General Electric Perry Simulator at Perry, Ohio for training. Time will be made available on the Perry Simulator for the simulator examination portion of the NRC license examination sequence.

Item No. I.B.1.2

Independent Safety Engineering Group

REQUIREMENT

Each applicant for an operating license shall establish an onsite Independent Safety Engineering Group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modification.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

RESPONSE

The Perry Nuclear Power Plant will implement administrative procedures to perform Independent Safety Engineering (ISE) functions in such a manner as to meet the intent of <NUREG-0737>, I.B.1.2.

Historically, ISE functions were accomplished through a dedicated Independent Safety Engineering Group composed of dedicated, full-time engineers or technically oriented individuals located onsite. The intended function of the Independent Safety Evaluation Group (ISEG) will be maintained by current Energy Harbor programs and processes that provide oversight of plant operating characteristics, NRC issuances, Licensing Information Services, and other sources of plant design and operating experience information that may indicate areas for improvement to insure overall safe operation of the station. Included within the oversight process are reviews of plant activities including Maintenance, Modifications, and Operational problems. The ISEG functions are directly incorporated into engineering, operations, performance improvement, and oversight functions through administrative processes such as the Corrective Action and Operating Experience programs. Organizational entities responsible for engineering assessment, corrective action review, oversight and assessment are structured to provide the necessary experience and independence.

Detailed recommendations regarding improvements will be presented to management through appropriate performance improvement or corrective action processes.

Item No. I.C.1

Guidance for the Evaluation and Development of Procedures for Transients and Accidents

REQUIREMENT

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in <NUREG-0660>, Appendix C (see Table C.1, Items, 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, Items 4, 12, 17, 18, 19, 20; and Table C.3, Items 6, 35, 37, 38, 39, 41, 47, 55, 57).

RESPONSE

The PNPP has been an active participant in the BWR Owners Group efforts to develop generic emergency procedure guidelines for boiling water reactors. This effort on the part of the BWR Owners Group is partially in direct response to the recommendations outlined in Item I.C.1 of <NUREG-0737>, Clarification of TMI Action Plan Requirements. As a result, these guidelines will be used as the basis for the emergency procedures to be drafted and utilized at the PNPP.

Item No. I.C.2

Shift Relief and Turnover Procedures

REQUIREMENT[1]

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

    a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

    b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

        (what to check and criteria for acceptable status shall be included on the checklist);

NOTE:

[1] This "REQUIREMENT" is taken from D. B. Vassallo's letter dated November 9, 1979, to all licensees of plants under construction since it was not provided in detail in either <NUREG-0660> or <NUREG-0737>.

c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

RESPONSE

Checklists and/or logs will be provided for the control room operators and shift supervisor. The checklists and/or logs will include items such as critical parameters, control console checks for availability and proper alignment of systems essential to the prevention and mitigation of operational transients and accidents and the identification of degraded systems or components (including time in degraded mode) that are addressed by Technical Specifications. Auxiliary operators will review plant status by log reviews. An administrative procedure will address the conduct of shift turnover.

Item No. I.C.3

Shift Supervisor Responsibilities

REQUIREMENT

Issue a corporate management directive that clearly establishes the command of duties of the shift supervisor and emphasizes the primary management responsibility for safe operation of the plant. Revise plant procedures to clearly define the duties, responsibilities and authority of the shift supervisor and the control room operators.

RESPONSE

A corporate management directive will be issued establishing the command duties of the shift supervisor that emphasizes the primary management responsibility for safe operation of the plant. Plant administrative procedures will define the duties, responsibilities and authority of the shift supervisor and control room operators.

Item No. I.C.4

Control Room Access

REQUIREMENT[1]

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

NOTE:

[1] This "REQUIREMENT" is taken from D. B. Vassallo's letter dated November 9, 1979, to all licensees of plants under construction since it is not provided in detail in either <NUREG-0660> or <NUREG-0737>.

RESPONSE

Administrative procedures will be developed to address the control of access to the control room and to define the authorities and responsibilities of plant management in the event of an emergency.

Item No. I.C.5

Procedures for Feedback of Operating Experience to Plant Staff

REQUIREMENT

Review administrative procedures to ensure that operating experience from within and outside the organization is continually provided to operators and other operational personnel and is incorporated in training programs.

RESPONSE

PNPP will participate in the INPO SEE-IN program. Procedures will be implemented to ensure that all Significant Operating Experience Reports (SOER's) and Significant Event Reports (SER's) are distributed for review, and recommendations for corrective actions appropriate to PNPP are provided to plant staff personnel and incorporated into the training program.

Item No. I.C.6

Guidance on Procedures for Verifying
Correct Performance of Operating Activities

REQUIREMENT

It is required <NUREG-0660> that licensees' procedures be reviewed and revised, as necessary, to assure than an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity (<NUREG-0585>, Recommendation 5), or both.

Implementation of automatic status monitoring, if required, will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases--one before and one after installation of automatic status monitoring equipment, if required, in accordance with Item 1.D.3.

RESPONSE

The PNPP has committed to compliance with <Regulatory Guide 1.47>, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." The status of bypassed systems shall be able to be identified by the on-shift Control Room SRO's. In addition, procedures will be developed which require the approval of an on-shift Control Room SRO to release any safety-related system or equipment for maintenance or surveillance. The approval of the on-shift Control Room SRO will also be required to return any safety-related system or equipment back into service. Procedures will also be developed to verify and document the functional acceptability of any equipment returned to service which is addressed by limiting conditions for operation in plant technical specifications. For the return-to-service of ECCS Systems, independent verification of proper systems alignment will be made unless functional testing can be performed without comprising plant safety and can prove that all equipment, valves and switches involved in the activity are correctly aligned.

Item No. I.C.7

NSSS Vendor Review of Procedures

REQUIREMENT

Operating license applicants are required to obtain reactor vendor review of their low-power, power-ascension and emergency procedures as a further verification of the adequacy of the procedures.

RESPONSE

CEI will provide for a review of low power testing, power ascension and emergency operating procedures by the NSSS vendor, General Electric Corporation, prior to implementation of these procedures.

Item No. I.C.8

Pilot Monitoring of Selected Emergency
Procedures for Near-Term Operating
License Applicants

REQUIREMENT

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of AC power, and steam-line break).

RESPONSE

The Cleveland Electric Illuminating Company is participating in the BWR Owners' Group program to finalize Emergency Procedure Guidelines for General Electric Boiling Water Reactors. Once these guidelines are converted into emergency procedures for PNPP and audited by the NRC, CEI will revise them, as necessary, before full power operation.

Item No. I.D.1

Control Room Design Review

REQUIREMENT

In accordance with Task Action Plan I.D.1, Control Room Design Reviews <NUREG-0660>, all licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

RESPONSE

CEI in conjunction with the BWR Owners' Group has conducted an assessment of the Perry Control Room to identify significant human factors deficiencies. The results of this survey are presently being evaluated to determine the priority and corrective actions required. This information should be available for NRC review in May of 1982.

We are presently awaiting NRC agreement on the BWR Owners' Group Control Room Survey program. This information had been submitted to V. A. Moore by W. J. Armstrong on August 25, 1981. A follow-up meeting was held with the NRC on March 10, 1982. No response has been received from the NRC as to the acceptability of the contents and methods of the Owners Group survey. We are awaiting an answer prior to sending the results of our survey.

Item No. I.D.2

Plant Safety Parameter Display Console

REQUIREMENT

In accordance with Task Action Plan I.D.2, Plant Safety Parameter Display Console <NUREG-0660>, each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

RESPONSE

The Cleveland Electric Illuminating Company will provide a safety parameter display for operating personnel.

CEI jointly sponsored a program, through the BWR Owners' Group, to develop appropriate parameter lists and displays for the monitor. In addition, two other alternative SPDS display sets have been defined for possible implementation at PNPP.

Simulation evaluations conducted for PNPP by General Electric have been completed. CEI has reviewed the results and specified the final SPDS design for PNPP, including the Control Room location of two SPDS video display terminals.

Item No. I.G.1

Training During Low Power Testing

REQUIREMENT

Define and commit to a special low power testing program, approved by NRC, to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

Further clarification of this item includes the need to perform a simulated loss of offsite and onsite AC power.

RESPONSE

The Cleveland Electric Illuminating Company is a member of the Licensing Review Group II (LRG-II) whose position is to develop a special low power test program using the guidelines provided in the report "BWR Owners' Group Program for Compliance with <NUREG-0737>, Item I.G.1, Training During Low Power Testing," which was transmitted to the NRC via a letter from D. B. Waters (Chairman-BWR Owners' Group) to D. E. Eisenhut (Director of Licensing-NRC) dated February 9, 1981. Licensed personnel and license candidates will participate in this training prior to full power operation, with the exception that each operating shift will see at least one turbine trip transient or load rejection by direct observation or by test critiques that include a review of actual recorded plant responses.

The LRG-II position is for each plant to review the results from preceding simulated loss of AC power tests, performed at other BWRs, in order to determine the scope of such testing on their plant. PNPP has performed this review and has concluded that conduct of the test poses an undue risk of damage to plant equipment due to the resulting high temperatures in the drywell (see PY-CEI/NRR-0338L dated September 12, 1985).

Training for station blackout events will be implemented as required by <Generic Letter 81-04>, as discussed in our September 12, 1985, letter.

Item No. II.B.1

Reactor Coolant System Vents

REQUIREMENT

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of <10 CFR 50, Appendix A>, with sufficient redundancy to ensure a low probability of inadvertent or irreversible actuation.

RESPONSE

The Perry Plant unit is provided with nineteen power-operated safety-grade relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of connection to the vent lines (main steamlines) from near the top of the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect natural accumulation of gases of the reactor core.

These power-operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves is provided in <Section 5.2.2>.

The BWR Owners’ Group position is that the requirement of single-failure criteria for prevention of inadvertent actuation of these valves, and the requirement that power be removed during normal operation, are not applicable to BWRs. These valves serve an important function in mitigating the effect of transients and provide ASME code overpressure protection. Therefore, the addition of a second “block” valve to the vent lines would result in a less safe design and a violation of the code. Moreover, the inadvertent opening of a relief valve in a BWR is a design-basis event and is a controllable transient.

In addition to these power-operated relief valves, the Perry Plant BWR/6’s included various other means of high-point venting. Among these are:

a. Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell;

b. Normally open reactor head vent line, which discharges to a main steamline;

c. Main steam-driven reactor core isolation cooling (RCIC) system turbines, operable from the control room, which exhaust to the suppression pool;

d. Main steam-driven reactor feedwater pumps operable from the control room, which exhaust to the plant condenser when not isolated. Condenser gases are continuously processed through the offgas system.

Although the power-operated relief valves fully satisfy the intent of the venting requirement, these other means also provide protection against the accumulation of noncondensibles in the reactor pressure vessel.

Under most circumstances, no selection of vent path is necessary because the relief valves (as part of the automatic depressurization system), HPCS, and RCIC will function automatically in their designed modes to ensure adequate core cooling and provide continuous venting to the suppression pool.

The reactor coolant vent line is located at the very top of the reactor vessel as shown in the schematic <Figure 5.1-3>. This 2-inch line contains two Safety Class 1 valves (B21-F001 and B21-F002) that are operated from the control room. The location of this line permits it to vent the entire reactor pressure vessel, with the exception of the reactor coolant isolation cooling (RCIC) head spray piping which comprises approximately 0.15 ft^3 of the volume above the elevation of the RPV. This small volume was considered in the original design of the RCIC system and is of no consequence to its operation. In addition, since this vent line is part of the original design for PNPP, it has already been considered in all design-basis accident analysis contained elsewhere in the FSAR.

Analyses of inventory-threatening events with very severe degradations of system performance have been conducted. These were submitted by GE for the BWR Owner's Group to the NRC Bulletins and Orders Task Force on November 30, 1979. The fundamental conclusions of those studies was that if only one ECC system is injecting into the reactor, adequate core cooling would be provided and the production of large quantities of hydrogen was avoided. Therefore, it is not desirable to interfere with ECCS functions to prevent inadvertent venting.

The small-break accident (SBA) guidelines emphasize the use of HPCS/RCIC as a first line of defense for inventory-threatening events which do not quickly depressurize the reactor. If these systems succeed in maintaining inventory, it is desirable to leave them in operation until the decision to proceed to cold shutdown is made. Thus the reactor will be vented via RCIC turbine steam being discharged to the suppression pool. Termination of this mode of venting could also terminate inventory makeup of the HPCS had failed also. This would necessitate reactor depressurization via the SRV, which of course is another means of venting.

If the HPCS/RCIC are unable to maintain inventory, the SBA guidelines call for use of ADS or manual SRV actuation to depressurize the reactor so that the low pressure core spray system can inject water. Thus, the reactor would be vented via the SRV to the suppression pool. Termination of this mode of venting is not recommended. It is preferable to remain unpressurized; however, if inventory makeup requires HPCS or RCIC restart, that can be accomplished manually by the operator. It is more desirable to establish and maintain core cooling than to avoid venting. It is emphasized, however, that emergency venting would not be in the interest of core cooling and, must be employed under Emergency Procedure Guidelines.

It is thus concluded that there is no reason to interfere with ECCS operation to avoid venting. It is further concluded that the Emergency Procedure Guidelines, by correctly specifying operation actions for HPCS, RCIC, and SRV operation, also correctly specify operator actions to vent the reactor.

Conclusion and Comparison with Requirements

The conclusions from this vent evaluation for PNPP are as follows:

a. Reactor vessel head vent valves exist to relieve head pressure (at shutdown) to the drywell via remote operator action.

b. The reactor vessel head can be vented during operating conditions via the SRVs to the suppression pool.

c. The RCIC system provides an additional vent pathway to the suppression pool.

d. The size of the vents is not a critical issue because BWR SRVs have substantial capacity, exceeding the full power steaming rate of the nuclear boiler.

e. The SRV’s vent to the containment suppression pool, where discharged steam is condensed without causing a rapid containment pressure/temperature transient.

f. The SRVs are not smaller than the NRC defined small LOCA. Inadvertent actuation is a design-basis event and a demonstrated controllable transient.

g. Inadvertent actuation is of course undesirable, but since the SRVs serve an important protective function, no steps such as removal of power during normal operation should be taken to prevent inadvertent actuation.

h. An indication of SRV position is provided in the control room. Temperature sensors in the discharge lines confirm possible valve leakage. This indication is being upgraded in accordance with <NUREG-0588>.

i. Each SRV is remotely operable from the control room.

j. Each SRV is seismically and Class 1E qualified.

k. Block valves are not required, so block valve qualifications are not applicable.

l. No new <10 CFR 50.46> conformance calculations are required, because the vent provisions are part of the systems in the plant's original design and are covered by the original design bases.

m. Plant procedures govern the operator's use of the relief mode for venting reactor pressure. These procedures will be available for Regional NRC inspection at the PNPP plant.

Item No. II.B.2

Design Review of Plant Shielding and
Environmental Qualification of Equipment
for Spaces/Systems Which May be Used
in Postaccident Operations

REQUIREMENT

With the assumption of a postaccident release of radioactivity equivalent to that described in <Regulatory Guide 1.3> and <Regulatory Guide 1.4> (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

RESPONSE

A review was conducted of the plant identified systems which were likely to contain highly radioactive fluids following a design basis LOCA. The radioactive material was assumed to be instantaneously mixed in those systems, connected either to the reactor coolant system or to the containment atmosphere, that are not isolated at the start of the accident. Nonessential systems that are isolated and have no postaccident function were not considered in the review.

After determining the systems and postaccident source distribution to be used for the shielding review, the SDC point kernel shielding code was used to calculate the associated postaccident radiation doses.

Areas which may require occupancy to permit an operator to aid in the mitigation of an accident are vital areas. The evaluation to determine the necessary vital areas included the control room, technical support center, post-LOCA hydrogen control system, containment isolation system, sampling and sample analysis areas, remote shutdown panel, ECCS alignment functions, motor control center, instrument panels, emergency power supplies, security center, and radwaste control panels. Of these it was determined that for the Perry Plant, the control room and technical support center will require continuous occupancy and the sampling station, sample analysis area, Auxiliary Building elevation 620’ east end in area of 1P57F0565B (outboard MSIV accumulator safety-related air isolation valve) and remote shutdown panel will require infrequent occupancy. The remote shutdown panel is available for frequent occupancy if required.

Item No. II.B.3

Postaccident Sampling Capability

REQUIREMENT

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a <Regulatory Guide 1.3> or <Regulatory Guide 1.4> release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a <Regulatory Guide 1.3> or <Regulatory Guide 1.4> release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (<Regulatory Guide 1.3> or <Regulatory Guide 1.4> source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

The following additional clarifications have also been taken into account in the applicant's response.

Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage consistent with the conditions stated below:

1. Demonstrate compliance with all requirements of <NUREG-0737>, II.B.3, for sampling, chemical and radionuclide analysis capability, under accident conditions.

2. Provide sufficient shielding to meet the requirements of GDC-19, assuming <Regulatory Guide 1.3> source terms.

3. Commit to meet the sampling and analysis requirements of <Regulatory Guide 1.97>, Revision 2.

4. Verify that all electrically powered components associated with postaccident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of offsite power.

5. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

6. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

7. State the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from online gas analysis equipment, if it is used.

8. Provide a method for verifying that reactor coolant dissolved oxygen is at <0.1 ppm if reactor coolant chlorides are determined to be >0.15 ppm.

9. Provide information on (a) testing frequency and type of testing to ensure long term operability of the postaccident sampling system, and (b) operator training requirements for postaccident sampling.

10. Demonstrate that the reactor coolant system and suppression chamber sample locations are representative of core conditions.

RESPONSE

The postaccident sampling system for PNPP will be installed prior to fuel load and will meet <NUREG-0737> requirements, including the above listed clarifications.

1. Sampling and some analysis will be performed with the Sentry PASS panel. Any additional analysis required will be performed by onsite technicians under approved procedures.

2. The Sentry PASS panel provides adequate shielding to meet the requirements of GDC-19, assuming <Regulatory Guide 1.3> source term. Samples are transported in lead “pigs” for laboratory analysis.

3. The sampling and analysis requirements of <Regulatory Guide 1.97>, Revision 2 for the Postaccident Sampling System are addressed in <Table 7.1-4>.

4. The PASS has battery inverters that supply 120V backup power to guarantee operation during the loss of offsite power.

5. Remotely actuated valves in the PASS are safety-related solenoid valves that have been environmentally qualified.

6. A procedure will be written in accordance with generic General Electric procedures for the determination of the extent of core damage under accident conditions.

7. The PASS does not have online gas analysis equipment for the reactor coolant system samples.

8. The Sentry PASS panel provides accurate readings of dissolved oxygen and reactor coolant chlorides.

9. The Sentry PASS panel will be routinely operated by chemistry technicians to ensure long term operability.

10. Sample locations are representative of core conditions because the samples are drawn from active lines that are recirculated through the core.

A detailed description of how Perry meets Item No. II.B.3 was provided by CEI to the NRC in letters from Mr. M. R. Edelman to Mr. B. J. Youngblood dated September 16, 1983, and October 14, 1983.

The previous paragraphs are considered historical/background for the original implementation of the PASS. The current bases for PASS are contained in the following paragraphs.

Significant improvements have been achieved since the TMI accident in the areas of understanding the risks associated with nuclear plant operations and developing better strategies for managing the response to potentially severe accidents. Recent insights into plant risks and alternate severe accident assessment tools have led the NRC staff to conclude that some TMI Action Plan items (in this case, the PASS requirements) could be revised without reducing the ability of licensees to respond to severe accidents.

In light of the above, the Boiling Water Reactor Owners Group (BWROG) developed Topical Report NEDO-32991, Revision 0, "Regulatory Relaxation for BWR Post Accident Sampling Stations (PASS)," which evaluated the PASS to determine its contribution to plant safety and accident recovery. The topical report considered the progression and consequences of core damage accidents, and assessed the accident progression with respect to plant abnormal and emergency operating procedures, severe accident management guidance, and emergency plans. The topical report concluded that the current PASS requirements developed in response to <NUREG-0737> could be eliminated since alternate means existed to obtain information that might be necessary for accident assessment.

The BWROG submitted the topical report to the NRC for review and approval in November, 2000. In performing the review of the topical report, the NRC staff reviewed the available sources of information for use by decision-makers in developing protective action recommendations and assessing core damage. Based on this review, the NRC found that the information provided by PASS is either unnecessary or is effectively provided by other indications of process parameters or measurement of radiation levels. Therefore, the NRC approved the topical report as documented in a Safety Evaluation Report (SER) dated June 12, 2001. A caveat was contained within the NRC SER that required licensees who desired to implement PASS reduction had to commit to three requirements. First, establish a capability for classifying fuel damage events at the

ALERT level threshold. The capability may utilize the normal sampling system or correlations of radiation readings to reactor coolant concentrations. Second, develop contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, suppression pool, and containment atmosphere. Third, maintain an I-131 site survey detection capability, including an ability to assess radioactive iodines released to the site environment, by using effluent monitors or portable sampling equipment.

Since the regulatory relaxation of PASS requirements would be germane to each BWR licensee, an Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-413, "Elimination of Requirements for a Post Accident Sampling System (PASS)," was approved for use. The NRC, in order to improve the efficiency of the licensing process, issued the PASS relaxation described in TSTF-413 under the Consolidated Line Item Improvement Process (CLIIP). The CLIIP was noticed for availability of use by BWR licensees in the Federal Register on December 27, 2001 (66 FR 66949) and March 20, 2002 (67 FR 13027).

PNPP requested to eliminate the PASS Technical Specification requirements, as described in the aforementioned CLIIP, by letter PY-CEI/NRR-2656L, dated October 30, 2002. Within this letter, PNPP also documented its commitment to the aforementioned caveat contained within the NRC SER. The NRC approved the submittal, as License Amendment 124, by letter PY-NRR/CEI-1087L, dated March 7, 2003.

Item No. II.B.4

Training for Mitigating Core Damage

REQUIREMENT

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

RESPONSE

Instruction to teach the use of equipment and systems to control or mitigate accidents in which the core is severely damaged has been developed and implemented. This training addresses the upgrade emergency procedures developed in response to <NUREG-0660> and <NUREG-0737>, Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents." The Perry Control Room Simulator is utilized for operator familiarization with conditions and procedures. The total scheduled presentation time for the entire program shall be 80 hours and will be integrated into the overall training program to maximize effectiveness.

Item No. II.D.1

Performance Testing of BWR and PWR Relief and Safety Valves

REQUIREMENT

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

RESPONSE

In a letter dated September 7, 1981, from D. R. Davidson to D. Eisenhut, CEI endorsed the BWR Owners' Group S/R Valve testing program. Additional, in a letter dated March 11, 1983, from M. R. Edelman to B. J. Youngblood, CEI provided information on the applicability of the generic safety/relief valve test results to the Perry Nuclear Power Plant.

Item No. II.D.3

Direct Indication of Relief and Safety Valve Position

REQUIREMENT

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE

The SRV open/close monitoring system selected for PNPP is a single channel safety grade system consisting of a sensing element and a pressure switch, connected to the discharge pipe at the downstream side of the SRV discharge pipe. The electrical output of the pressure switch operates a relay which provide input to the annunciator, process computer and indicator lights. This system will be environmentally and seismically qualified. This system is identical to that recently proposed for Grand Gulf and approved by NRC.

Item No. II.E.4.1

Dedicated Hydrogen Penetrations

REQUIREMENT

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of <10 CFR 50, Appendix A>, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

RESPONSE

The Perry Plant is designed with two 100 percent redundant hydrogen recombiners inside the containment. This position is therefore not applicable to the Perry Plant. The Postaccident External Purge System is presently designed to meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of <10 CFR 50, Appendix A>. Refer to USAR <Section 6.2.5.2.3> for additional information on combustible gas control in containment.

The present system is designed based on hydrogen generation rate calculations using <Regulatory Guide 1.7>, Revision 2. The Cleveland Electric Illuminating Company, as a member of the Hydrogen Control

Owners Group, has a program underway to improve the capability of the Mark III containment in dealing with significant amounts of hydrogen, well in excess of those considered under <10 CFR 50.44>.

PNPP procedures for the use of combustible gas control systems will be reviewed and revised, as applicable.

Item II.E.4.2

Containment Isolation Dependability

REQUIREMENT

1. Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

2. All plant personnel shall give careful consideration to the definition of essential and nonessential systems; identify each system determined to be essential; identify each system determined to be nonessential; describe the basis for selection of each essential system; modify their containment isolation designs accordingly; and report the results of the reevaluation to the NRC.

3. All nonessential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

5. The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

6. Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979, must be sealed closed as defined in SRP 6.2.4, Item II.3.f during Operational Conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

7. Containment purge and vent isolation valves must close on a high radiation signal.

RESPONSE

The containment isolation system for PNPP has been reviewed in accordance with <NUREG-0737>. The results of the review are as follows:

1. In order to evaluate the adequacy of the PNPP containment isolation system, FSAR Table 6.2-32 "Containment Isolation Valve Summary" was reviewed for accuracy, completeness and consistency with the NRC Standard Review Plan Section 6.2.4. The most significant changes appear in the columns labeled "Essential (TMI)" and "Isolation Signal."

2. Because the definition of essential and nonessential systems has been altered since the TMI-2 incident, the containment penetrations were re-evaluated as to their importance in postaccident situations. This re-evaluation was done using <Table 1A-2>. This table provides an assessment of the PNPP systems which can be considered "Essential" or "Nonessential" for isolation conditions consistent with <NUREG-0578>, Requirement 2.1.4. As used in this assessment, those systems identified as essential are regarded as indispensable or are back-up systems in the event of a loss-of-coolant accident. The nonessential systems have been judged to be not required in loss-of-coolant accident situations. However, depending

upon the circumstances, it may be highly desirable not to isolate a "nonessential" system. For this reason, the <NUREG-0578> definition of "essential" is deliberately flexible. As a result, the specification of "essential" is very judgmental with certain systems. The feedwater penetrations and some instrument air penetrations were upgraded to "essential" under the new TMI-2 definition.

3. All nonessential systems with non-manual containment isolation valves are actuated by at least one automatic isolation signal.

4. Systems, once isolated, should be capable of being quickly returned to service as the need arises. The review of the FSAR Table 6.2-32 also included examining the effect of resetting the containment isolation signal.

   All automatic isolation valves, with the possible exception of the main steam isolation valves, will remain in the "as is" position when the containment isolation signal(s) is reset.

   A further investigation into the control function of the main steam isolation valves will be made to determine if modification(s) is required to keep the valve closed after resetting the containment isolation signal. Also, those valves that are identified with a RM_c* in Table 6.2-32, may require a separate remote-manual switch in the control room.

5. An evaluation is underway to determine the minimum pressure setpoint.

6. A design review has shown that the purge valves meet BTP CSB6-4.

7. PNPP containment purge and vent valves are to close on high radiation signals. Those that do not isolate on high radiation signals are to be “sealed closed” valves.

TABLE 1A-2

ESSENTIAL/NONESSENTIAL EQUIPMENT

(Preliminary Perry Unique Listing Using The

Owner Group/GE Systems Work as a Guide)

		ESSENTIAL	COMMENTS
1.	Standby Liquid Control	Yes	Should be available as back-up to CRD system.
2.	Core Spray (High & Low Pressure)	Yes	Safety System
3.	Nuclear Closed Cooling Water	No	Used for normal operation only. Not required for DBA, but is used for the recirc., cleanup system operation and fuel pool heat exchangers.
4.	Combustible Gas Control System	Yes	Combustible gas control function necessary to eliminate hydrogen/oxygen combustible atmosphere.
5.	Automatic Depressurization System	Yes	Safety System/Control RPV pressure.

TABLE 1A-2 (Continued)

		ESSENTIAL	COMMENTS
6.	Annulus Exhaust Gas Treatment	Yes	Necessary to control emissions to environment.
7.	Containment Chiller Water Cooling	Yes	Necessary to cool system pumps and motors.
8.	Reactor Core Isolation Cooling	Yes	Necessary for core cooldown following isolation from the turbine condenser and feedwater makeup.
9.	Emergency Service Water System	Yes	Necessary to remove heat following accident. Includes the ultimate heat sink.
10.	Control Complex Chilled Water	Yes	Cools Control Room.
11.	Instrument Air	Yes	Regarded as essential because this system supports safety equipment. Back-up accumulators are available for the safety equipment should the system fail.
12.	Service Air	No	Serves no safety or shutdown function.

TABLE 1A-2 (Continued)

		ESSENTIAL	COMMENTS
13.	Main Steam[1]	Yes	Not required for shutdown but can be used as alternate cooling mode.
14.	Feedwater Line[1]	Yes	Not required for shutdown but can be used as alternate cooling mode.
15.	Reactor Water Sample	No	Not required for shutdown, but would be necessary for postaccident assessment. Postaccident sample is a separate issue.
16.	Control Rod Drive (Cooling)	Yes	No credit taken for reflood, but is desirable.
17.	Reactor Water Cleanup[1]	Yes	Not required during and immediately following an accident. Necessary in long term recovery.
18.	Radwaste Collection	No	Not required for shutdown.
19.	Recirculation System	No	Not required for J-P plants because core can be cooled by nat. cir.

TABLE 1A-2 (Continued)

		ESSENTIAL	COMMENTS
20.	RHR-Shutdown Cooling[1]	Yes	Not ESF, but desirable to use if available. Not redundant, but safety grade.
21.	RHR-Containment Spray	Yes	Necessary to control pressure.
22.	RHR-Suppression Pool Cooling	Yes	Heat Sink for postaccident cooling.
23.	RHR-LPCI Function	Yes	Safety function.
24.	RCIC Steam Supply Line[1]	Yes	Used in conjunction with RCIC.
25.	Drywell Cooling	No	Used only in normal operation. Desirable to keep running.
26.	Demineralized Water	No	Not assumed available in ECCS analysis.
27.	Condensate Water	Yes	Not assumed available in ECCS analysis, but is used in RCIC and HPCS.

TABLE 1A-2 (Continued)

		ESSENTIAL	COMMENTS
28.	Fuel Pool Cooling	No	Boiling O.K., but make-up is necessary. Heat exchangers cooled by NCCW system.
29.	Traversing In-Core Probe (TIP)	No	Not required for reactor shutdown cooling.
30.	Fire Protection System	No	Availability is necessary, as the “accident” may be the result of a fire.
31.	Fire Protection System	No	Serves no purpose during and immediately after accident. Longer term availability necessary.
32.	Safety-Related Instrument Air	Yes	Use for ADS function.
33.	Nonsafety-Related Instrument Air	No	Serves no safety or shutdown function.
34.	Suppression Pool Cleanup	No	Not required for reactor shutdown.

NOTE:

(1) These systems (or portions of these systems) have been changed from the GE/Owner’s group designation of nonessential to essential.

Item No. II.F.1

Additional Accident-Monitoring Instrumentation

REQUIREMENT

The <NUREG-0737> requirements evolved from three basic requirements in <NUREG-0578> (Items 1 through 3 below) and were subsequently clarified by NRC letters dated September 27, 1979, and November 9, 1979. The letters also included additional requirements resulting in Items 4 through 6 below. A summary of these items is as follows:

1. Noble gas effluent radiological monitors;

2. Provisions for continuous sampling for plant effluents for postaccident releases of radioactive iodines and particulates, and onsite laboratory facilities;

3. Containment high-range radiation monitor;

4. Containment pressure monitor;

5. Containment water level monitor; and

6. Containment hydrogen concentration monitor.

The individual requirements for each item have been omitted from this synopsis due to their length and detail required for an adequate recitation.

RESPONSE

1. and 2. Sampling systems with high range noble gas monitors and particulate and radioiodine (P/I) collectors will be added to the following effluent flow paths:

a) Main Plant Vent

b) Heater Bay/Turbine Building Vent

c) Offgas Vent

This equipment will provide monitors with range extension to include the high level noble gas concentration in accordance with <Regulatory Guide 1.97> and <NUREG-0737>. Each monitor consists of an intermediate and high range channel. The intermediate range channel has a detectable range from 1.7 x 10^{-3} μCi/cc (Xe-133). Power is to be derived from diesel backed Class 1E buses.

This equipment will also contain three P/I collectors which will be used in conjunction with the normal range radiation monitor P/I collectors. They will be used to continuously collect P/I samples through the required range. The high range sampling systems will reflect the following design criteria:

1) Collection capability of 0.7 μCi/cc each of gaseous iodine and particulates, which is based on the Perry specific shielding envelope.

2) Provisions to limit occupational dose to personnel through shielding and operating procedures; applicable for the sampling station design, sample handling and transport operations, and analysis operations.

3) Representative sampling via guidelines of ANSI N13.1 - 1969.

4) Sampling systems initiated by a containment isolation signal or associated by plant effluent normal range monitor signals.

5) Analysis capabilities in the Technical Support Center via multichannel analyzers and detectors to determine iodine and particulate concentrations.

3. High range gamma monitors will be added to the reactor building and to the drywell to provide conformance with <NUREG-0737> Table II.F.1-3 with a range of 1 R/hr to 10^7 R/hr and to respond to the requirements of <Regulatory Guide 1.97>, Revision 2.

   They will be powered from independent 120V ac, diesel-backed buses and will be provided with continuous readout and multipoint recorders in the control room. Although the calibration procedure for the monitors will vary from model to model, it will be by calibration source below 10 R/hr., and by electronic signal input for ranges above 10 R/hr.

   Two monitors are located in the primary containment (drywell) at approximately core midplane, 630’ elevation, spread about 32° apart centered at 270° azimuth for Unit 1.

Two monitors shall be located in the Reactor Building at about the 689′ level with a 30° spread about the 225° azimuth for Unit 1.

4. Containment Pressure Monitors are to be added in the plant design to meet <NUREG-0737> and <Regulatory Guide 1.97> requirements.

   Two redundant channels will be provided with 2 monitors per channel meeting the range requirements of -5 psig to 60 psia. Normal range monitors are provided to cover a range of 10 inch Hg to 20 psig. Wide range monitors are utilized and reflect a 10 inch Hg to 60 psig range. Qualification of these channels will be in accordance with PNPP's environmental qualification program. Class 1E power is supplied to these channels. Continuous indication and recording will be provided. Overall accuracy of the containment pressure measurement is less than ±1%.

5. Containment suppression pool water level monitors are to be added in the plant design to meet <NUREG-0737> and <Regulatory Guide 1.97> requirements.

   Two redundant channels will be provided with three monitors per channel to meet the level range requirements at the bottom of ECCS Suction Line level to 5′ above normal suppression pool level. Class 1E power is supplied to these channels. Continuous indication and recording will be provided. Overall accuracy of the suppression pool water level monitors is less than ±1%.

6. Containment and drywell hydrogen monitors have been added to meet <NUREG-0737> and <Regulatory Guide 1.97> requirements.

Two redundant channels are provided to meet hydrogen concentration requirements (0 to 10%). These channels are functional from 12 psia to containment and drywell design pressure conditions. Four sample points (Containment Dome, Drywell Dome, Drywell, and Suppression Pool Area) are utilized for each channel. Class 1E power is supplied to these channels. Continuous indication and recording is provided. Overall accuracy of the containment and drywell hydrogen monitors is -1% to +10% of the actual hydrogen concentration at the recorder readout.

Item No. II.F.2

Inadequate Core Cooling Instruments

REQUIREMENT

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including Primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures and a schedule for installing the equipment shall be provided.

RESPONSE

The response to this item was provided to the NRC in letters dated January 14, 1983, November 1, 1983, and January 14, 1985.

Item No. II.K.1.5

Safety-Related Valve Position

REQUIREMENT

Review all valve positions, positioning requirements, positive controls, and related test and maintenance procedures to ensure proper ESF functioning.

RESPONSE

Perry Nuclear Power Plant is equipped with valve position status monitoring that satisfies the requirements of <Regulatory Guide 1.47> as discussed in FSAR Section 7.1. Perry Plant procedures for tagging, maintenance and surveillance will assure verification of valve position status on the affected portions of system to verify ESF systems are functional after the performance of surveillance tests, and maintenance activities. These plant procedures will be available for review by Region III Division of Inspection and Enforcement, approximately six months prior to fuel load.

Item No. II.K.1.10

Safety-Related System Operability Status Assurance

REQUIREMENT

Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to ensure that operability status is known.

RESPONSE

Perry Plant procedures for removing safety-related systems from service and restoring to service will assure the operability status is known and can be identified by the on-shift Control Room SRO's. Release of all ESF equipment from service will require an on-shift Control Room SRO's approval. Plant procedures will include verification of operability of safety-related equipment after restoration following surveillance and maintenance activities. These procedures will be available for review by Region III Division of Inspection and Enforcement, approximately six months prior to fuel load.

Item No. II.K.1.22

Auxiliary Heat Removal System

Procedures

REQUIREMENT

For boiling water reactors, describe automatic and manual actions for proper functioning of auxiliary heat removal systems when FW system is not operable.

RESPONSE

Initial Core Cooling

Following a loss of feedwater and reactor scram, a low reactor water level signal (Level 2) will automatically initiate high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems. These systems operate in the reactor coolant make up injection mode to inject water into the vessel until a high water level signal (Level 8) trips the system.

Following a high reactor water Level 8 trip, the HPCS System will automatically re-initiate when reactor water level decreases to low water Level 2. The RCIC System will automatically re-initiate after a high water Level 8 trip. (See response to II.K.3.13).

The HPCS and RCIC Systems have redundant supplies of water. Both HPCS and RCIC can take suction from either the condensate storage tank (CST) or suppression pool. Normally RCIC takes suction from the condensate storage tank (CST). The HPCS System suction will automatically transfer from the CST to the suppression pool if the CST water is depleted or the suppression pool water level increases to a high level.

The RCIC System suction is automatically transferred from the CST to the suppression pool, when the CST low level is reached. The operator can manually initiate the HPCS and RCIC Systems from the control room before the Level 2 automatic initiation level is reached. The operator has the option of manual control after automatic initiation and can maintain reactor water level by throttling system flow rates. The operator can verify that these systems are delivering water to the reactor vessel by:

a. Verifying reactor water level increases when systems initiate.

b. Verify systems flow using flow indicators in the control room.

c. Verify system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to the reactor.

Therefore, the HPCS and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as Low Pressure Core Spray (LPCS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

Containment Cooling

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety relief valve (SRV) discharge and RCIC turbine exhaust to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in the containment (suppression pool) cooling mode, i.e., RHR suction from and discharge to the suppression pool.

The operator could verify proper operation of the RHR system containment cooling function from the control room by:

a. Verifying RHR and Emergency Service Water (ESW) system flow using system control room flow indicators.

b. Verify correct RHR and ESW system flow paths using control room position indication of motor-operated valves.

c. On branch lines that could divert flow from the required flow paths, close the motor-operated valves and note the effect on RHR and ESW flow rate.

Even though the RHR is in the containment cooling mode, core cooling is its primary function. Thus, if a high drywell pressure signal or low reactor water level is received at any time during the period when the RHR is in the containment cooling mode, the RHR system will automatically revert to the LPCI injection mode. The Low Pressure Core Spray (LPCS) system would automatically initiate and both the LPCI and LPCS systems would inject water into the reactor vessel if the reactor pressure is below system discharge pressure.

Extended Core Cooling

When the reactor has been depressurized, the RHR system can be placed in the long term shutdown cooling mode. The operator manually terminates the containment cooling mode of one of the RHR containment cooling loops and places the loop in the shutdown cooling mode.

In this operating mode, the RHR system can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

Item No. II.K.1.23

Reactor Vessel Level Instrumentation

REQUIREMENT

For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status.

RESPONSE

Reactor vessel water level control room indication is continuously provided by 5 sets (range) of level monitors for normal, transient and accident conditions. "Top of Active Fuel" (363.5" above vessel zero) is the reference level zero for all sets. Those monitors used to provide automatic safety equipment initiation are arranged in a redundant array with two instruments in each of two or more independent electronic divisions.

a. Shutdown water-level range: 1 channel with level indicator in the control room is used to monitor reactor-water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The instrument is calibrated for 120°F at 0 psig in the vessel and 90°F in the drywell. The reactor vessel nozzles taps utilized for this channel are at 518" above vessel zero and at the top of the Head Spray flange (approximately 867" above vessel zero).

b. Upset water-level range: 1 channel with level recorder in the control room is utilized to provide water level indication extended above the upper range of the narrow-range water level monitors. The instrument is calibrated for saturated water and steam conditions at 1,025 psig in the vessel and normal operating temperature in the drywell. The reactor vessel nozzle taps utilized for this range are identical to the shutdown water level monitor.

c. Narrow water-level range: 3 channels with 3 level indicators and 1 recorder in the control room are utilized by the feedwater control system and reactor plant safeguards. The instruments are calibrated for saturated water and steam conditions at 1,025 psig in the vessel and normal operating temperature in the drywell. Water level switch trip uncertainty is ±1.5″ of water level at calibration conditions. These monitors utilize the reactor vessel nozzle taps at 518″ and 606″ above vessel zero.

d. Wide water-level range: 3 channels with 2 level recorders and 1 level indicator in the control room are provided for reactor plant safeguards to monitor vessel water level using the 364″ and 606″ level reactor vessel taps. The instruments are calibrated for 1,025 psig in the vessel, normal operating temperature in the drywell, and 20 Btu/lb subcooling below the 518″ level reactor vessel nozzle tap and saturated water and steam conditions above this tap with no jet pump flow. Water level switch trip uncertainty is ±6″ of water level at calibration conditions.

e. Fuel zone, water-level range: 3 channels with 2 level indicators and one recorder are utilized to provide water level indication above the top of the active fuel elements to below the bottom of the fuel elements. The 3 channels utilize the RV nozzle tap 606″ above vessel zero. Two channels also use the jet pump diffuser level tap, 156.5″ above vessel zero while the third channel uses the pressure below core plate tap, 24″ above vessel zero. The instruments are calibrated for saturated water and steam conditions 0 psig in the vessel and the drywell with no jet pump. Water level indication uncertainty is ±6” of water level at calibration conditions.

The safety-related systems or functions served by safety-related reactor water level instrumentation are:

Reactor Protection System (RPS)
Reactor Core Isolation Cooling System (RCIC)
High Pressure Core Spray System (HPCS)
Low Pressure Core Spray System (LPCS)
Residual Heat Removal/Low Pressure Injection (RHR/LPCI)
Automatic Depressurization System (ADS)
Nuclear Steam Supply Shutoff System (NS)[4]

All systems automatically initiate at one of three designated low reactor water levels. In addition, the RCIC and HPCS systems shutdown on high reactor water level. The RCIC and HPCS systems automatically restart if low reactor level is reached again. The RPS initiates a scram at either a high or low reactor water level.

Additional instrumentation which the operator can use to determine changes in reactor coolant inventory or other abnormal conditions are:

Drywell High Pressure
Containment High Radioactivity Levels
Suppression Pool High Temperature
Safety Relief Valve (SRV) Discharge High Temperature SRV Position Indication (Pressure)
High/Low Feedwater Flow Rates
High/Low Main Steam Flow
High Containment, Steam Tunnel and Equipment Area Differential Temperatures
High Differential Flow-Reactor Water Cleanup System

Abnormal Reactor Pressure

High Suppression Pool Water Level

High Drywell and Containment Sump Fill and Pumpout Rate High Drywell Sump Level (Flow Rate)

Valve Stem Leakoff High Temperature

Low RCIC Steam Supply Pressure

High RCIC Steam Supply Flow

Low Main Steam Line Pressure

An example of the use of this additional information by the operator is as follows: Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High SRV discharge temperature open position indication (pressure) would pinpoint loss of coolant via an open valve.

Other instrumentation that can signal abnormal plant status but does not necessarily indicate loss of coolant are:

High Neutron Flux

High Process Monitor Radiation Levels

Main Turbine Status Instrumentation

Abnormal Reactor Recirculation Flow

High Electrical Current (Amperes) to Recirc Pump Motors

Operators will be instructed in use of other available information to verify proper functioning of safety systems as a continuing part of training.

Additional control room indication as a result of <Regulatory Guide 1.97> evaluations is addressed in FSAR Table 7.1-4.

Item No. II.K.3.3

Reporting Safety and Relief Valve Failures Promptly and Challenges Annually

REQUIREMENT

Ensure that any PORV or safety valve that fails to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

RESPONSE

The special post-TMI reporting requirements for SRV challenges/failures are no longer required. Requirements for reporting all challenges to the SRV’s were deleted by License Amendment 120.

Item No. II.K.3.13

Separation of HPCI and RCIC System Initiation Levels

REQUIREMENT

Currently, the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system both initiate on the same low water level signal and both isolate on the same high water level signal. The HPCI system will restart on low water level, but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the RCIC system initiation logic should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analysis.

RESPONSE

CEI has endorsed the position of the BWR Owners' Group delineated in the letter from Mr. R. H. Buchholz to Mr. D. G. Eisenhut dated October 1, 1980. That position is basically that "...the current design is satisfactory, and a significant reduction in thermal cycles is not necessary;" and "...no significant reduction in thermal cycles is achievable by separating the setpoints."

Modification of the initiation logic for automatic restart of the RCIC system on low water level has been incorporated into the Perry design and is discussed in <Section 7.4.1.1>.

Item No. II.K.3.15

Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC

REQUIREMENT

The high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe break detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe break detection circuitry should be modified so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

RESPONSE

The BWR Owners' Group has evaluated this issue and has recommended the addition of a time delay to the HPCI/RCIC break detection circuitry. CEI contracted with General Electric to provide this change to the RCIC steam line break detection circuitry. This change is discussed in <Section 7.6.1.3.2b>.

Item No. II.K.3.16

Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification

REQUIREMENT

The record of relief valve failures to close for all boiling-water reactors (BWRs) in the past 3 years of plant operation is approximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

1. Additional anticipatory scram on loss of feedwater,

2. Revised relief valve actuation setpoints,

3. Increased emergency core cooling (ECC) flow,

4. Lower operating pressures,

5. Earlier initiation of ECC systems,

6. Heat removal through emergency condensers,

7. Offset valves setpoints to open fewer valves per challenge,

8. Installation of additional relief valves with a block or isolation valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME Code,

9. Increasing high steam line flow setpoint for main steam line isolation valve (MSIV) closure,

10. Lowering the pressure setpoint for MSIV closure,

11. Reducing the testing frequency of the MSIVs,

12. More stringent valve leakage criteria, and

13. Early removal of leaking valves.

An investigation of the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

RESPONSE

The Cleveland Electric Illuminating Company has participated in a BWR Owners' Group evaluation of possible ways to reduce the challenges and failures of safety relief valves. The results of this feasibility study were submitted to the NRC in a letter from D. B. Waters to D. G. Eisenhut dated March 31, 1981. The study concluded that BWR/6 plants already include design features which significantly reduce the likelihood of stuck open relief valve (SORV) events; no further design modifications are necessary. It is the Cleveland Electric Illuminating Company's position that further modifications to the Perry Nuclear Power Plant would not significantly reduce the frequency of SORV events.

Item No. II.K.3.17

Report on Outages of Emergency Core-Cooling
Systems Licensee Report and Proposed Technical Specification Changes

REQUIREMENT

Several components of the emergency core cooling (ECC) systems are permitted by Technical Specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes for the outages (i.e., controller failures, spurious isolation).

RESPONSE

In 1981, PNPP was in the construction phase and therefore did not have five years of previous ECCS outage data to provide in order to meet this reporting request. Although the intent of this item was for the NRC staff to quickly develop a historical data base from the five years previous to 1981 in order for them to evaluate whether a need existed for cumulative outage requirements in the Technical Specifications, CEI responded to this item with a commitment to provide data in the future when it became available.

Therefore, ECCS outage reports were submitted for the years 1986 through 1989. However, since the original NRC evaluation of the data base had already been completed, these annual reports were discontinued. ECCS component failure data and system reliability data are reported to INPO’s Nuclear Plant Reliability Data System (NPRDS) on an ongoing basis. Also, significant problems with Emergency Core Cooling Systems are reported to the NRC in accordance with <10 CFR 50.73>.

Item No. II.K.3.18

Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences

REQUIREMENT

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor vessel water level, provided no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

RESPONSE

Cleveland Electric Illuminating Company has participated in the BWR Owners' Group evaluation of logic modifications to simplify ADS actuation. The results of this study were submitted to the NRC in a letter from D. B. Waters to D. G. Eisenhut dated March 31, 1981. The BWR O/G reevaluated the recommendations due to recently identified conflicts between the proposed modifications to ADS actuation logic and the Emergency Procedures Guidelines. As discussed in the February 5, 1982, letter from T. J. Dente to D. G. Eisenhut, the BWR O/G provided a supplement to the original owners' group report in a letter from T. J. Dente to D. G. Eisenhut dated October 28, 1982.

Based on the BWR Owners' Group design modification options, found to be acceptable by the NRC staff per letter dated April 27, 1983, from B. J. Youngblood to M. R. Edelman. CEI will remove the high drywell pressure trip in conjunction with the addition of a manual switch which inhibits ADS actuation. The design details of this modification were submitted to the NRC in a letter from M. R. Edelman to B. J. Youngblood dated July 1, 1983. This modification will be implemented before scheduled fuel load date. It is discussed in <Section 7.3.1.1.1.2> and <Section 15.6.4.2.1>.

Item No. II.K.3.21

Restart of Core Spray and Low Pressure
Coolant-Injection Systems

REQUIREMENT

The core spray and low pressure coolant injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to ensure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

RESPONSE

The Cleveland Electric Illuminating Company endorses the BWR Owners' Group position in the letter from D. B. Waters to D. G. Eisenhut dated December 29, 1980. That position is the current LPCI, LPCS and HPCS system designs are adequate and no design changes are required.

Originally, a modification was planned for the HPCS system as discussed in the LRG-II position paper for Issue 1-RSB. This automatic reset modification of the HPCS would reset the auto-initiation signal for low water level and block the continuing auto-initiation signal for high drywell pressure to allow auto-restart of HPCS pump on low water level after the operator stopped the HPCS pump. Decrease in drywell pressure below trip level returns HPCS logic to original status.

However, the NRC current position, identified in a letter from J. R. Miller to D. L. Holtzacher dated February 26, 1982, is that the automatic restart of HPCS after manual termination is optional and not necessarily required. The following justification is provided for not modifying the HPCS logic. A revised LRG-II position was submitted May 17, 1982, to reflect this justification.

Immediately following a LOCA that produces either high drywell pressure or low reactor water level, the HPCS will automatically start. Injection of emergency cooling water into the reactor will occur. Flow from the high pressure core spray system is automatically terminated when the reactor water level reaches its high level trip point (Level 8). This control feature prevents unnecessary flooding of the reactor vessel and steamlines. Termination of HPCS injection can occur either automatically or by operator action. In the event of the former, the HPCS system will restart automatically if and when reactor water level decreases from the high level trip point to the low level initiation setpoint. For the latter event, a manual action is required to restart HPCS. It was the NRC’s concern for reliance upon the operator to restart the HPCS after manual termination that prompted the proposed design modification. Such a modification is not necessary for the following reasons:

1. The ECCS logic design which permits operator intervention is based on a legitimate assumption that the operators are not likely to prematurely terminate ECCS flow and thereby jeopardize the core cooling process. In actual practice, one of the highest priority activities for an operator in an accident situation is to assure that emergency systems have started correctly and are effectively maintaining core coverage. This guidance is provided to the operator through the plant’s emergency operating procedures.

If the operator should terminate HPCS system flow, such termination would be based on event-specific conditions, such as:

(A) Adequate coolant flow from other systems (Feedwater, RCIC) is available.

(B) HPCS system equipment problems (gross seal leakage, pipe breaks, equipment flooding),

(C) Required vessel coolant makeup rate much less than HPCS system capability (6,000 gpm) and well within RCIC system capability (600 gpm).

2. For the long term core cooling situation, the plant operators manually set up the auxiliary systems to support eventual termination of the incident. Consequently, adequate core cooling is dependent upon correct operator actions. Such actions are not constrained by strict time requirements. This aspect of ECCS design is considered fully acceptable because of the time available between attaining Level 1 and the occurrence of high fuel clad temperatures.

3. A key incentive of vessel water level control is to keep the core covered but also to prevent water level from reaching Level 8 where in addition to HPCS, the RCIC and feedwater systems (if operating) would be tripped off.

4. Automatic vessel water level control will be available from the RCIC system. This system will be capable of automatic restart on Level 2 after automatic termination at Level 8 as provided for in response to TMI Action Plan Item II.K.3.13.

5. Inadequate core cooling as a result of the operator failing to reinitiate the HPCS system would not occur because eventually the ADS initiation level would be reached. This would result in reactor blowdown and core flooding by the low pressure ECCS.

The manual override option is deliberate and is considered to be an important safety feature of the BWR ECCS network. This feature provides the plant operators with flexibility for dealing with unforeseen but credible conditions requiring a particular system to be shut down. This option, complemented by the other means available to automatically maintain adequate core cooling, provides adequate justification for not implementing the HPCS system automatic restart after manual termination modification.

No. II.K.3.22

Automatic Switchover of Reactor Core Isolation Cooling System Suction -- Verify Procedures and Modify Design

REQUIREMENT

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

RESPONSE

The RCIC pump suction is provided with automatic switchover from condensate storage tank to suppression pool, as described in Perry FSAR Section 7.4.1.1.

Item No. II.K.3.24

Confirm Adequacy of Space Cooling for High-Pressure Coolant Injection and Reactor Core Isolation Cooling Systems

REQUIREMENT

Long term operation of the reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) system may require space cooling to maintain the pump room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating current power. The RCIC and HPCI systems should be designed to withstand a complete loss of offsite alternating current power to their support systems, including coolers, for at least 2 hours.

RESPONSE

PNPP utilizes safety-related pump rooms cooled by unit coolers and support systems designed to withstand the consequences of a complete loss of offsite AC power. Loss of offsite AC power results in power being supplied from the engineered safety features bus. Refer to <Section 9.4.5> for a further discussion of engineered safety features ventilation systems.

Item No. II.K.3.25

Effect of Loss of Alternating-Current Power on Pump Seals

REQUIREMENT

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

RESPONSE

The Cleveland Electric Illuminating Company has participated in the BWR Owners' Group evaluation of the effect of the loss of pump seal cooling for a period of 2 hours. This evaluation was submitted in a letter from D. B. Waters to D. G. Eisenhut, dated May 1981. The study indicates that the loss of pump seal cooling for 2 hours is not a safety problem, but may require seal repairs prior to resuming operation. Even in the case of both seal cooling systems failing, followed by extreme degradation of the pump seals, the primary coolant loss is analyzed to be less than 70 gallons per minute. Consequently, no hazard to the health and safety of the public will result from total loss of recirculation pump seal cooling water.

In addition, a supplement of the BWR Owners' Group evaluation was submitted in a letter from T. J. Dente to D. G. Eisenhut dated September 21, 1981. This supplement describes three tests performed on Representative BWR reactor recirculation pumps in which all seal cooling water was lost. The test results show that pump seal leakage is acceptably low following a loss of seal cooling water for as long as two hours. These test results are representative and bounding for the Byron Jackson reactor recirculation pumps utilized at Perry.

Item No. II.K.3.27

Provide Common Reference Level for Vessel Level Instrumentation

REQUIREMENT

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

RESPONSE

The Cleveland Electric Illuminating Company's position is to provide all reactor vessel water level instruments referenced to the top of the active fuel.

In addition, this common reference level for all water level indicators will be incorporated in operator training, training documents and maintenance procedures for Perry.

Item No. II.K.3.28

Verify Qualification of Accumulators on ADS Valves

REQUIREMENT

Safety analysis reports claim that air or nitrogen accumulators for the ADS valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated the ECC systems are designed to withstand a hostile environment and still perform their function 100 days after an accident. The Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

RESPONSE

The ADS accumulators are designed to provide two SRV actuations at 70% of drywell design pressure, which is equivalent to 4 actuations at atmospheric pressure. The ADS valves are designed to operate at 70% of drywell design pressure because that is the maximum pressure for which rapid reactor depressurization through the ADS valves is required. The greater drywell design pressures are associated only with the short duration primary system blowdown in the drywell immediately following a large pipe rupture for which ADS operation is not required. For large breaks which result in higher drywell pressure, sufficient reactor depressurization occurs due to the break to preclude the need for ADS. One ADS actuation at 70% of drywell design pressure is sufficient to depressurize the reactor and allow inventory makeup by the low pressure ECC systems. However, for conservatism, the accumulators are sized to allow 2 actuations at 70% of drywell design pressure.

The ADS accumulators and piping from the receiver tanks are ASME Section III, Class 3 safety grade components. The pneumatic supply system for the ADS accumulators is provided by the safety-related instrument air system, described in FSAR Section 6.8.

Item No. II.K.3.30

Revised Small-Break Loss-of-Coolant Accident

Methods to Show Compliance with <10 CFR 50, Appendix K>

REQUIREMENT

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with <10 CFR 50, Appendix K> should be revised, documented and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test Facilities.

RESPONSE

The General Electric Company has evaluated the NRC request to demonstrate the BWR small-break LOCA analysis methods are in compliance with <10 CFR 50, Appendix K>. Documentation that GE's present analytical methods are acceptable was provided in a letter from R. H. Buchholz, GE to D. G. Eisenhut dated June 26, 1981.

Item No. II.K.3.31

Plant-Specific Calculations to Show Compliance
with <10 CFR 50.46>

REQUIREMENT

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with <10 CFR 50.46> should be submitted for NRC approval by all licensees.

RESPONSE

The Perry Plant specific analysis using NRC approved models is provided in <Section 6.3.3>.

Item II.K.3.44

Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure

REQUIREMENT

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result in a stuck-open relief valve should be included in this category.

RESPONSE

The Cleveland Electric Illuminating Company jointly sponsored, through the BWR Owners' Group, an evaluation of the worst anticipated transient (loss of feedwater event) with the worst single failure (loss of a high pressure inventory makeup or heat removal system) to demonstrate adequate core cooling capability. These results were submitted to the NRC via a letter from D. B. Waters, Chairman BWR Owners' Group, to D. G. Eisenhut, Director NRC, dated December 29, 1980. NRC letter "<NUREG-0737>, Item II.K.3.44 – Evaluation of Anticipated Transients Combined with Single Failure," dated August 7, 1981, from D. G. Eisenhut to BWR Owners Group Licensees found this report acceptable subject to licensee confirmation that assumptions and initial conditions were representative. Subsequent PNPP letter from D. R. Davidson to D. G. Eisenhut, "<NUREG-0737>, Item II.K.3.44" dated November 3, 1981, provided that confirmation. A summary of the results of the analysis follows.

The anticipated transients in NRC <Regulatory Guide 1.70>, Revision 3 were reviewed for all BWR product line BWR/2 through BWR/6 from a core cooling viewpoint. The loss of feedwater event was identified to be the most limiting transient which would challenge core cooling. The BWR/6 is designed so that the HPCS or ADS with subsequent low pressure makeup is independently capable of maintaining the water level above the top of the active fuel given a loss of feedwater. The detailed analysis shows that even with the worst single failure in combination with the worst transient the core remains covered.

Furthermore, even with degraded conditions involving one SORV in addition to the worst transient with the worst single failure, studies show that the core remains covered during the whole course of the transient either due to RCIC operation or due to automatic depressurization via the ADS or manual depressurization by the operator so that low pressure inventory makeup can be used.

It is concluded that for anticipated transients combined with the worst single failure, the core remains covered. Additionally, it is concluded that for severely degraded transients beyond the design basis where it is assumed that an SRV sticks open and an additional failure occurs, the core remains covered with proper operator action.

For power uprate to 3,758 MWt, the Loss of Feedwater transient with the worst single failure was re-analyzed (Reference NEDC-31984P, Generic Evaluations of General Electric Boiling Water Reactor Power Uprate Volume I Licensing Topical Report, July 1991). Results of the analysis indicate that the use of RCIC alone will meet the acceptance criteria of maintaining water level inside the shroud above the top of active fuel.

Item No. II.K.3.45

Evaluation of Depressurization with Other Than ADS

REQUIREMENT

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) [e.g., early blowdown with one or two safety relief valves (SRVs)] should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

RESPONSE

The Cleveland Electric Illuminating Company participated in the BWR Owners' Group generic evaluation of alternate modes of depressurization other than full actuation of the ADS. The results of this program were submitted to the NRC in a letter from D. B. Waters to D. G. Eisenhut dated December 29, 1980. The BWR Owners' Group evaluation showed that vessel integrity limits are not exceeded for full blowdown, and slower depressurization rates have little benefit to vessel fatigue.

Item No. II.K.3.46

Michelson Concerns on the Importance of Natural Circulation During a Very Small Break LOCA and Other Related Items

REQUIREMENT[1]

A number of concerns related to decay heat removal following a very small break LOCA and other related items were questioned by Mr. C. Michelson of the Tennessee Valley Authority. These concerns were identified for PWRs. GE was requested to evaluate these concerns as they apply to BWRs and to assess the importance of natural circulation during a small-break LOCA in BWRs. GE has not yet responded to the Michelson concerns. A brief description of natural circulation was addressed in NEDO-24708. The submittal was incomplete, however, in that natural circulation for purpose of depressurizing the reactor vessel was not addressed. GE should provide a response to the Michelson concerns as they relate to BWR plants.

RESPONSE

General Electric Company has provided a response to the questions posed by Mr. C. Michelson as they relate to BWR plants. These responses were prepared on behalf of the BWR Owners' Group and issued in a letter to Mr. D. F. Ross of the NRC from R. H. Buchholz of GE dated February 21, 1980, and titled "Response to Questions Posed by Mr. C. Michelson."

NOTE:

[1] This REQUIREMENT is taken from <NUREG-0626> since it is not provided in detail in either <NUREG-0660> or <NUREG-0737>.

Item No. III.A.1.1

Upgrade Emergency Preparedness

REQUIREMENT

Comply with “Emergency Facilities,” <10 CFR 50, Appendix E>, <Regulatory Guide 1.101>, “Emergency Planning for Nuclear Power Plants,” and for the offsite plans, meet essential elements of <NUREG-75/111> (Reference 28) or have a favorable finding from FEMA.

RESPONSE

This information is found in the PNPP Emergency Plan.

Item III.A.1.2

Upgrade Emergency Support Facilities

REQUIREMENT

Establish an interim onsite technical support center separate from, but close to, the control room for engineering and management support of reactor operations during an accident. The center shall be large enough for the necessary utility personnel and five NRC personnel, have direct display or callup of plant parameters, and dedicated communications with the control room, the emergency operations center, and the NRC. Provide a description of the permanent technical support center.

Establish an onsite operational support center, separate from but with communications to the control room for use by operations support personnel during an accident.

Designate a near-site emergency operations facility with communications with the plant to provide evaluation of radiation releases and coordination of all onsite and offsite activities during an accident.

These requirements shall be met before fuel loading. See <NUREG-0578>, Sections 2.2.2.b, 2.2.2.c (Reference 4), and letters of September 27 (Reference 23) and November 9, 1979, (Reference 24) and April 25, 1980, (Reference 29).

RESPONSE

The Cleveland Electric Illuminating Company will establish a Technical Support Center (TSC), an Operational Support Center (OSC) and an Emergency Operations Facility (EOF) to satisfy the intent of <NUREG-0696>, "Functional Criteria for Emergency Response Facilities." These support facilities will be completed prior to fuel load.

The TSC will occupy about 5,000 square feet at the 603’-6” elevation of the Service Building. The OSC will be located in Room 599-05 at the 599’-0” elevation of the Control Complex. Communication will be provided with the TSC and Control Rooms. The location and design of a near-site EOF are now in the planning stage.

Further descriptions of these emergency support facilities can be found in Section 7.0, “Emergency Facilities and Equipment” of the PNPP Emergency Plan.

Item No. III.D.1.1

Integrity of Systems Outside Containment
Likely to Contain Radioactive Material

REQUIREMENT

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate leak reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

RESPONSE

The Cleveland Electric Illuminating Company has established a Leakage Surveillance and Preventative Maintenance Program that will be applied to the following systems in the manner summarized below: Low Pressure Core Spray, High Pressure Core Spray, Residual Heat Removal, Reactor Core Isolation Cooling, Feedwater Leakage Control, Combustible Gas Control Hydrogen Analysis, and Postaccident Sampling.

Visual Examination - Water systems will be inspected while the systems are operating and visually checked for leaks. Potential leakage paths include valve, pump and flange seals and test connections. Leakage will be eliminated to the extent practicable; any leakage not eliminated will be measured and compared to an overall water leakage limit.

Leakage Collection Past Boundary Valves - Leakage past valves in branch lines that are potential leakage paths to the atmosphere will be measured. Leakage will be collected downstream of the boundary valves when the system is in a non-secured status (e.g., standby readiness, full operation, test mode). Where it is impractical to measure leakage with the system in a non-secured status, the boundary valves will be removed and bench-tested. Leakage will be compared to an overall water leakage limit.

Radioactivity Grab Sample - While RCIC is in operation using reactor steam, a grab air sample will be taken from the RCIC room to determine if a steam leak exists. An isotopic analysis shall be performed to determine if steam leakage exists when the grab sample exceeds the permissible airborne activity level. Steam leaks will be identified and eliminated.

Leak Detection - The H_2 Analysis System will be pressurized with air or nitrogen to the post-LOCA operating pressure and then inspected. Leaks will be identified with an ultrasonic leak detector or equivalent leak detection method (e.g., bubble test). Leaks will be eliminated.

Heat Exchangers - Heat exchangers located outside the containment that are associated with the potentially contaminated systems identified above will be included in the leakage surveillance program. Leakage from the potentially contaminated side of the heat exchanger will be eliminated to the extent practicable. Leakage not eliminated will be measured and compared to an overall water leakage limit.

A report describing implementation of this program, along with initial test results, will be submitted to the NRC prior to achieving 100 percent power.

Item No. III.D.3.3

Improved Inplant Iodine Instrumentation
Under Accident Conditions

REQUIREMENT

a. Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

b. Each applicant for a fuel loading license to be issued prior to January 1, 1981, shall provide the equipment, training and procedures necessary to accurately determine the presence of airborne radio-iodine in areas within the plant where plant personnel may be present during an accident.

RESPONSE

Fixed continuous air monitors and portable air monitors and air samplers are utilized to determine the concentrations of airborne radioactivity throughout the plant.

The fixed air monitors, described in <Section 12.3.4> provide continuous data to indicate trends throughout the various plant areas. Particulate filters and charcoal cartridges are removed periodically to identify the specific nuclides encountered.

Portable air samplers are used to collect particulate and charcoal grab samples of areas of specific concern, for example, in preparation and conduct of specific work functions, to verify significant indicated changes by one or more fixed air monitors, or periodic air sampling throughout the plant.

In plant iodine analysis under accident conditions is accomplished by collection of iodine samples utilizing Silver Zeolite Iodine Sampling cartridges. The cartridges and filters are analyzed by gamma spectroscopy using computer analysis techniques.

An Emergency Plan implementing procedure will be prepared to address sampling and appropriate personnel (Radiation Monitoring team members and shift radiation protection technicians) will be trained in these procedures.

USAR <Table 12.5-4> lists the quantities of air samplers available. USAR <Table 12.3-10> lists the Airborne Radiation Monitors.

Item No. III.D.3.4

Control-Room Habitability Requirements

REQUIREMENT

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room", "General Design Criteria for Nuclear Power Plants," of <10 CFR 50, Appendix A>).

RESPONSE

This requirement has been met for PNPP as detailed within the FSAR. Section 6.4 fully describes the control HVAC system layout and functional design including protection of the control room from toxic and radioactive gases. Subsection 2.2.3 reports the results of the evaluation of potential accidents involving nonradioactive hazardous materials including gaseous fuels, liquified gases, explosives, and toxic chemicals.

<APPENDIX 1B>

PNPP LICENSE COMMITMENTS

<Appendix 1B>

PNPP LICENSE COMMITMENTS

<Appendix 1B> was initially a compilation of the remaining plant specific license commitments for the Perry Nuclear Power Plant required to resolve the remaining license conditions in the SER and its supplements. The information contained in this appendix provides CEI's commitments which form the basis for resolving those issues which are a condition for issuance of an operating license and ensuring that the NRC requirements for other longer term issues are met during plant operation. Commitments added after issuance of the operating license are commitments which the NRC has specifically requested to be documented in the USAR.

No changes may be made to the commitments in this appendix unless prior approval is obtained from the NRC. The NRC documents which provide such approval shall be referenced in explanatory paragraphs following the commitment. Changes which add information to update the status of actions relating to a commitment or which document closure of identified items may be made without prior NRC approval.

License Commitments

1. Permanent Slope Protection System   SER - 2.5.5

   CEI shall finalize the design of a permanent slope protection system (as described in FSAR Section 2.4.5.5.3) if the toe or crest of the 3 H:1 V bluff of the Lake Erie shoreline erodes and encroaches closer than 250 ft or 115 ft, respectively, to the emergency service water pumphouse. Construction of the permanent slope protection system shall be completed before the toe of the bluff recedes to within 204 ft of the emergency service water pumphouse.

2. Shift Operating Experience   SSER 6 - 13.1.2.3

   CEI shall have a licensed senior operator on each operating shift, who has had at least 6 months of hot operating experience at a large commercial BWR, including at least 6 weeks of experience at power levels greater than 20% of full-rated thermal power, and who has had BWR startup and shutdown experience, for a period of 1 year from fuel loading or until the attainment of 100% rated thermal power level, whichever occurs later.

   This commitment has been satisfied. CEI notified the NRC of the fulfillment of this commitment in a letter dated July 1, 1987 (PY-CEI/NRR-0679L).

3. Post-Fuel Loading Initial Test Program   SSER 3 - 14 (TMI I.G.1)

   CEI shall complete the Initial Test Program (ITP), set forth in Chapter 14 of the FSAR. Any changes to this program shall be made in accordance with the provisions of <10 CFR 50.59> and shall be reported in accordance with <10 CFR 50.59(b)> within one month.

This commitment has been satisfied. CEI completed the Initial Test Program, as set forth in Chapter 14 of the FSAR, in November 1987. Changes made were in accordance with the provisions of <10 CFR 50.59> and were reported in accordance with <10 CFR 59.59(b)> within one month.

4. Turbine System Maintenance Program SSER 3 - 3.5.1.3.3

Within 3 years from the date of the operating license, CEI shall submit a turbine system maintenance program based on the turbine manufacturer's calculations of missile generation probabilities. Prior to review and approval of that program by the NRC staff, CEI shall volumetrically inspect all low-pressure rotors at the second refueling outage and every alternate outage thereafter, and conduct turbine steam valve maintenance in accordance with staff's recommendations.

This commitment has been satisfied. CEI submitted the turbine system maintenance program in a letter dated March 20, 1989 (PY-CEI/NRR-0977L). This submittal included turbine missile generation probabilities and provided appropriate volumetric measurement intervals based upon these probabilities.

The PNPP turbine system maintenance program was approved by NRC in their letter dated August 23, 1989 (PY-NRR/CEI-0478L).

The current turbine system inspection and maintenance program is described in USAR <Section 10.2.3.6>.

5. Containment Purge Criteria SSER 4 - 6.2.4

CEI shall administer the three programs described in letters dated February 19, 1985, and March 26, 1985, to assess the need for use of the containment purge system, and to minimize its use consistent with ALARA guidelines. These are the data gathering and containment access management programs and interim guidelines for containment purge operation. Based upon the results of these three programs, purge criteria to be used for the remainder of plant life shall be proposed to the NRC prior to startup from the first refueling outage.

This commitment has been satisfied. CEI submitted the required purge criteria by letter dated June 30, 1989 (PY-CEI/NRR-1025L). In this letter, CEI proposed a purge limit of 2,000 hrs/year. NRC responded by letter dated July 18, 1989, limiting purge operation to 1,000 hours/year. CEI may reapply for additional time if conditions warrant.

Requirement deleted by Amendment 69 in the Technical Specifications. Refer to SER for Amendment 69, page 104, item 38.

6. Inservice Inspection Program SSER 4 - 6.6.3

CEI shall submit the initial inservice inspection program required by <10 CFR 50.55(a)> for NRC staff review and approval within one year from the date of the operating license.

This commitment has been satisfied. CEI submitted the 10-Year Inservice Inspection Program Plan in a letter dated March 31, 1987 (PY-CEI/NRR-0614L).

7. <Regulatory Guide 1.97>, Revision 2 SSER 6 - 7.5.2.2

CEI shall implement applicable modifications which are consistent with the conclusions of Topical Report NEDO-31558 "Requirements for Postaccident Neutron Monitoring System" and which are based upon the NRC staff's safety evaluation of the report, on a schedule to be provided 6 months after receipt of the NRC staff SER or prior to startup following the second refueling outage, whichever is sooner.

This commitment has been satisfied. The above commitment is a revision of the original commitment. This revision was proposed by CEI letter PY-CEI/NRR-0969L dated 3/3/89, and was approved by the NRC (with the addition of the second refueling outage clause) by their letter dated 7/14/89. At that point in the evolution of this issue, CEI had also agreed to continue to follow the development/progress of postaccident neutron flux monitoring technology which meets the Category 1 requirements of <Regulatory Guide 1.97>, Revision 2.

On January 29, 1990, the NRC staff issued an SER which found the NEDO-31558 functional criteria to be unacceptable. Issuance of the SER resulted in CEI submittal of a schedule for neutron monitoring modifications in a letter dated July 27, 1990 (PY-CEI/NRR-1203L). However, the letter noted that the schedule was dependent on NRC resolution of three issues; (1) resolution of neutron monitoring system design criteria, to clearly define the requirements for an acceptable system, (2) issuance of a favorable NRC SER on the incore neutron monitoring system which is described by NEDO-31439 (Note: the NRC staff issued a favorable SER on the incore neutron monitoring system on October 3, 1990), and (3) resolution of the BWROG appeal of the NRC's 1/29/90 SER that rejected the NEDO-31558 alternatives. THE BWROG filed their appeal on August 20, 1990.

By letter dated October 14, 1992, the Director of NRR, Dr. Murley, ruled on the BWROG appeal. He informed the BWROG that Category 1 neutron flux monitoring instrumentation is not needed for currently designed BWRs to cope with loss-of-coolant accidents (LOCA), anticipated transients without scram (ATWS), or other accidents that do not result in severe core damage conditions. He further concluded that instruments to monitor the progression of core melt accidents are best addressed by the severe accident management program. Based on Dr. Murley's decision, the NRC staff then proceeded to issue a Safety Evaluation that found NEDO-31558 to be

acceptable. This Safety Evaluation was forwarded to the BWROG by letter from Bruce A. Boger to C. L. Tully dated January 13, 1993.

A Perry Nuclear Power Plant-specific review of the NEDO-31558 report was provided to the NRC on February 7, 1994 (PY-CEI/NRR-1669L). The reviews concluded that the PNPP design for neutron monitoring were consistent with the NEDO-31558 guidance, and no modifications were identified as being necessary.

The NRC closed out this issue for PNPP in a letter dated February 23, 1994 (PY-NRR/CEI-0685L).

8. Detailed Control Room Design Review SSER 7/8 - 18.2

(a) Prior to exceeding 5% of thermal-rated power, CEI shall provide, for NRC staff review and approval, the results of the communication equipment and preliminary sound surveys in the control room and at the remote shutdown panel, the results of the augmented process for verifying that improvements do not introduce new human engineering discrepancies, and implementation schedules for correcting human engineering discrepancies in accordance with commitments made in CEI letters (PY-CEI/NRR-0357L) dated October 2, 1985, (PY-CEI/NRR-0373L) dated October 14, 1985, and (PY-CEI/NRR-0379L) dated October 21, 1985.

This portion of the commitment has been satisfied. CEI submitted the Summary Report for the DCRDR, Supplements 1 and 2 and additional information to the NRC. In Supplement No. 10 to the PNPP SER, the NRC staff concluded that the DCRDR was sufficiently complete to allow full power licensing of Unit 1.

(b) Before start of the 100-hour warranty run, CEI shall implement corrections to human engineering discrepancies per commitments in Supplement 2 to the Detailed Control Room Design Review Summary Report, dated May 28, 1986, and in a letter from M. R. Edelman to W. R. Butler, dated August 26, 1986.

This portion of the commitment has been satisfied. This portion of the commitment was created by the NRC in the full-power Operating License (NPF-58) dated November 13, 1986. CEI notified NRC of the completion of these HED's in a letter dated October 12, 1987 (PY-CEI/NRR-0728L).

(c) Before startup following the first refueling outage, CEI shall implement corrections to human engineering discrepancies per commitments in:

(1) The Detailed Control Room Design Review Summary Report, dated January 10, 1985.

(2) Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 14, 1985.

(3) Revision 1 to Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 21, 1985.

(4) Supplement 2 to the Detailed Control Room Design Review Summary Report, dated May 28, 1986.

(5) The Control Room Validation Summary Report, dated July 11, 1986.

(6) Errata sheets to Supplement 2 to the Detailed Control Room Review Summary Report, attached to Letter PY-CEI/NRR-0510L, dated July 29, 1986.

(7) Detailed Control Room Design Review - First Refuel HED Revisions Report, attached to Letter PY-CEI/NRR-0946L, dated February 10, 1989.

Before startup following the first refueling outage, CEI shall also provide results of the final sound surveys in the control room and at the remote shutdown facilities for NRC review per the commitment in Supplement 1 to the Detailed Control Room Design Review Summary Report, dated October 14, 1985.

This portion of the commitment has been satisfied. The commitment underwent two revisions prior to closure. The first revision was in the issuance of the full-power Operating License (NPF-58) dated November 13, 1986. The second revision was made in Amendment 23 to the Operating License dated July 6, 1989. CEI notified NRC of the completion of this portion of the commitment in a letter dated July 11, 1989 (PY-CEI/NRR-1031L). This letter confirmed completion of the first refueling outage HED's, and provided the results of the final sound survey.

(d) Before startup following the second refueling outage, CEI shall complete the augmented verification of human engineering discrepancy corrections implemented after full-power licensing per the commitment in Supplement 2 to the Detailed Control Room Design Review Summary Report, dated May 28, 1986. CEI shall also correct any problems identified by the augmented verification before startup following the second refueling outage per the commitment in a letter from M. R. Edelman to W. R. Butler, dated August 26, 1986.

This portion of the commitment has been satisfied. This portion of the commitment was revised once, with the issuance of the full-power Operating License (NPF-58) dated November 13, 1986. In addition, a CEI letter dated July 11, 1989 (PY-CEI/NRR-1031L) identified a new HED (HED-617) which was also to be completed prior to startup following the second refueling outage.

CEI notified NRC of the completion of this portion of the commitment in a letter dated November 30, 1990 (PY-CEI/NRR-1260L). This letter confirmed completion of the augmented verifications of the HED's implemented after full-power licensing, and of the completion (and re-verification) of several items identified during the augmented verification process. The letter also confirmed completion and augmented verification of HED-617.

Additionally, CEI shall complete the validation of the Perry Nuclear Power Plant emergency instructions and issue a Summary Report prior to achieving initial criticality.

This portion of the commitment has been satisfied. CEI submitted the Summary Report for the DCRDR and several additional documents to the NRC. In Supplement No. 8 to the PNPP SER, the NRC concluded that this element of the DCRDR as it relates to the Perry Emergency Instructions is satisfied.

9. Emergency Planning SSER 7 - 13.3

(a) Prior to exceeding 5% of rated thermal power, CEI shall obtain letters of agreement from all school districts for the supply of buses for evacuation purposes.

This portion of the commitment has been satisfied. In Supplement No. 10 to the PNPP SER, the NRC concluded that CEI has complied with this commitment.

(b) Prior to exceeding 5% of rated thermal power, CEI shall verify that the training of fire personnel in radiological monitoring and decontamination procedures has been completed and CEI shall verify that necessary decontamination equipment has been provided at the fire department facilities for each reception center.

This portion of the commitment has been satisfied. In Supplement No. 10 to the PNPP SER, the NRC concluded that CEI has complied with this commitment.

10. TDI Diesel Engines SSER 8 - 9.6.3

CEI shall comply with the following requirements related to the TDI diesel engines:

(a) Changes to the maintenance and surveillance program for the TDI diesel engines, as identified and approved by the NRC staff in the supplemental safety evaluation report in the letter dated July 8, 1986, shall be subject to the provisions of <10 CFR 50.59>.

This portion of the commitment has been revised once per Amendment 24 to the Operating License dated September 15, 1989.

This portion of the commitment deleted by Amendment 74 to the Operating License dated November 16, 1995.

(b) Crankshafts shall be inspected as follows:

The oil holes and fillets of the three main bearing journals subject to the highest torsional stresses (Nos. 4, 6, 8) shall be examined with fluorescent liquid penetrant and, as necessary, eddy current, during each 5 year major disassembly. The same inspections on oil holes and fillets shall be performed on at least three crankpin journals between Journals 3 and 8.

This portion of the commitment deleted by Amendment 74 to the Operating License dated November 16, 1995.

(c) Cylinder blocks shall be inspected at intervals calculated using the cumulative damage index (CDI) model and using inspection methodologies described by Failure Analysis Associates, Inc., (FaAA) in report entitled "Design Review of TDI R-4 Series Emergency Diesel Generator Cylinder Blocks" (FaAA-84-9-11) dated December 1984. Liquid penetrant inspection of the cylinder liner landing area should be performed anytime liners are removed.

This portion of the commitment deleted by Amendment 74 to the Operating License dated November 16, 1995.

(d) The following air roll tests shall be performed as specified below, except that air rolls shall not be performed on an operable TDI Standby Diesel if the other TDI Standby Diesel is already inoperable.

The engines shall be rolled over with the airstart system and the cylinder stopcocks open prior to planned starts, unless that start occurs within 4 hours of a shutdown. The engines shall also be rolled over the the airstart system and the cylinder stopcocks open after 4 hours, but no more than 8 hours after engine shutdown and then rolled over once again approximately 24 hours after each shutdown. In the event an engine is removed from service for any reason other than the rolling over procedure prior to expiration of the 8 hour or 24 hour periods noted above, that engine need not be rolled over while it is out-of-service. The licensee shall air roll the engine over with the stopcocks open at the time it is returned to service. The origin of any water detected in the cylinders must be determined and any cylinder head which leaks due to a crack shall be replaced. No cylinder heads that contain a through-wall weld repair where the repair was performed from one side only shall be used on the engines.

This portion of the commitment has been revised once per Amendment 24 to the Operating License (NPF-58) dated September 15, 1989.

This portion of the commitment deleted by Amendment 74 to the Operating License dated November 16, 1995.

(e) If inspection of either TDI generator reveals cracks in the crankshaft or in the cylinder block between stud holes of adjacent cylinders, this condition shall be reported promptly to the NRC staff and the affected engine(s) shall be considered inoperable. The engine(s) shall not be restored to "operable" status until the proposed disposition and/or corrective actions have been approved by the NRC staff.

This portion of the commitment deleted by Amendment 74 to the Operating License dated November 16, 1995.

(f) Operating beyond the first refueling outage shall require staff approval based on the staff's final review of the Owners Group generic findings and of the overall implementation status of Owners Group recommendations at Perry.

This portion of the commitment has been removed. Staff approval for operation beyond the first refueling outage was obtained on July 8, 1986 with the issuance of NRC's "Safety Evaluation Report Re the Operability/Reliability of the Emergency Diesel Generators manufactured by Transamerica Delaval, Inc. - Perry Nuclear Power Plant (Unit 1 and Unit 2)". The cover letter transmitting this SER removed this license commitment. SSER 10 further stated that this item was "deleted in its entirety". The Perry Full Power Operating License issued November 13, 1986 reflects the removal of this license commitment.

The special requirements of the Design Review/Quality Revalidation (DR/QR) Program imposed on the TDI Diesel Generators were deleted by Amendment 74 to the PNPP Operating License, dated November 16, 1995. The Cooper Enterprise Preventative Maintenance Plan provides the basis for the PNPP diesel generator maintenance program.

11. Hydrogen Control for Degraded Core Accidents SSER 7 - 6.2.7 (TMI II.B.8)

Prior to exceeding 5% rated thermal power, CEI shall have made a further confirmatory analysis of equipment in the containment that has not been qualified for pressure survivability, or has narrow margins of pressure survivability: these are the containment vacuum breakers, hydrogen mixing compressor and discharge check valves.

This portion of the commitment has been satisfied. CEI has provided further confirmatory analysis to the NRC. In Supplement No. 10 to the PNPP SER, the NRC concluded that the analysis was an acceptable response to this license commitment.

Prior to exceeding 5% rated thermal power, CEI shall ensure that written procedures are available for operation of the hydrogen igniter system.

This portion of the commitment has been satisfied. CEI has developed a Emergency Operating Procedure (EOP) that provides plant operators with guidance on the use of the hydrogen igniter system. This information was provided to the NRC. In Supplement No. 10 to the PNPP SER, the NRC concluded that this information satisfies the license commitment.

12. Emergency Containment Venting SSER 8 - 13.5.2.2

CEI shall submit the plant unique analysis and resulting venting pressure value for the Perry facility prior to operation above 5% power. Sufficient justification for the selected emergency vent paths and the effects of emergency venting shall also be provided.

This commitment has been satisfied. CEI provided the NRC with plant-specific analysis and the resulting venting pressure value. In Supplement No. 10 to the Perry SER, the NRC concluded that CEI's response satisfies the license commitment.

13. Reactor Internals Vibration Test Program SSER 4 - 3.9.2.3

CEI shall submit a final report, summarizing the results of the prototype reactor internals test program vibration analyses, measurements and inspection programs, within 180 days of completion of vibration testing per <Regulatory Guide 1.20>.

This commitment has been satisfied. CEI letters dated January 15, 1988 (PY-CEI/NRR-0771L) and May 11, 1988 (PY-CEI/NRR-0842L) provided preliminary and final reports respectively, which included the results of the PNPP reactor internal vibration analyses, measurements and inspection programs.

A CEI letter dated January 30, 1991 (PY-CEI/NRR-1288L) provided an addendum to the final report, which described the results of a fatigue analysis of the in-core guide tubes during single recirculation loop operation.

14. Instrument Setpoint Methodology SSER 7 - 7.2.2.8

Six months after receipt of an NRC Safety Evaluation Report on GE Instrument Setpoint Methodology (NEDC-31336), and subject to any stipulations therein, CEI shall provide for NRC staff review and approval, a detailed technical report documenting the basis and methodology for establishing protection system trip setpoints and allowable values, based on the Instrument Setpoint Methodology Group (ISMG) effort, as discussed in CEI letter dated October 17, 1985 (PY-CEI/NRR-0368L).

The above commitment is a revision of the original commitment. This revision was proposed by CEI letter PY-CEI/NRR-0969L dated 3/3/89, and was approved by the NRC (with one minor change) by their letter dated 7/14/89.

This commitment has been satisfied. CEI submitted the instrument setpoint methodology report in a letter dated October 15, 1993 (PY-CEI/NRR-1706L). The NRC responded to this letter on July 18, 1995 (Letter from Jon B. Hopkins to Mr. Donald C. Shelton) concluding that PNPP satisfies the commitment contained in this item to document the basis and methodology for establishing protection system trip setpoints and allowable values.

15. Leak Reduction Program   SER - 11.5 (TMI III.D.1.1)

CEI shall implement the PNPP leak reduction program described in letters to the NRC dated May 29, 1985, (PY-CEI/NRR-0237L) and September 24, 1985, (PY/CEI/NRR-0349L) which includes the high pressure and low pressure core spray, residual heat removal, reactor core isolation cooling, feedwater leakage control, combustible gas control hydrogen analysis, and postaccident sampling systems. CEI shall provide the NRC with leakage data from the various systems and components in the program prior to reaching full power operation.

This commitment has been satisfied. CEI supplied the results of the PNPP leak reduction program to the NRC in a letter dated June 28, 1987, (PY-CEI/NRR-0676L), prior to reaching full power operation.

NRC responded to this letter on September 20, 1988 (Letter from T. G. Colburn to Mr. A. Kaplan) stating that CEI has "adequately met the acceptance criteria for this item and therefore, we (NRC) consider this item closed for (Perry)."

16. Gaseous Effluent Sampling System Representative Sample   SSER 8 - 11.5

Work is underway at the Pacific Northwest Laboratory of DOE under an NRR technical assistance contract to develop definitive guidance on making sampling line loss measurements. CEI shall perform such measurements on the radioiodine and particulate sampling system on a schedule to be determined after NRC staff guidance is provided on the method for determining line loss, if the NRC staff concludes that such measurements are necessary for these systems.

The above commitment is a revision of the original commitment. This revision was proposed by CEI letter PY-CEI/NRR-0969L dated 3/3/89, and was approved by the NRC (with one change) by their letter dated 7/14/89.

17. Silicone Sealant   SSER 8 - 6.5

Within one year of obtaining the operating license, CEI shall provide the results of its program for qualification of silicone sealant in conjunction with ductwork utilized, or shall propose other measures to assure the integrity of the external portions of the control room emergency recirculation system to allow NRC adequate time for review prior to the first refueling outage.

This commitment has been satisfied. CEI provided the results of its qualification testing on silicone sealants to the NRC in a letter dated July 30, 1986 (PY-CEI/NRR-0505L). A meeting with the staff was held on August 11, 1987 to discuss the results of the qualification testing. Several commitments were made in this meeting and were provided to the NRC formally by letter dated August 27, 1987 (PY-CEI/NRR-0703L).

NRC issued an SER dated October 9, 1987 which found the use of silicone sealant at PNPP to be acceptable.

18. Qualitative Assessment of Drywell Bypass Leak Tightness - PY-CEI/NRR-2119L

At least once per operating cycle, a qualitative assessment of drywell bypass leak tightness will be performed, unless the Technical Specification Drywell Bypass Leak Rate Test will be performed in its place. At a minimum, this assessment will be performed during refueling outages, following completion of work on the drywell structure or penetrations. The assessment will involve verifying that a differential pressure can be established between the drywell and the containment. Although the assessment is not as comprehensive as the Technical Specification Drywell Bypass Leak Rate Test, it will provide reasonable assurance of the ability of the drywell to perform its design basis function. Refer to Letters PY-CEI/NRR-2007L and PY-CEI/NRR-2119L which contain a Regulatory Commitment for the performance of this assessment <L02323>.