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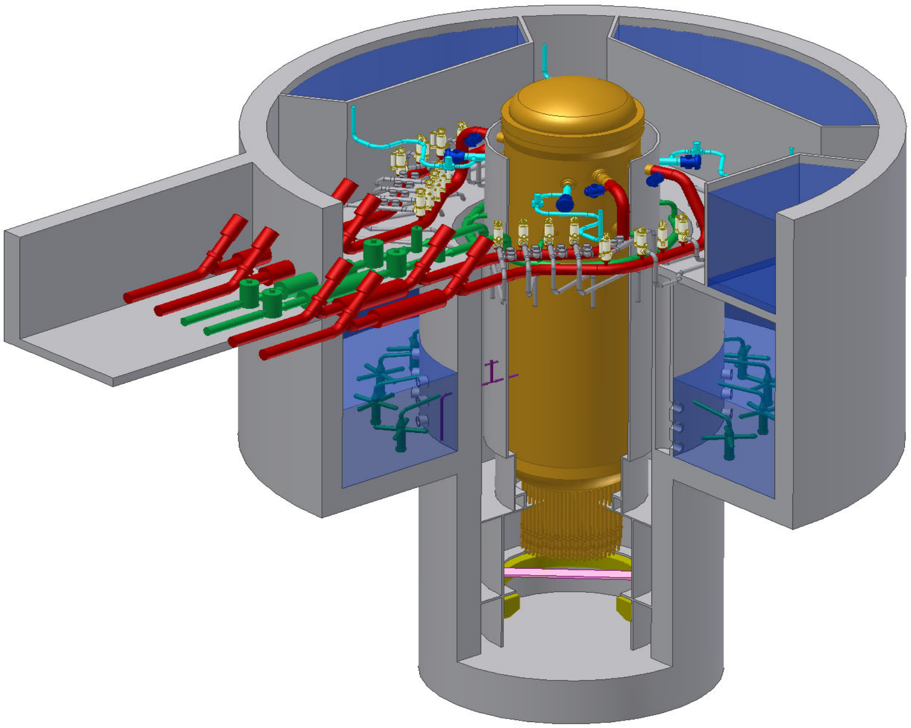

ESBWR Design Control Document

Tier 2

Chapter 18

Human Factors Engineering

(Conditional Release - pending
closure of design verifications)

Contents

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Global Abbreviations And Acronyms List

Term	Definition
10 CFR	Title 10, Code of Federal Regulations
A/D	Analog-to-Digital
AASHTO	American Association of Highway and Transportation Officials
AB	Auxiliary Boiler
ABS	Auxiliary Boiler System
ABWR	Advanced Boiling Water Reactor
ac / AC	Alternating Current
AC	Air Conditioning
ACF	Automatic Control Function
ACI	American Concrete Institute
ACS	Atmospheric Control System
AD	Administration Building
ADS	Automatic Depressurization System
AEC	Atomic Energy Commission
AFIP	Automated Fixed In-Core Probe
AGMA	American Gear Manufacturer's Association
AHS	Auxiliary Heat Sink
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
AL	Analytical Limit
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
API	American Petroleum Institute
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
APR	Automatic Power Regulator
APRS	Automatic Power Regulator System
ARI	Alternate Rod Insertion
ARMS	Area Radiation Monitoring System
ASA	American Standards Association
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
ASTM	American Society of Testing Methods

Term	Definition
AT	Unit Auxiliary Transformer
ATLM	Automated Thermal Limit Monitor
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
AWS	American Welding Society
AWWA	American Water Works Association
B&PV	Boiler and Pressure Vessel
BAF	Bottom of Active Fuel
BHP	Brake Horse Power
BOP	Balance of Plant
BPU	Bypass Unit
BPWS	Banked Position Withdrawal Sequence
BRE	Battery Room Exhaust
BRL	Background Radiation Level
BTP	NRC Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAV	Cumulative absolute velocity
C&FS	Condensate and Feedwater System
C&I	Control and Instrumentation
C/C	Cooling and Cleanup
CB	Control Building
CBHVAC	Control Building HVAC
CCI	Core-Concrete Interaction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CIRC	Circulating Water System
CIS	Containment Inerting System
CIV	Combined Intermediate Valve
CLAVS	Clean Area Ventilation Subsystem of Reactor Building HVAC
CM	Cold Machine Shop
CMS	Containment Monitoring System
CMU	Control Room Multiplexing Unit
COL	Combined Operating License
COLR	Core Operating Limits Report
CONAVS	Controlled Area Ventilation Subsystem of Reactor Building HVAC
CPR	Critical Power Ratio
CPS	Condensate Purification System
CPU	Central Processing Unit

Term	Definition
CR	Control Rod
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CRHA	Control Room Habitability Area
CRT	Cathode Ray Tube
CS&TS	Condensate Storage and Transfer System
CSDM	Cold Shutdown Margin
CS / CST	Condensate Storage Tank
CT	Main Cooling Tower
CTVCF	Constant Voltage Constant Frequency
CUF	Cumulative usage factor
CWS	Chilled Water System
D-RAP	Design Reliability Assurance Program
DAC	Design Acceptance Criteria
DAW	Dry Active Waste
DBA	Design Basis Accident
dc / DC	Direct Current
DCS	Drywell Cooling System
DCIS	Distributed Control and Information System
DEPSS	Drywell Equipment and Pipe Support Structure
DF	Decontamination Factor
D/F	Diaphragm Floor
DG	Diesel-Generator
DHR	Decay Heat Removal
DM&C	Digital Measurement and Control
DOF	Degree of freedom
DOI	Dedicated Operators Interface
DOT	Department of Transportation
dPT	Differential Pressure Transmitter
DPS	Diverse Protection System
DPV	Depressurization Valve
DR&T	Design Review and Testing
DTM	Digital Trip Module
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC

Term	Definition
ECCS	Emergency Core Cooling System
E-DCIS	Essential DCIS (Distributed Control and Information System)
EDO	Environmental Qualification Document
EFDS	Equipment and Floor Drainage System
EFPY	Effective full power years
EHC	Electrohydraulic Control (Pressure Regulator)
ENS	Emergency Notification System
EOC	Emergency Operations Center
EOC	End of Cycle
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedures
EPDS	Electric Power Distribution System
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
ESF	Engineered Safety Feature
ETS	Emergency Trip System
FAC	Flow-Accelerated Corrosion
FAPCS	Fuel and Auxiliary Pools Cooling System
FATT	Fracture Appearance Transition Temperature
FB	Fuel Building
FBHV	Fuel Building HVAC
FCI	Fuel-Coolant Interaction
FCM	File Control Module
FCS	Flammability Control System
FCU	Fan Cooling Unit
FDDI	Fiber Distributed Data Interface
FFT	Fast Fourier Transform
FFWTR	Final Feedwater Temperature Reduction
FHA	Fire Hazards Analysis
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FO	Diesel Fuel Oil Storage Tank
FOAKE	First-of-a-Kind Engineering
FPE	Fire Pump Enclosure
FTDC	Fault-Tolerant Digital Controller

Term	Definition
FTS	Fuel Transfer System
FW	Feedwater
FWCS	Feedwater Control System
FWS	Fire Water Storage Tank
GCS	Generator Cooling System
GDC	General Design Criteria
GDCS	Gravity-Driven Cooling System
GE	General Electric Company
GE-NE	GE Nuclear Energy
GEN	Main Generator System
GETAB	General Electric Thermal Analysis Basis
GL	Generic Letter
GM	Geiger-Mueller Counter
GM-B	Beta-Sensitive GM Detector
GSIC	Gamma-Sensitive Ion Chamber
GSOS	Generator Sealing Oil System
GWSR	Ganged Withdrawal Sequence Restriction
HAZ	Heat-Affected Zone
HCU	Hydraulic Control Unit
HCW	High Conductivity Waste
HDVS	Heater Drain and Vent System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HEP	Human error probability
HEPA	High Efficiency Particulate Air/Absolute
HFE	Human Factors Engineering
HFF	Hollow Fiber Filter
HGCS	Hydrogen Gas Cooling System
HIC	High Integrity Container
HID	High Intensity Discharge
HIS	Hydraulic Institute Standards
HM	Hot Machine Shop & Storage
HP	High Pressure
HPNSS	High Pressure Nitrogen Supply System
HPT	High-pressure turbine
HRA	Human Reliability Assessment
HSI	Human-System Interface
HSSS	Hardware/Software System Specification
HVAC	Heating, Ventilation and Air Conditioning
HVS	High Velocity Separator

Term	Definition
HWCS	Hydrogen Water Chemistry System
HWS	Hot Water System
HX	Heat Exchanger
I&C	Instrumentation and Control
I/O	Input/Output
IAS	Instrument Air System
IASCC	Irradiation Assisted Stress Corrosion Cracking
IBC	International Building Code
IC	Ion Chamber
IC	Isolation Condenser
ICD	Interface Control Diagram
ICS	Isolation Condenser System
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
IED	Instrument and Electrical Diagram
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IIS	Iron Injection System
ILRT	Integrated Leak Rate Test
IOP	Integrated Operating Procedure
IMC	Induction Motor Controller
IMCC	Induction Motor Controller Cabinet
IRM	Intermediate Range Monitor
ISA	Instrument Society of America
ISI	In-Service Inspection
ISLT	In-Service Leak Test
ISM	Independent Support Motion
ISMA	Independent Support Motion Response Spectrum Analysis
ISO	International Standards Organization
ITA	Inspections, Tests or Analyses
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria
ITA	Initial Test Program
LAPP	Loss of Alternate Preferred Power
LCO	Limiting Conditions for Operation
LCW	Low Conductivity Waste
LD	Logic Diagram
LDA	Lay down Area
LD&IS	Leak Detection and Isolation System
LERF	Large early release frequency
LFCV	Low Flow Control Valve

Term	Definition
LHGR	Linear Heat Generation Rate
LLRT	Local Leak Rate Test
LMU	Local Multiplexer Unit
LO	Dirty/Clean Lube Oil Storage Tank
LOCA	Loss-of-Coolant-Accident
LOFW	Loss-of-feedwater
LOOP	Loss of Offsite Power
LOPP	Loss of Preferred Power
LP	Low Pressure
LPCI	Low Pressure Coolant Injection
LPCRD	Locking Piston Control Rod Drive
LPMS	Loose Parts Monitoring System
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LWMS	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MAPLHGR	Maximum Average Planar Linear Head Generation Rate
MAPRAT	Maximum Average Planar Ratio
MBB	Motor Built-In Brake
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRP	Main Control Room Panel
MELB	Moderate Energy Line Break
MLHGR	Maximum Linear Heat Generation Rate
MMI	Man-Machine Interface
MMIS	Man-Machine Interface Systems
MOV	Motor-Operated Valve
MPC	Maximum Permissible Concentration
MPL	Master Parts List
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSL	Main Steamline
MSLB	Main Steamline Break
MSLBA	Main Steamline Break Accident
MSR	Moisture Separator Reheater
MSV	Mean Square Voltage
MT	Main Transformer
MTTR	Mean Time To Repair

Term	Definition
MWS	Makeup Water System
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NCIG	Nuclear Construction Issues Group
NDE	Nondestructive Examination
NE-DCIS	Non-Essential Distributed Control and Information System
NDRC	National Defense Research Committee
NDT	Nil Ductility Temperature
NFPA	National Fire Protection Association
NIST	National Institute of Standard Technology
NMS	Neutron Monitoring System
NOV	Nitrogen Operated Valve
NPHS	Normal Power Heat Sink
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHX	Non-Regenerative Heat Exchanger
NS	Non-seismic (non-seismic Category I)
NSSS	Nuclear Steam Supply System
NT	Nitrogen Storage Tank
NTSP	Nominal Trip Setpoint
O&M	Operation and Maintenance
O-RAP	Operational Reliability Assurance Program
OBCV	Overboard Control Valve
OBE	Operating Basis Earthquake
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System
OIS	Oxygen Injection System
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLU	Output Logic Unit
OOS	Out-of-service
ORNL	Oak Ridge National Laboratory
OSC	Operational Support Center
OSHA	Occupational Safety and Health Administration
OSI	Open Systems Interconnect
P&ID	Piping and Instrumentation Diagram
PA/PL	Page/Party-Line
PABX	Private Automatic Branch (Telephone) Exchange
PAM	Post Accident Monitoring
PAR	Passive Autocatalytic Recombiner
PAS	Plant Automation System

Term	Definition
PASS	Post Accident Sampling Subsystem of Containment Monitoring System
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PCT	Peak cladding temperature
PCV	Primary Containment Vessel
PFD	Process Flow Diagram
PGA	Peak Ground Acceleration
PGCS	Power Generation and Control Subsystem of Plant Automation System
PH	Pump House
PL	Parking Lot
PM	Preventive Maintenance
PMCS	Performance Monitoring and Control Subsystem of NE-DCIS
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PQCL	Product Quality Check List
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PRNM	Power Range Neutron Monitoring
PS	Plant Stack
PSD	Power Spectra Density
PSS	Process Sampling System
PSWS	Plant Service Water System
PT	Pressure Transmitter
PWR	Pressurized Water Reactor
QA	Quality Assurance
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RBC	Rod Brake Controller
RBCC	Rod Brake Controller Cabinet
RBCWS	Reactor Building Chilled Water Subsystem
RBHV	Reactor Building HVAC
RBS	Rod Block Setpoint
RBV	Reactor Building Vibration
RC&IS	Rod Control and Information System
RCC	Remote Communication Cabinet
RCCV	Reinforced Concrete Containment Vessel
RCCWS	Reactor Component Cooling Water System

Term	Definition
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RDA	Rod Drop Accident
RDC	Resolver-to-Digital Converter
REPAVS	Refueling and Pool Area Ventilation Subsystem of Fuel Building HVAC
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	residual heat removal (function)
RHX	Regenerative Heat Exchanger
RMS	Root Mean Square
RMS	Radiation Monitoring Subsystem
RMU	Remote Multiplexer Unit
RO	Reverse Osmosis
ROM	Read-only Memory
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RRPS	Reference Rod Pull Sequence
RSM	Rod Server Module
RSPC	Rod Server Processing Channel
RSS	Remote Shutdown System
RSSM	Reed Switch Sensor Module
RSW	Reactor Shield Wall
RTIF	Reactor Trip and Isolation Function(s)
RT_{NDT}	Reference Temperature of Nil-Ductility Transition
RTP	Reactor Thermal Power
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SA	Severe Accident
SAG	Safety Analysis Guidelines
SAR	Safety Analysis Report
SB	Service Building
S/C	Digital Gamma-Sensitive GM Detector
SC	Suppression Chamber
S/D	Scintillation Detector
S/DRSRO	Single/Dual Rod Sequence Restriction Override
S/N	Signal-to-Noise
S/P	Suppression Pool
SAS	Service Air System

Term	Definition
SB&PC	Steam Bypass and Pressure Control System
SBO	Station Blackout
SBWR	Simplified Boiling Water Reactor
SCEW	System Component Evaluation Work
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDS	System Design Specification
SEOA	Sealed Emergency Operating Area
SER	Safety Evaluation Report
SF	Service Water Building
SFP	Spent fuel pool
SIL	Service Information Letter
SIT	Structural Integrity Test
SIU	Signal Interface Unit
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SMU	SSLC Multiplexing Unit
SOV	Solenoid Operated Valve
SP	Setpoint
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System
SPTMS	Suppression Pool Temperature Monitoring Subsystem of Containment Monitoring System
SQVs	Squib Valves
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRS	Software Requirements Specification
SRSRO	Single Rod Sequence Restriction Override
SRSS	Sum of the squares
SRV	Safety Relief Valve
SRVDL	Safety relief valve discharge line
SSAR	Standard Safety Analysis Report
SSC(s)	Structure, System and Component(s)
SSE	Safe Shutdown Earthquake
SSLC	Safety System Logic and Control

Term	Definition
SSPC	Steel Structures Painting Council
ST	Spare Transformer
STP	Sewage Treatment Plant
STRAP	Scram Time Recording and Analysis Panel
STRP	Scram Time Recording Panel
SV	Safety Valve
SWH	Static water head
SWMS	Solid Waste Management System
SY	Switch Yard
TAF	Top of Active Fuel
TASS	Turbine Auxiliary Steam System
TB	Turbine Building
TBCE	Turbine Building Compartment Exhaust
TBE	Turbine Building Exhaust
TBLOE	Turbine Building Lube Oil Area Exhaust
TBS	Turbine Bypass System
TBHV	Turbine Building HVAC
TBV	Turbine Bypass Valve
TC	Training Center
TCCWS	Turbine Component Cooling Water System
TCS	Turbine Control System
TCV	Turbine Control Valve
TDH	Total Developed Head
TEMA	Tubular Exchanger Manufacturers' Association
TFSP	Turbine first stage pressure
TG	Turbine Generator
TGSS	Turbine Gland Seal System
THA	Time-history accelerograph
TLOS	Turbine Lubricating Oil System
TLU	Trip Logic Unit
TMI	Three Mile Island
TMSS	Turbine Main Steam System
TRM	Technical Requirements Manual
TS	Technical Specification(s)
TSC	Technical Support Center
TSI	Turbine Supervisory Instrument
TSV	Turbine Stop Valve
UBC	Uniform Building Code
UHS	ultimate heat sink
UL	Underwriter's Laboratories Inc.

Term	Definition
UPS	Uninterruptible Power Supply
USE	Upper Shelf Energy
USM	Uniform Support Motion
USMA	Uniform support motion response spectrum analysis
USNRC	United States Nuclear Regulatory Commission
USS	United States Standard
UV	Ultraviolet
V&V	Verification and Validation
Vac / VAC	Volts Alternating Current
Vdc / VDC	Volts Direct Current
VDU	Video Display Unit
VW	Vent Wall
VWO	Valves Wide Open
WD	Wash Down Bays
WH	Warehouse
WS	Water Storage
WT	Water Treatment
WW	Wetwell
XMFR	Transformer
ZPA	Zero period acceleration

18. HUMAN FACTORS ENGINEERING

18.1 INTRODUCTION

As discussed in SRP 18R1, this Chapter reviews the Human Factors Engineering (HFE) programs for the ESBWR. SRP 18R1 provides guidance for three types of licensing applications. As discussed in Subsection 1.1.2.2: Per 10 CFR 52.45, this DCD Tier 2 supports the Final Design Approval (FDA) and standard Design Certification (DC) for the ESBWR Standard Plant. In accordance with a standard design certification under Part 52, this Chapter provides technical information, which is technically relevant to the ESBWR design. The technical information includes the HFE program. However, as discussed in SRP 18R1, because technology is continually advancing, details of the HFE design need not be complete before the NRC issuance of a design certification. As such, this presentation under 10 CFR Part 52 primarily focuses on the HFE design process.

This chapter describes the ESBWR Human-System Interface (HSI) design goals and bases, the standard HSI design features and the detailed HSI design and implementation process, with embedded design acceptance criteria, for the ESBWR standard plant operator interface. The ESBWR Emergency Procedure Guidelines (EPGs) and the inventory of instrumentation and controls needed by the control room staff for the performance of Emergency Operating Procedures (EOPs) are also described. The incorporation of Human Factors Engineering (HFE) principles into all phases of the design of these interfaces is provided for as described in this chapter.

Design goals and design bases for the HSI in the main control room and in remote locations are established in Section 18.2. The overall design and implementation process is described in Section 18.3. Section 18.4 contains a description of the main control room standard HSI design features and HSI technologies. The Remote Shutdown System is described in Section 18.5. Section 18.6 discusses how the systems that make up the HSI are integrated together and with the other systems of the plant. Section 18.7 discusses the detailed design implementation process. Section 18.8 discusses the HFE related COL license information requirement. The ESBWR Emergency Procedure Guidelines (EPGs), which provide the basis for human factors evaluations of emergency operations, are contained in Appendix 18A. Appendix 18B discusses the differences between the ESBWR Emergency Procedure Guidelines and the U.S. Boiling Water Reactor Owners Group (BWROG) EPG/SAG Revision 2. The input data and results of calculations performed during the preparation of the ESBWR Emergency Procedure Guidelines are contained in Appendix 18C. Appendix 18D presents a characterization of a main control room HSI equipment implementation that incorporates the ESBWR standard design features discussed in Section 18.4. A general description of the design and implementation process for the ESBWR HSI is presented in Appendix 18E. The inventory and supporting analysis of emergency operation information and controls specific to ESBWR will be developed in COL and captured in Appendix F and H. The design development and validation testing of the standard control room design features equipment and configuration are described in Appendix 18G..

18.2 DESIGN GOALS AND DESIGN BASES

The primary goal for HSI designs is to facilitate safe, efficient and reliable operator performance during all phases of normal plant operation, abnormal events and accident conditions. To achieve this goal, information displays, controls and other interface devices in the control room and other plant areas are designed and are implemented in a manner consistent with good human factors engineering practices. Further, the following specific design bases are adopted:

- During all phases of normal plant operation, abnormal events and emergency conditions, the ESBWR is operable by two reactor operators. In addition, the operating crew will include one assistant control room shift supervisor, one control room shift supervisor, and auxiliary equipment operators as required by task analysis. During accidents, technical assistance is available to the operating crew from personnel in the technical support center. Four licensed operators are on shift at all times, consistent with the staffing requirements of 10 CFR 50.54m. The main control room staff size and roles are evaluated in Subsection 18.8.2.
- The HSI design promotes efficient and reliable operation through expanded application of automated operation capabilities.
- The HSI design utilizes only proven technology.
- Safety-related systems monitoring and control capability is provided in full compliance with pertinent regulations regarding divisional separation and independence.
- The HSI design is highly reliable and provides functional redundancy such that sufficient displays and controls are available in the main control room and remote locations to conduct an orderly reactor shutdown and to cool the reactor to cold shutdown conditions, even during design basis equipment failures.
- The principal functions of the Safety Parameter Display System (SPDS) as required by Supplement 1 to NUREG-0737, is integrated into the HSI design.
- Accepted human factors engineering principles are utilized for the HSI design in meeting the relevant requirements of General Design Criterion 19.
- The design bases for the Remote Shutdown System is as specified in Section 7.4.

18.3 PLANNING, DEVELOPMENT, AND DESIGN

18.3.1 Introduction

An integrated program plan to incorporate Human Factors Engineering (HFE) principles and to achieve an integrated design of the control and instrumentation systems and HSI of the ESBWR was prepared and implemented. The program plan, titled "Design of Controls, Instrumentation and Human Machine Interfaces", presents a comprehensive, synergistic design approach with provisions for task analyses and human factors evaluations. Also included are formal decision analysis procedures to facilitate selection of design features which satisfy top level requirements and goals of individual systems and the overall plant. Procedures developed as part of the program plan address the following areas:

- Development of system functional and performance requirements.
- Analysis of tasks and allocation of functions.
- Evaluation of human factors and human-machine interfaces.
- Design of hardware and software.
- Verification and validation of hardware, software, and procedure development (refer to Chapter 13).

The program plan and the associated procedures provided guidance for the conduct of the ESBWR HSI design development activities, including (1) definition of the standard design features of the control room HSI and (2) definition of the inventory of controls and instrumentation necessary for the control room crew to follow the operation strategies given in the ESBWR Emergency Procedure Guidelines and to complete the important operator actions described in the Probabilistic Risk Assessment (Subsection 18.3.3).

18.3.2 Standard Design Features

The ESBWR control room HSI design contains a group of standard features, which form the foundation for the detailed HSI design. These features are described in Subsection 18.4.2.

The development of the control room HSI standard design features was accomplished through consideration of existing control room operating experience; a review of trends in control room designs and existing control room data presentation methods; evaluation of new HSI technologies, alarm reduction and presentation methods; and validation testing of two dynamic control room prototypes. The prototypes were evaluated under simulated normal and abnormal reactor operating conditions and utilized experienced nuclear plant control room operators. Following the completion of the prototype tests and employing their results, the standard control room HSI design features were finalized.

18.3.3 Inventory of Controls and Instrumentation

The ESBWR Emergency Procedure Guidelines (EPGs), presented in Appendix 18A, and the important operator actions identified in the Probabilistic Risk Assessment (PRA), presented in Chapter 19, provided the bases for an analysis of the information and control capability needs of the main control room operators based upon the operation strategies. This analysis defines a minimum set of controls, displays, and alarms which will enable the operating crew to perform the actions that would be specified in the Emergency Operating Procedures (EOPs) and the important operator actions identified in the PRA. The COL applicant will complete the analysis (Subsection 18.8.8).

18.3.4 Detailed Design Implementation Process

The process by which the detailed equipment design implementation of the ESBWR HSI is completed is discussed in Section 18.7 and in Appendix 18E. This process builds upon the standard HSI design features that are discussed in Subsections 18.4.2 and 18.4.3. Embedded in the process (see Figure 18E-1) are a number of NRC conformance reviews in which various aspects and outputs of the process are evaluated against established acceptance criteria presented in Tables 18E-1 through 18E-5.

18.4 CONTROL ROOM STANDARD DESIGN FEATURES

This Section presents the standard design features of the Human System Interface (HSI) in the control room.

18.4.1 Introduction

The standard design features are based upon proven technologies and have been demonstrated, through broad scope control room dynamic simulation tests and evaluation, to satisfy the ESBWR HSI design goals and design bases as given in Section 18.2. The specific technologies utilized in the main control room HSI are listed in Subsection 18.4.3. Appendix 18D presents an example of a control room HSI design implementation that incorporates these design features. Validation of the implemented MCR design includes evaluation of the standard design features performed as part of the design implementation process as defined by the acceptance criteria presented in Tables 18E-1 through 18E-5.

18.4.2 Standard Design Feature Descriptions

18.4.2.1 Listing of Features

The ESBWR control room HSI design incorporates the following standard features:

(1) A single, integrated control console staffed by two operators; the console has a low profile such that the operators can see over the console from a seated position.

(2) The use of computer system driven on-screen control Video Display Units (VDUs) for safety-related system monitoring and nonsafety-related system control and monitoring.

(3) The use of a separate set of on-screen control VDUs for safety-related system control and monitoring (Essential DCIS) and separate on-screen control VDUs for nonsafety-related system control and monitoring (Non-Essential DCIS); the operation of these two sets of VDUs is entirely independent. Further, the first set of VDUs and all equipment associated with their functions of safety-related system control and monitoring are divisionally separated and qualified to Class 1E standards.

(4) The use of dedicated function switches on the control console.

(5) Operator selectable automation of pre-defined plant operation sequences.

(6) The incorporation of an operator selectable semi-automated mode of plant operations, which provides procedural guidance on the main control console VDUs but does not control plant systems and equipment.

(7) The capability to conduct all plant operations in an operator manual mode.

(8) The incorporation of a large display panel that presents information for use by the entire control room operating staff.

(9) The inclusion on the large display panel of fixed-position displays of key plant parameters and major equipment status.

(10) The inclusion in the fixed-position displays of both Class1E-qualified and non-1E display elements.

(11) The independence of the fixed-position displays from the Non-Essential Distributed Control and Instrumentation System (NE-DCIS.

(12) The inclusion within the large display panel of a large VDU, which is driven by the Non-Essential Distributed Control and Instrumentation System (NE-DCIS).

(13) The incorporation of a “monitoring only” supervisor’s console which includes VDUs on which display formats available to the operators on the main control console are also available to the supervisors.

(14) The incorporation of the Safety Parameter Display System (SPDS) function as part of the plant status summary information, which is continuously displayed on the fixed-position displays on the large display panel.

(15) A spatial arrangement between the large display panel, the main control console and the shift supervisor’s console, which allows the entire control room operating crew to conveniently view the information presented on the large display panel.

(16) The use of fixed-position alarm tiles on the large display panel.

(17) The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms.

(18) The use of VDUs to provide alarm information in addition to the alarm information provided through the fixed-position alarm tiles on the large display panel.

The COL applicant shall validate the design of each of the main control room standard design features (Subsection 18.8.5).

The remainder of this subsection provides further descriptions of these standard design features.

18.4.2.2 Main Control Console

The main control console (MCC) comprises the workstations for the two control room plant operators. It is configured such that each operator is provided with controls and monitoring information necessary to perform their assigned tasks and allows the operators to view all of the displays on the large display panel (Subsection 18.4.2.7) from a seated position.

The main control console, in concert with the large display panel, provides the controls and displays required to operate the plant during normal plant operations, abnormal events and emergencies. These main control console controls and displays include the following:

- On-screen control VDUs for safety-related system monitoring and nonsafety-related system control and monitoring, which are driven by the NE-DCIS (Subsection 18.4.2.3).
- A separate set of on-screen controls VDUs for safety-related system control and monitoring and separate on-screen control VDUs for nonsafety-related system control and monitoring; the operation of these two sets of VDUs is entirely independent of the NE-DCIS. Further, the first set of VDUs and the equipment associated with their functions of safety-related system control and monitoring are divisionally separated and qualified to Class 1E standards (Subsection 18.4.2.4).
- Dedicated function switches (Subsection 18.4.2.5).

The main control console is also equipped with a limited set of dedicated displays for selected functions (e.g., the Turbine Control).

In addition to the above equipment, the main control console is equipped with both intra-plant and external communications equipment and a laydown space is provided for hard copies of procedures and other documents required by the operators during the performance of their duties.

18.4.2.3 NE-DCIS Driven VDUs

A set of on-screen control VDUs is incorporated into the main control console design to support the following activities:

- Monitoring of plant systems, both safety-related and nonsafety-related.
- Control of nonsafety-related system components; and presentation of system and equipment alarm information.

This set of VDUs is driven by the plant NE_DCIS. Thus, data collected by the NE-DCIS is available for monitoring on these VDUs. The available display formats can be displayed on any of these VDUs.

18.4.2.4 NE-DCIS Independent VDUs

Sets of VDUs that are independent of the NE-DCIS are also installed on the main control console. Independent processors drive these VDUs. They are divided into two subsets:

(1) The first subset consists of those VDUs that are dedicated, divisionally separated devices. The VDUs in this group can only be used for monitoring and control of equipment within a given safety division. The VDUs are qualified, along with their supporting display processing equipment, to Class 1E standards.

(2) The second subset of NE-DCIS independent VDUs is used for monitoring and control of nonsafety-related plant systems. The VDUs in this subset are not qualified~~.~~ to Class 1E standards.

18.4.2.5 Dedicated Function Switches

Dedicated function switches are installed on the main control console. These devices provide faster access and feedback compared to that obtainable with soft controls. These dedicated switches are implemented in hardware, so that they are located in a fixed position and are dedicated in the sense that each individual switch is used only for a single function, or for two very closely related functions (e.g., valve open/close).

The dedicated function switches on the main control console are used to support such functions as initiation of automated sequences of safety-related and nonsafety-related system operations, manual scram and reactor operating mode changes.

18.4.2.6 Automation Design

The ESBWR incorporates selected automation of the operations required during a normal plant startup/shutdown and during normal power range maneuvers. Subsection 7.7 describes the Plant Automation System (PAS) function that is the primary ESBWR function for implementing the automation features for normal ESBWR plant operations.

18.4.2.6.1 Automatic Operation

When placed in automatic mode, the PAS performs sequences of automated plant operations by sending supervisory mode change commands and set point changes to lower-level, nonsafety-related plant system controllers. The PAS cannot directly change the status of a safety-related system. When a change in the status of a safety-related system is required to complete the selected operation sequence, the PAS provides prompts to guide the operator in manually performing the change using the appropriate safety-related operator interface controls on the main control console.

The operator can stop an automatic operation at any time. The PAS logic also monitors plant status, and will automatically revert to manual operating mode when a major change in plant status occurs (e.g., reactor scram or turbine trip). When such abnormal plant conditions occur, PAS automatic operation is suspended and the logic in the individual plant systems and equipment directs the automatic response to the plant conditions. Similarly, in the event that the operational status of the PAS or interfacing systems changes (e.g., equipment failures), operation reverts to manual operating mode. When conditions permit, the operator may manually reinitiate PAS automatic operation.

The COL applicant shall evaluate the effects of automation strategies on operator reliability and the appropriateness of the ESBWR automation design (Subsection 18.8.3). Also, COL applicant shall consider malfunctions of the PAS (Subsection 18.8.10).

18.4.2.6.2 Semi-Automated Operation

The PAS also includes a semi-automatic operational mode, which provides automatic operator guidance for accomplishing the desired normal changes in plant status; however, in this mode, the PAS performs no control actions. The operator must activate the necessary system and equipment controls for the semi-automatic sequence to proceed. The PAS monitors the plant status during the semi-automatic mode in order to check the progression of the semi-automatic sequence and to determine the appropriate operator guidance to be activated.

18.4.2.6.3 Manual Operation

The manual mode of operation in the ESBWR corresponds to the manual operations of conventional BWR designs in which the operator determines and executes the appropriate plant control actions without the benefit of computer-based operator aids. The manual mode provides a default operating mode in the event of an abnormal condition in the plant. The operator can completely stop an automated operation at any time by simply selecting the manual-operating mode. The PAS logic will also automatically revert to manual mode when abnormal conditions occur.

18.4.2.7 Large Display Panel

The large display panel provides information on overall plant status with real-time data during all phases of plant operation. The information on the large display panel can be viewed from the main control console and the supervisor's console. The large display panel includes fixed-position displays (Subsection 18.4.2.8), a variable display (Subsection 18.4.2.9) and spatially dedicated alarm windows (Subsection 18.4.2.12).

18.4.2.8 Fixed-Position Display

The fixed-position portion of the large display panel provides key plant information for viewing by the entire control room staff. The dynamic display elements of the fixed-position displays are driven by dedicated microprocessor-based controllers, which are independent of the NE-DCIS

Those portions of the large display panel, which present safety-related information, are qualified to Class 1E standards. The COL applicant shall address the human factors aspects of TMI Item I.E.3, Safety System Status Monitoring (Subsection 18.8.9).

The information presented in the fixed-position displays includes the critical plant parameter information, as defined by the SPDS requirements of NUREG–0737, Supplement 1, and the Type A Post-Accident Monitoring (PAM) instrumentation required by Regulation Guide 1.97. (Refer to Subsection 18.4.2.11 for a discussion of the SPDS and to Section 7.5 for a discussion of the PAM variables.)

18.4.2.9 Large Variable Display

The large variable display that is included on the large display panel is a VDU, which is driven by the plant NE-DCIS. Any screen format resident in the NE-DCIS can be shown on this large variable display.

18.4.2.10 Supervisors' Console

A console provided for the control room supervisors is equipped with VDUs on which any screen format resident in the NE-DCIS and available to the operators at the main control console is also available to the supervisors. The location of this console in the control room is discussed in Subsection 18.4.2.15.

18.4.2.11 Safety Parameter Display System

NUREG–0737 provides guidance for implementing Three Mile Island (TMI) action items. NUREG–0737, Supplement 1, clarifies the TMI action items related to emergency response capability, including item I.D.2, "Safety Parameter Display System" (SPDS). The principal purpose of the SPDS is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to prevent core damage. During emergencies, the SPDS serves as an aid in evaluating the current safety status of the plant, in executing symptom–based emergency operating procedures, and in monitoring the impact of engineered safeguards or mitigation activities. Selection of the parameters for inclusion in the SPDS display is based upon the ESBWR Emergency Procedure Guidelines (EPGs, Appendix 18A). The SPDS also operates during normal operation, continuously displaying information from which the plant safety status can be readily and reliably assessed. The ESBWR does not provide a separate SPDS, but rather, the principal functions of the SPDS (as required by NUREG–0737, Supplement 1) are integrated into the overall control room display capabilities. Displays of critical plant variables sufficient to provide information to plant operators about the following critical safety functions are provided on the large display panel as an integral part of the fixed-position displays:

- Reactivity control.
- Reactor core cooling and heat removal from the primary system.

- Reactor coolant system integrity.
- Radioactivity control; and Containment conditions.

Displays to assist the plant operator in execution of symptom–based emergency operating procedures are available at the main control console VDUs. Examples of these VDU displays are trend plots and operator guidance. Information regarding entry conditions to the symptomatic emergency procedures is provided through the fixed-position display of the critical plant parameters on the large display panel. The critical plant parameters on the large display panel are also viewable from the control room supervisor's monitoring station. The supplemental SPDS displays on the VDUs on the main control console are also accessible at the control room supervisor's monitoring station and may be provided in the Technical Support Center (TSC) and, optionally, in the Emergency Operations Facility (EOF). (Refer to Section 13.3 for the requirements on the TSC and EOF.)

Entry conditions to the symptom-based Emergency Operating Procedures (EOPs) are annunciated on the dedicated hardware alarm windows on the large display panel. The large display panel also displays the containment isolation status, safety-related systems status, and the following critical parameters:

(1) Reactor Pressure Vessel (RPV) pressure.

(2) RPV water level.

(3) Core neutron flux (startup range and power range instruments).

(4) Suppression pool temperature.

(5) Suppression pool water level.

(6) Drywell temperature.

(7) Drywell pressure.

(8) Drywell water level.

(9) Control rod scram status.

(10) Drywell oxygen concentration. (when monitors are in operation)

(11) Drywell hydrogen concentration (when monitors are in operation)

(12) Wetwell oxygen concentration (when monitors are in operation)

(13) Wetwell hydrogen concentration (when monitors are in operation), and (14) Containment radiation levels.

The oxygen monitoring instrumentation system is normally in continuous operation and hence the large display panel also includes continuous fixed-position display of wetwell and drywell oxygen concentrations. The hydrogen monitoring instrumentation is automatically started on a Loss-Of-Coolant Accident (LOCA) signal and, hence, continuous display is not required. Additional post-accident monitoring parameters, such as effluent stack radioactivity release (refer to Section 7.5 for a list of post-accident monitoring parameters), may be displayed at the large variable display or at the main control console VDUs on demand by the operator.

The SPDS is required to be designed so that the displayed information can be readily perceived and comprehended by the control room operating crew. Compliance with this requirement is assured because of the incorporation of accepted human factors engineering principles into the overall control room design implementation process (refer to Subsection 18.7 for a discussion of the design implementation process).

All of the continuously displayed information necessary to satisfy the requirements for the SPDS, as defined in NUREG-0737, Supplement 1, is a COL information requirement.

THE COL applicant shall evaluate the SPDS against the requirements of Paragraph 3.8a of NUREG-0737, Supplement 1, and confirm that the design meets the applicable criteria.

18.4.2.12 Fixed-Position Alarms

Specially dedicated fixed-position alarm tiles on the large display panel annunciate the key, plant-level alarm conditions that indicate entry into the emergency operating procedures or otherwise potentially affect plant availability or plant safety, or indicate the need of immediate operator action.

18.4.2.13 Alarm Processing Logic

Alarm prioritization and filtering logic is employed in the ESBWR design to enhance the presentation of meaningful alarm information to the operator and reduce the amount of information, which the operators must absorb, and process during abnormal events.

Alarm prioritizing is accomplished in the ESBWR through the designation of three categories of alarm signals. The first of these is the important alarms. These are defined as those alarms, which notify the operators of changes in plant status regarding safety, and include those items, which are to be checked in the event of accidents or anticipated operational occurrences. The important alarms are displayed on the fixed-position tiles discussed in Subsection 18.4.2.12.

The second category is the system-specific alarms, which are provided to notify the operators of system-level abnormalities or non-normal system statuses. Examples of these are:

- Main pump trips caused by system process, power source or control abnormalities.
- Valve closures in cooling or supply lines.
- Decreases in supply process values.
- Loss of a backup system.
- System isolation.
- Bypassing safety-related systems and systems which are undergoing testing.

The system-specific alarms are also shown on the fixed-position tiles discussed in Subsection 18.4.2.12.

Equipment alarms make up the third category of alarms in the prioritizing scheme and are discussed in Subsection 18.4.2.14.

Alarm suppression in the ESBWR is based upon the following concepts:

- Suppression based on the operating mode. The plant-operating mode is defined on the basis of the hardware or process status, and alarms that are not relevant to the current operating mode are suppressed. For example, alarms that are needed in the “RUN” mode but are unnecessary in the “SHUTDOWN” mode are suppressed.
- Suppression of subsidiary alarms. Alarms are suppressed if they are logically consequent to the state of operation of the hardware or to the process status. For example, scram initiation (a plant-level alarm condition announced with a fixed-position alarm tile on the large display panel) will logically lead to a fine motion control rod drive (FMCRD) hydraulic control unit scram accumulator low pressure (also an alarm condition). Such subsidiary alarms are suppressed if they simply signify logical consequences of the systems operation.
- Suppression of redundant alarms. When there are overlapping alarms, such as “high” and “high-high” or “low” and “low-low”. Only the most severe of the conditions is alarmed and the others are suppressed.

Operators may activate or deactivate the alarm suppression logic at any time.

18.4.2.14 Equipment Alarms

Alarms that are not indicated by fixed-position alarm tiles on the large display panel (i.e., those alarms of nominally lower level importance such as those related to specific equipment status) are displayed to the control room operating staff through the main control console VDUs. The supplemental alarm indications and supporting information regarding the plant-level alarms, which are presented on the large display panel, are also presented on the VDUs.

18.4.2.15 Control Room Arrangement

In the ESBWR main control room arrangement, the main control console is located directly in front of the large display panel for optimum viewing efficiency by the plant operators seated at the main console. The shift supervisor’s console is also placed in front of the large display panel, but at a somewhat greater distance than the main control console. The shift supervisor is, thus, in a position behind the control console operators. This arrangement allows all control room personnel to view the contents of the large panel displays (see Figure 18D-1).

18.4.3 Control Room HSI Technology

The ESBWR main control room standard design features described in the preceding subsections include, in their design, equipment that utilizes a variety of technologies to control and monitor the plant processes. This subsection provides a summary listing and description of the technologies, which are utilized in these control and monitoring functions. For this purpose, the term “technology” is taken to have the following definition: “The equipment, including both hardware and software, employed to directly accomplish the functions of control and monitoring of the plant processes”.

Hardware such as consoles, panels, cabinets, control room lighting, Heating Ventilation Air Conditioning (HVAC) and plant communication equipment, which has a supporting role but is not directly involved in the control and monitoring processes, is excluded.

The scope of this section is limited to the main control room and the remote shutdown station areas of the plant and includes all technology, regardless of use in prior designs.

The list format includes a brief description of each item of equipment:

(1) Hardware switches such as multi-position rotary, pushbutton, rocker, toggle and pull-to-lock types.

(2) Soft switch, the functions of which may be changed through the execution of software functions.

(3) Continuous adjustment controls, such as rotary controls and thumbwheels.

(4) Visual Display Units (VDUs) with full color screens, including large reverse projection screens, cathode ray tubes and flat panel display screens.

(5) On-screen control utilized with the units in 2 and 4, above.

(6) VDU screen format such as large screen optical projection display formats; text displays, including menus and tabular information and graphical displays, including trend plots, 2-D Plots, P&IDs and other diagrams and pictorial information.

(7) Analog meters which employ a hardware medium to pictorially or graphically present quantitative and qualitative information concerning plant process parameters; this includes analog meters using digitally controlled Light Emitting Diodes (LEDs) and digital readouts.

(8) Fixed-position digital displays which present alphanumeric information in a hardware medium. These can be back-lit.

(9) Fixed-position hardware mimic displays which schematically represent plant systems and components and their relationships utilizing pictorial elements, labels and indicator lights.

(10) Fixed-position alarm tiles which use light to indicate the alarm state.

(11) An audio signal system that is coordinated to the alarm tiles in #10, above, and utilizes prioritization and alarm reduction logic and pre-defined set points to alert operators to plant status changes.

(12) Printers and printer/plotters used to provide hard copy output in the form of plots, logs and text.

(13) Keyboards that are composed of alphanumeric and/or assignable function keys and function as computer input devices.

18.5 REMOTE SHUTDOWN SYSTEM

The Remote Shutdown System (RSS) provides a means to safely shut down the plant from outside the main control room. It provides control of the plant systems needed to bring the plant to hot shutdown, with the subsequent capability to attain cold shutdown, in the event that the control room becomes uninhabitable.

The RSS design is described in Subsection 7.4.2. All of the controls and instrumentation required for RSS operation are identified in Subsection 7.4.2 and in Figure 7.4-2.

The COL applicant shall evaluate alternate design approaches for reliability and confirmation of the adequacy of the RSS design (Subsection 18.8.6).

18.6 SYSTEMS INTEGRATION

18.6.1 Safety-Related Systems

The operator interfaces with the safety-related systems through a variety of methods. Dedicated hardware switches are used for system initiation and logic reset, while system mode changes are made with other hardware switches. Safety-related VDUs provide capability for individual safety equipment control, status display and monitoring; nonsafety-related VDUs are used for additional safety-related system monitoring. The large fixed-position display provides plant overview information. Instrumentation and control aspects of the microprocessor-based Safety System Logic and Control (SSLC) are described in Subsection 7.3.4.

Divisional separation for control, alarm and display equipment is maintained. The SSLC processors provide alarm signals to their respective safety-related alarm processors and provide display information to the divisionally dedicated VDUs. The SSLC microprocessors communicate with their respective divisional VDU controllers through Essential Distributed Control and Information System (E-DCIS). The divisional VDUs have on-screen control capability and are classified as safety-related equipment. These VDUs provide control and display capabilities for individual safety systems if control of a system component is required.

Divisional isolation devices are provided between the safety-related systems and nonsafety-related communication networks so that failures in the nonsafety-related equipment will have no impact on the ability of the safety-related systems to perform their design functions. The nonsafety-related communication network is part of Non-Essential Distributed Control and Information System (NE-DCIS) described in Subsection 7.7.7.

Safety-related system process parameters, alarms and system status information from the SSLC are communicated to the NE-DCIS through isolation devices for use by other equipment connected to the communication network. Selected operator control functions are performed through dedicated hardware control switches, which are Class 1E qualified and divisionally separated on the main control console. These hardware switches communicate with the safety-related systems logic units through hardwire transmission lines.

The divisionally dedicated VDUs are classified as safety-related equipment. These VDUs provide control and display capabilities for individual safety-related systems if control of a system component is required. Normally, such control actions are performed for equipment surveillance purposes only, as the normal method of system control is through the mode-oriented master sequence switches.

18.6.2 Nonsafety-Related Systems

Operational control of nonsafety-related systems is accomplished using nonsafety-related, on-screen control VDUs and dedicated hardware switches. Hardware switches for nonsafety-related equipment on the MCC are connected with nonsafety-related systems logic units by hardwired cabling.

Nonsafety-related data is processed through the NE-DCIS, which provides redundant and distributed control and instrumentation data communications network to support the monitoring and control of interfacing plant systems.

Alarms for entry conditions into the emergency operating procedures are provided by the alarm processing units, both safety and nonsafety-related. Equipment level alarm information is presented by the computer system on the MCC VDUs.

The fixed position wide display panel (WDP) provides the critical plant operating information such as power, water level, temperature, pressure, flow and status of major equipment and avail-availability of safety systems with mimic on the main control room during plant normal, abnormal and emergency operating condition.

18.7 DETAILED DESIGN OF THE OPERATOR INTERFACE SYSTEM

The standard design features of the ESBWR main control room HSI, discussed in Subsection 18.4.2, provide the framework for the detailed equipment hardware and software designs developed following the design and implementation process described in Appendix 18E. This process is made up of eight major elements, as illustrated in Figure 18E-1.

As part of the Appendix 18E discussion of the HSI design and implementation plan elements, detailed acceptance criteria are specified and used to govern and direct all ESBWR HSI design implementations, which reference the Certified Design. These detailed acceptance criteria, presented in Table 18E-3 of Appendix 18E, encompass the set of necessary and sufficient design implementation related activities required to maintain the implemented HSI design in compliance with accepted Human Factors Engineering (HFE) principles and accepted digital electronics equipment and software development methods.

Also, as part of the detailed design implementation process described in Appendix 18E, operator task analysis is performed as a basis for evaluating details of the design implementation and HSI requirements are specified.. The evaluation of the integrated control room design includes the confirmation of the ESBWR main control room standard design features.

18.8 COL INFORMATION

18.8.1 Plant Specific Reactor Building Operating Values for EPGs/SAGs

The plant specific Reactor Building operating parameter values of temperature, radiation, and water level must be determined to support the generation of the Emergency Procedure and Severe Accident Guidelines (EPG/SAGs) in Appendix 18A.

18.8.2 EPG/SAG Appendix C: Calculation Input Data and Results

Plant specific input data as described in the BWR Owners Group EPG/SAG Rev. 2, Appendix C: Calculation is required to generate the plant specific variables and limits curves as results, that are used in the plant specific EPGs/SAGs in Appendix 18A.

18.8.3 HSI Design Implementation Process

The HSI Design Implementation Process as described in Appendix 18E is the responsibility of the COL applicant. In addition, the following specific COL information is in effect.

18.8.4 Number of Operators Needing Controls Access

The number of operators needing access to the controls on the main control panel is evaluated and the ESBWR control room staffing arrangement (Subsection 18.2, Item 1) is confirmed to be adequate. In addition, the roles and responsibilities of the shift supervisor and assistant shift supervisor are clearly specified. The results of the evaluation are placed in the Human Factors Engineering (HFE) Issue Tracking System (Subsection II.2 of Table 18E-1).

18.8.5 Automation Strategies and Their Effect on Operator Reliability

Automation strategies for plant operation are evaluated for effects on operator reliability and the appropriateness of the ESBWR automation design (Subsection 18.4.2.6.1) confirmed. This evaluation is performed according to the criteria of Subsection II of Table 18E-1 and the results of the evaluation are placed in the HFE Issue Tracking System.

18.8.6 SPDS Integration With Related Emergency Response Capabilities

The design of the Safety Parameter Display System (SPDS, Subsection 18.4.2.11) is evaluated against the requirements of Paragraph 3.8a of NUREG-0737, Supplement 1, and confirmed to be in compliance with all applicable criteria. The results of the evaluation are placed in the HFE Issue Tracking System.

18.8.7 Standard Design Features Design Validation

The design of each of the main control room standard design features (Subsection 18.4.2.1) is validated using the applicable criteria in Subsection VII of Table 18E-1 and Table 18E-2. The results of the validation are placed in the HFE Issue Tracking System.

18.8.8 Remote Shutdown System Design Evaluation

Digital versus analog design approaches for the Remote Shutdown System (RSS) are evaluated for reliability and the adequacy of the ESBWR RSS design (Subsection 18.5) confirmed. The results of the evaluation are placed in the HFE Issue Tracking System.

18.8.9 Local Valve Position Indication

The necessity for providing local Valve Position Indication (VPI) for each valve in any of the following categories is evaluated:

(1) All power-operated valves (e.g., motor, hydraulic and pneumatic),

(2) All large manual valves (i.e., 5 cm or larger),

(3) Small manual valves (i.e., less than 5 cm) which are important to safe plant operations.

These evaluation records are placed in the HFE Issue Tracking System.

18.8.10 Operator Training

The operator training program meets the requirements of 10 CFR 50.120 and 10 CFR 55 (Subsection II.1.c of Table 18E-1).

18.8.11 Safety System Status Monitoring

The human factors aspects of TMI Item I.E.3, "Safety System Status Monitoring", are addressed as part of the detailed design implementation process (Subsection 18.4.2.8).

18.8.12 PAS Malfunction

Malfunctions of the Plant Automation System (PAS) function of the process computer system are considered (Subsection 18.4.2.6.1).

18.8.13 Local Control Stations

All operations at local control stations, which are critical to plant safety, as defined in Paragraph V.1.c of Table 18E-1, are considered. The results of these evaluations are incorporated into the HFE Issue Tracking System.

18.8.14 As-Built Evaluation of MCR and RSS

A report is prepared which documents that the as-built Main Control Room (MCR) and Remote Shutdown Station (RSS) conform to the certified and validated main control room and remote shutdown station configurations. Aspects of the as-built MCR and RSS considered in this report are the area and panel layouts, operator environment, alarms, displays, controls and general human-system interface characteristics.

18.8.15 Accident Monitoring Instrumentation

The instrumentation described in TMI Item II.F.1, "Additional Accident Monitoring Instrumentation," is evaluated with regard to the impact of the inclusion of that instrumentation in the MCR HSI on the potential for operator error. The results of this evaluation are placed in the HFE Issue Tracking System.

18.8.16 In-Core Cooling Instrumentation

The instrumentation described in TMI Item II.F.2, “Instrumentation For Detection of Inadequate Core-Cooling,” is evaluated with regard to the impact of the inclusion of that instrumentation in the MCR HSI on the potential for operator error. The results of this evaluation are placed in the HFE Issue Tracking System.

18.8.17 Performance of Critical Tasks

The HSI is evaluated with respect to providing the controls, displays and alarms necessary for timely performance of critical tasks. Critical tasks include, as a minimum, those operator actions which have significant impact on the Probabilistic Risk Assessment (PRA) results, as presented in Section 19D.7, and the operator actions to isolate the reactor and inject water for the postulated event scenarios of a common-mode failure of the Safety System Logic and Control System and/or the Essential Distributed Control and Information System concurrent with the design basis main steamline, feedwater line or shutdown cooling line break Loss Of Coolant Accident (LOCA, Paragraph V.2.d of Table 18E-1). The results of this evaluation are placed in the HFE Issue Tracking System.

18.8.18 Plant Status and Post-Accident Monitoring

The main control instrumentation described in TMI Item I.D.5 (2), “Plant Status and Post-Accident Monitoring” is evaluated with regard to the impact of the inclusion of that instrumentation in the MCR HSI on the potential for operator error and the results of the evaluation are placed in the HFE Issue Tracking System.

18.8.19 Performance of HSI Verification and Validation on a dynamic simulator

Human reliability analysis assumptions such as decision-making and diagnosis strategies for dominant sequences are validated by walkthrough analyses with personnel with operational experience using a plant-specific control room mockup or simulator (Subsection VI.1.f of Table 18E-1). Reviews should be conducted before the final quantification stage of the PRA.

The results of this evaluation are placed in the HFE issue Tracking System.

18.8.20 Emergency Operation Information and Control

Analysis based upon the operation strategies given in the Emergency Procedure Guidelines (EPGs) and the significant operator actions determined by the PRA are performed. As a result of these analysis the minimum inventory of controls, displays and alarms needs of the main control room operators, is determined.

Information and control needs for each operative instruction or action are developed through task analysis.

The results of this evaluation are placed in the HFE issue Tracking System and Appendix 18F.

18.8.21 Supporting Analysis for Emergency Operation Information and Controls

Guidelines developed from a research program of advanced control panel designs are used to specify the type of implementation device for controls, displays and alarms: Fixed Position Controls, Alarm and Displays, Divisional and Non-divisional VDUs.

Results of the operational analysis of each step of the ESBWR EPGs and PRA important operator actions are considered.

EPG considerations are included to address issues associated with ATWS stability changes. The results of this evaluation are placed in the HFE Issue Tracking System and Appendix 18H.

18A. EMERGENCY PROCEDURE AND SEVERE ACCIDENT GUIDELINES

18A.1 INTRODUCTION

The EPGs/SAGs are divided into Emergency Procedure Guidelines (EPGs) and Severe Accident Guidelines (SAGs). The EPGs define strategies for responding to emergences and to events that may degrade into emergencies until primary containment flooding is required. The EPGs and SAGs have been modified based on the ESBWR system design features. The following generic symptomatic EPGs have been developed:

- RPV Control Guideline
- Primary Containment Control Guideline
- Reactor Building Control Guideline
- Radioactivity Release Control Guideline

The SAGs define strategies applicable after containment flooding is required. They comprise two guidelines:

- RPV and Containment Flooding
- Containment and Radioactivity Release Control

The RPV Control Guideline maintains adequate core cooling, shuts down the reactor, and cools down the RPV to cold shutdown conditions. This guideline is entered whenever low RPV water level, high RPV pressure, or high drywell pressure occurs, or whenever a condition that requires reactor scram exists and reactor power is above the APRM downscale trip or cannot be determined.

The Containment Control Guideline maintains containment integrity and protects equipment in the containment with respect to the consequences of all mechanistic events. This guideline is entered whenever suppression pool temperature, drywell temperature, drywell pressure, suppression pool water level, or containment hydrogen concentration is above its high operating limit or suppression pool water level is below its low operating limit. Suppression pool and drywell temperatures are determined by plant-specific procedures for determining bulk suppression pool water temperature and drywell atmosphere average temperature, respectively.

The Reactor Building Control Guideline protects the controlled areas, limits radioactivity release to the controlled areas, and either maintains controlled area integrity or limits radioactivity release from the controlled areas. This guideline is entered whenever a controlled area temperature, radiation level, or water level is above its maximum normal operating value or controlled area differential pressure reaches zero.

The Radioactivity Release Control Guideline limits radioactivity release into areas outside the containment and controlled areas. This guideline is entered whenever offsite radioactivity release rate is above that which requires an Alert.

Table 18A-1 is a list of the abbreviations used in the guidelines.

Brackets [ ] enclose plant unique setpoints, design limits, pump shutoff pressures, etc., and parentheses ( ) within brackets indicate the source for the bracketed variable. Included in these guidelines is the current state of design values for the ESBWR expressed in both SI and English units. Brackets are also used to indicate potential systems which might be employed for those systems whose use in the indicated application is not yet well defined such as use of the fire protection system to inject water into the RPV.

At various points throughout these guidelines, operator precautions are identified by a circled number in reverse type in the right margin. The number within the circle refers to a numbered "Caution" contained in the Operator Cautions section. These "Cautions" are brief and succinct red flags for the operator.

The emergency procedure guidelines for the ESBWR design address all systems that may be used to respond to an emergency.

At various points within these guidelines, limits are specified beyond which certain actions are required. The bases and calculational methods for these limits are defined in the BWROG Emergency Procedure and Severe Accident Guidelines, Revision 2, Appendices B and C, respectively. While conservative, these limits are derived from engineering analyses utilizing best-estimate (as opposed to licensing) models. Consequently, these limits are generally not as conservative as the limits specified in a plant's Technical Specifications. This is not to imply that operation beyond the Technical Specifications is recommended in any emergency. Rather, such operation is required and is now permitted under certain degraded conditions in order to safely mitigate the consequences of those degraded conditions. The limits specified in the guidelines establish the boundaries within which continued safe operation of the plant can be assured. Therefore, conformance with the guidelines does not ensure strict conformance with a plant's Technical Specifications or other licensing bases.

At other points within these guidelines, defeating safety system interlocks and initiation logic is specified. This is also required in order to safely mitigate the consequences of degraded conditions, and it is generally specified only when conditions exist for which the interlock or logic was not designed. Defeating other interlocks may also be required due to instrument failure, etc., but these interlocks cannot be identified in advance and are therefore not specified in the guidelines.

The entry conditions for these emergency procedure guidelines are symptomatic of both emergencies and events that may degrade into emergencies. The guidelines specify actions appropriate for both. Therefore, entry into procedures developed from these guidelines is not conclusive that an emergency has occurred.

Each procedure developed from these emergency procedure guidelines (EPGs) is entered whenever any of its entry conditions occurs, irrespective of whether that procedure has already been entered or is presently being executed. When RPV and Containment Flooding is required, the EPGs transition to the Severe Accident Guidelines (SAGs). The EPGs are exited and the operator returns to non-emergency procedures when either one of the exit conditions specified in the procedure is satisfied or it is determined that an emergency no longer exists. After a procedure developed from these guidelines has been entered, subsequent clearing of all entry conditions for that procedure is not, by itself, conclusive that an emergency no longer exists.

Procedures developed from these emergency procedure guidelines specify symptomatic operator actions, which will maintain the reactor plant in a safe condition and optimize plant response and margin to safety irrespective of the initiating event. However, for certain specific events (e.g., earthquake, tornado, blackout, or fire), emergency response and recovery can be further enhanced by additional auxiliary event-specific operator actions which may be provided in supplemental event-specific procedures intended for use in conjunction with the symptomatic procedures. As with actions specified in any other procedure intended for use with the symptomatic procedures, these event-specific operator actions must not contradict or subvert the operator actions specified in the symptomatic procedures and must not result in loss or unavailability of equipment the operation of which is specified in these procedures.

The ESBWR EPG/SAG was derived from Rev. 2 of the generic BWR Owners' Group Emergency Procedure and Severe Accident Guidelines. Adaptations were required to accommodate the unique design philosophy, plant configuration and specific systems and components of the ESBWR. Appendix 18B provides the bases for the modified EPG as follows:

- Section 18B.1 briefly describes the ESBWR features which greatly impacted the ESBWR EPG;
- Section 18B.2 discusses the major EPG changes arising from these features; and
- Section 18B.3 tabulates, step-by-step, the differences between EPG/SAG Rev. 2 and the ESBWR EPG and provides the bases for these differences.

18A.2 OPERATOR CAUTIONS

This section lists cautions which are applicable at one or more specific points within the guidelines. Where a caution is applicable, it is identified by a circled number in reverse type in the right margin.

❶

RPV water level indications are affected by instrument run temperatures and RPV pressure:

- If the temperature near any instrument run is above the RPV Saturation Temperature, the instrument may be unreliable due to boiling in the run.
- Each instrument in the following table may be used to determine RPV water level only when the instrument reads above the Minimum Indicated Level or the temperatures near all the instrument reference leg vertical runs are below the Maximum Run Temperature.

Instrument	Range (in.)	Maximum Run Temperature (°F) DW	Maximum Run Temperature (°F) RB	Minimum Indicated Level (in.)
Fuel Zone	-317 to -17	324	NA	-301

- Each instrument in the following table may be used to determine RPV water level only when the instrument reads above the Minimum Indicated Level associated with the highest temperature near an instrument reference leg vertical run:

Instrument	Highest Drywell Run Temperature (°F) Between Low	High	Minimum Indicated Level (in.)
Narrow Range (0 to +60 in.)	—	278	0
	278	350	5
	350	450	13
	450	550	25

Instrument	Highest Drywell Run Temperature (°F) Between Low	High	Minimum Indicated Level (in.)
Wide Range (-150 to +60 in.)	—	211	-150
	211	250	-147
	250	350	-138
	350	450	-128
	450	550	-115

Shutdown Range	—	150	11
(-17 to +383)	150	250	25
	250	350	43
	350	450	67
	450	550	99

❷ A rapid increase in injection into the RPV may induce a large power excursion and result in substantial core damage.

❸ A steam explosion may occur if core debris drops into the lower drywell with a drywell water level greater than [0.7 m]. Containment flooding above [0.7 m] should be prevented until the lower drywell floor thermocouples indicate that all or most of the core debris has been ejected from the RPV.

18A.3 RPV CONTROL EMERGENGY PROCEDURE GUIDELINE

PURPOSE

The purpose of this guideline is to:

- Maintain adequate core cooling,
- Shut down the reactor, and
- Cool down the RPV to cold shutdown conditions ((120°F) 49°C <RPV water temp < 100°C (212°F) (cold shutdown conditions)]).

ENTRY CONDITIONS

The entry conditions for this guideline are any of the following:

- RPV water level below [1978 cm (778.75 in) (low level scram setpoint)]
- RPV pressure above [7.72 MPa (1120 psig (high RPV pressure scram setpoint)]
- Drywell pressure above [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)]
- A condition which requires reactor scram, and reactor power above [6% (APRM downscale trip)] or cannot be determined

OPERATOR ACTIONS

If while executing the following steps, Primary Containment Flooding is or has been required, enter [procedure developed from the RPV and Primary Containment Flooding Severe Accident Guideline].

RC-1 If a reactor scram has not been initiated, initiate a reactor scram.

Execute [Steps RC/L, RC/P, and RC/Q] concurrently.

RC/L Monitor and control RPV water level. ❶

RC/L-1 Initiate each of the following which should have initiated but did not:

- Isolation
- ECCS
- All Isolation Condensers
- [Emergency Diesel Generator]

If while executing the following step:

- Any control rod cannot be determined to be inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] and it has not been determined that the reactor will remain shutdown under all conditions without boron, enter [procedure developed from Emergency Procedure Guideline Contingency #3].
- RPV water level cannot be determined, enter [procedure developed from Emergency Procedure Guideline Contingency #2].

If while executing the following step primary containment water level and suppression chamber pressure cannot be maintained below Primary Containment Pressure Limit, but only if adequate core cooling can be assured, terminate injection into the RPV from sources external to the primary containment.

If RPV water level cannot be restored and maintained above [745.3 cm (293.4 in) (top of active fuel)] enter [procedure developed from RPV and Primary Containment Flooding Severe Accident Guideline].

RC/L-2 Restore and maintain RPV water level between [1978 cm (778.75 in). (low level scram setpoint or shutdown cooling RPV water level interlock, whichever is higher)] and [2189 cm (861.8 in) (high level trip setpoint)] with one or more of the following systems:

- Condensate/Feedwater
- CRD High Pressure Makeup mode
- FAPCS LPCI mode

If RPV water level cannot be restored and maintained above [1978 cm (778.75 in). (low level scram setpoint or shutdown cooling RPV water level interlock, whichever is higher)], maintain RPV water level above [1300 cm (511.8 in) (Level 1.5)].

RPV water level control may be augmented by one or more of the following systems:

- Fire System
- [Interconnections with other units]

If RPV water level cannot be restored and maintained above [1300 cm (511.8 in) (Level 1.5)], EMERGENGY RPV DEPRESSURIZATION IS REQUIRED.

RC/L-3 Restore and maintain RPV water level above [1978 cm (778.75 in) (low level scram setpoint or shutdown cooling RPV water level interlock, whichever is higher)] with one or more of the following systems:

- Condensate/Feedwater
- CRD High Pressure Makeup mode
- FAPCS LPCI mode

RPV water level control may be augmented by one or more of the following systems:

[
- Fire System
- Interconnections with other units
]

RC/P Monitor and control RPV pressure

If while executing the following steps:

- Emergency RPV Depressurization is anticipated and either all control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] or it has been determined that the reactor will remain shutdown under all conditions without boron, rapidly depressurize the RPV with the main turbine bypass valves and IC, irrespective of the resulting cooldown rate.

- Emergency RPV Depressurization is or has been required, enter [procedure developed from Emergency Procedure Guideline Contingency #1].

- RPV water level cannot be determined, enter [procedure developed from Emergency Procedure Guideline Contingency #2].

RC/P-1 If any SRV is cycling, initiate IC and manually open SRVs until RPV pressure drops to [6.45 MPa (935 psig) (RPV pressure at which all turbine bypass valves are fully open)].

If while executing the following steps:

- Suppression pool temperature cannot be maintained below the Heat Capacity Temperature Limit, maintain RPV pressure below the Limit, exceeding [55.55°C/hr (100°F/hr) (RPV cooldown rate LCO)] cooldown rate if necessary.
- Suppression pool water level cannot be maintained below the SRV Tail Pipe Level Limit, maintain RPV pressure below the Limit, exceeding [55.55°C/hr (100°F/hr) (RPV cooldown rate LCO)] cooldown rate if necessary.

If while executing the following steps:

- Boron Injection is required, and
- The main condenser is available, and
- There has been no indication of a steam line break,

open MSIVs, defeating MSL and Offgas high radiation interlocks and low RPV water level interlocks if necessary, to re-establish the main condenser as a heat sink.

RC/P-2 Stabilize RPV pressure at a pressure below [7.72 MPa (1120 psig) (high RPV pressure scram setpoint)] with the main turbine bypass valves.

RPV pressure control may be augmented by one or more of the following systems:

- IC
- SRVs, only when suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary; open SRVs in the following sequence if possible: [(SRV opening sequence)]; if the continuous SRV pneumatic supply is or becomes unavailable, place the control switch for each SRV in the [CLOSE or AUTO] position.
- RWCU/SDC high pressure shutdown cooling mode, defeating regenerative heat exchangers and filter/demineralizers and, if necessary, defeating SLC and other isolation interlocks.
- MSL drains
- RWCU (blowdown mode), only if no boron has been injected into the RPV or it has been determined that the reactor will remain shutdown under all conditions without boron; refer to [sampling procedures] prior to initiating blowdown.

> If while executing the following steps:
> - the reactor is not shutdown, return to [Step RC/P-2]
> - the RPV water temperature cannot be cooled down below 100°C {212°F} and further cooldown is required, initiate [Alternate Shutdown Cooling Procedure].

RC/P-3 When either:

- All control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)], or
- It has been determined that the reactor will remain shutdown under all conditions without boron, or
- [96.8 Kg (213 pounds) (Cold Shutdown Boron Weight)] of boron have been injected into the RPV, or
- The reactor is shutdown and no boron has been injected into the RPV,

depressurize the RPV and maintain the cooldown rate below [55.55°C/hr (100°F/hr) (RPV cooldown rate LCO)], defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary.

If one or more SRVs are being used to depressurize the RPV and the continuous SRV pneumatic supply is or becomes unavailable, depressurize with sustained SRV opening.

RC/Q Monitor and control reactor power.

> If while executing the following steps:
>
> - All control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)], enter [scram procedure].
> - It has been determined that the reactor will remain shutdown under all conditions without boron, enter [scram procedure].
> - The reactor is shutdown and no boron has been injected into the RPV, enter [scram procedure].

RC/Q-1 Confirm or place the reactor mode switch in SHUTDOWN.

RC/Q-2 If ARI has not initiated, initiate ARI.

> Execute [Steps RC/Q-3 and RC/Q-4] concurrently.

RC/Q-3 Either:

- When periodic neutron flux oscillations in excess of [25% (Large Oscillation Threshold)] peak-to-peak commence and continue, or
- Before suppression pool temperature reaches the Boron Injection Initiation Temperature,

BORON INJECTION IS REQUIRED; inject boron into the RPV with SLC.

If boron cannot be injected with SLC, inject boron into the RPV by one or more of the following alternate methods:

- CRD
- RWCU/SDC
- Feedwater
- Hydro pump

If boron is not being injected into the RPV by RWCU/SDC and RWCU/SDC is not isolated, bypass regenerative heat exchangers and filter/demineralizers.

RC/Q-4 Insert control rods as follows:

RC/Q-4.1 Reset ARI, defeating ARI logic trips if necessary.

RC/Q-4.2 Insert control rods with one or more of the following methods:

- De-energize scram solenoids.
- Vent the scram air header.
- Reset the scram, defeating RPS logic trips if necessary, and initiate a manual scram.
- Open individual scram test switches.
- Drive control rods, defeating RCIS and RWM interlocks if necessary.

18A.4 PRIMARY CONTAINMENT CONTROL EMERGENCY PROCEDURE GUIDELINE

PURPOSE

The purpose of this guideline is to:

- Maintain primary containment integrity, and
- Protect equipment in the primary containment.

ENTRY CONDITIONS

The entry conditions for this guideline are any of the following:

- Suppression pool temperature above [43.3°C (110°F) (most limiting suppression pool temperature LCO)]
- Drywell temperature above [57.22°C (135°F) (drywell temperature LCO or maximum normal operating temperature, whichever is higher)]
- Drywell pressure above [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)]
- Suppression pool water level above [5.50 m (18.05 ft) (maximum suppression pool water level LCO)]
- Suppression pool water level below [5.40 m (17.72 ft) (minimum suppression pool water level LCO)]
- Primary containment hydrogen concentration above [2% (high hydrogen alarm setpoint)]

OPERATOR ACTIONS

If while executing the following steps, Primary Containment Flooding is or has been required, enter [procedure developed from the Containment and Radioactivity Release Control Severe Accident Guideline].

Execute [Steps SP/T, DW/T, PC/P, SP/L, and PC/G] concurrently.

SP/T Monitor and control suppression pool temperature below [43.3°C (110°F) (most limiting suppression pool temperature LCO)] using available suppression pool cooling.

When suppression pool temperature cannot be maintained below [43.3°C (110°F) (most limiting suppression pool temperature LCO)]:

SP/T-1 Operate all available suppression pool cooling using only those FAPCS pumps not required to assure adequate core cooling by continuous operation in the LPCI mode.

SP/T-2 Before suppression pool temperature reaches the Boron Injection Initiation Temperature, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

SP/T-3 Before the Heat Capacity Temperature Limit is reached; operate all Isolation Condensers to depressurize the RPV.

SP/T-4 When suppression pool temperature and RPV pressure cannot be maintained below the Heat Capacity Temperature Limit, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

DW/T Monitor and control drywell temperature below [57.22°C (135°F) drywell temperature LCO or maximum normal operating temperature, whichever is higher] using available drywell cooling.

When drywell temperature cannot be maintained below [57.22°C (135°F) (drywell temperature LCO or maximum normal operating temperature, whichever is higher)]:

DW/T-1 Operate all available drywell cooling, defeating isolation interlocks if necessary.

DW/T-2 Before drywell temperature reaches [171.1°C (340°F) (maximum temperature at which ADS is qualified or drywell design temperature, whichever is lower)], enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

DW/T-3 When drywell temperature cannot be restored and maintained below [171.1°C (340°F) (maximum temperature at which ADS is qualified or drywell design temperature, whichever is lower)], EMERGENCY RPV DEPRESSURIZATION IS REQUIRED;

PC/P Monitor and control primary containment pressure below [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)] using the following systems:

- Containment Inerting System (CIS); use [CIS operating procedure].
- Reactor Building HVAC System; use [RBHVAC operating procedure]

PC/P-1 When suppression chamber pressure cannot be maintained below the Pressure Suppression Pressure, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

PC/P-2 Before suppression chamber pressure reaches Primary Containment Pressure Limit, vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary, to control pressure below Primary Containment Pressure Limit.

PC/P-3 When suppression chamber pressure exceeds Primary Containment Pressure Limit, vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary and irrespective of whether adequate core cooling is assured, to control pressure below Primary Containment Pressure Limit.

SP/L Monitor and control suppression pool water level.

SP/L-1 Maintain suppression pool water level between [5.50 m (18.05 ft) (maximum suppression pool water level LCO)] and [5.40 m (17.72 ft) (minimum suppression pool water level LCO)] using FAPCS; refer to [sampling procedure] prior to discharging water.

If suppression pool water level cannot be maintained above [5.40 m (17.72 ft) (minimum suppression pool water level LCO)], execute [Step SP/L-2].

If suppression pool water level cannot be maintained below [5.50 m (18.05 ft) (maximum suppression pool water level LCO)], execute [Step SP/L-3].

SP/L-2 SUPPRESSION POOL WATER LEVEL BELOW [5.40 m (17.72 ft) (minimum suppression pool water level LCO)]

Maintain suppression pool water level above [** m (** ft) (elevation of the downcomer openings)] [(0.61 m (2 ft) above the elevation of the top of the horizontal vents)].

If suppression pool water level cannot be maintained above [** m (** ft) (elevation of the downcomer openings)] [(0.61 m (2 ft). above the elevation of the top of the horizontal vents)], EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

SP/L-3 SUPPRESSION POOL WATER LEVEL ABOVE [5.50 m (18.05 ft) (maximum suppression pool water level LCO)]

Execute [Steps SP/L-3.1 and SP/L-3.2] concurrently.

SP/L-3.1 Maintain suppression pool water level below the SRV Tail Pipe Level Limit.

If suppression pool water level cannot be maintained below the SRV Tail Pipe Level Limit, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

If suppression pool water level and RPV pressure cannot be maintained below the SRV Tail Pipe Level Limit but only if adequate core cooling is assured, terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

If suppression pool water level and RPV pressure cannot be restored and maintained below the SRV Tail Pipe Level Limit, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

SP/L-3.2 Maintain suppression pool water level below [16.85 m (55.28 ft) (elevation of bottom of suppression chamber to drywell vacuum breaker openings [less vacuum breaker opening pressure in meters of water]).

If suppression pool water level cannot be maintained below [16.85 m (55.28 ft) (elevation of bottom of suppression chamber to drywell vacuum breaker openings [less vacuum breaker opening pressure in meters of water])] and if adequate core cooling can be assured:

Terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

PC/G Monitor and control primary containment hydrogen and oxygen concentrations.

If while executing the following steps the hydrogen or oxygen monitoring system is or becomes unavailable, sample the drywell and suppression chamber for hydrogen and oxygen in accordance with [sampling procedure].

Control hydrogen and oxygen concentrations in the drywell as follows:

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">**Drywell Oxygen Concentration**</td></tr>
<tr><td rowspan="3">< 5%</td><td colspan="3">≥ 5% or cannot be determined to be below 5%</td></tr>
<tr><td colspan="3">**Suppression Chamber Hydrogen Concentration**</td></tr>
<tr><td>None Detected</td><td>< 6%</td><td>≥ 6% or cannot be determined to be below 6%</td></tr>
<tr><td rowspan="3">**Drywell Hydrogen Concentration**</td><td>None Detected</td><td>No action required</td><td>No action required</td><td rowspan="2">[PC/G-2]</td><td rowspan="3">[PC/G-3]</td></tr>
<tr><td>< 6%</td><td rowspan="2">[PC/G-1]</td><td></td></tr>
<tr><td>≥ 6% or cannot be determined to be below 6%</td><td colspan="2"></td></tr>
</table>

Control hydrogen and oxygen concentrations in the suppression chamber as follows:

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">**Suppression Chamber Oxygen Concentration**</td></tr>
<tr><td rowspan="3">< 5%</td><td colspan="3">≥ 5% or cannot be determined to be below 5%</td></tr>
<tr><td colspan="3">**Drywell Hydrogen Concentration**</td></tr>
<tr><td>None Detected</td><td>< 6%</td><td>≥ 6% or cannot be determined to be below 6%</td></tr>
<tr><td rowspan="3">**Suppression Chamber Hydrogen Concentration**</td><td>None Detected</td><td>No action required</td><td>No action required</td><td rowspan="2">[PC/G-5]</td><td rowspan="3">[PC/G-6]</td></tr>
<tr><td>< 6%</td><td rowspan="2">[PC/G-4]</td><td></td></tr>
<tr><td>≥ 6% or cannot be determined to be below 6%</td><td colspan="2"></td></tr>
</table>

PC/G-1 Reduce drywell hydrogen and oxygen concentrations by one or both of the following methods:

- If the offsite radioactivity release rate is expected to remain below the offsite release rate specified in [Technical Specifications], vent and purge the drywell as follows, defeating isolation interlocks (except high radiation interlocks) if necessary:

If while executing the following steps the offsite radioactivity release rate reaches the offsite release rate specified in [Technical Specifications], secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

(1) Refer to [sampling procedure].

(2) Vent the drywell.

(3) If the drywell can be vented, purge the drywell by injecting nitrogen into the drywell.

(4) When hydrogen is no longer detected in the drywell, secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

PC/G-2 Reduce drywell hydrogen and oxygen concentrations by one or both of the following methods:

- If adequate core cooling is not assured or the offsite radioactivity release rate is expected to remain below the offsite radioactivity release rate which requires a General Emergency, vent and purge the drywell as follows, defeating isolation interlocks if necessary:

If while executing the following steps adequate core cooling is assured and it has been determined that the offsite radioactivity release rate has reached the offsite radioactivity release rate which requires a General Emergency, secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

(1) Vent the drywell.

(2) If the drywell can be vented, purge the drywell by injecting nitrogen into the drywell at the maximum rate.

(3) When no hydrogen is detected in the drywell and either drywell oxygen concentration is below 5% or no hydrogen is detected in the suppression chamber, secure vent and purge

not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

PC/G-3 EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure; vent and purge the drywell as follows, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary:

PC/G-3.1 Vent the drywell.

PC/G-3.2 If the drywell can be vented, purge the drywell by injecting air or nitrogen, whichever will more rapidly return hydrogen concentrations to below 6% or oxygen concentration to below 5%, into the drywell at the maximum rate.

PC/G-4 Reduce suppression chamber hydrogen and oxygen concentrations by one or both of the following methods:

- If the offsite radioactivity release rate is expected to remain below the offsite release rate specified in [Technical Specifications], vent and purge the suppression chamber as follows, defeating isolation interlocks (except high radiation interlocks) if necessary:

If while executing the following steps the offsite radioactivity release rate reaches the offsite release rate specified in [Technical Specifications], secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

(1) Refer to [sampling procedure].

(2) If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

(3) If the suppression chamber can be vented, purge the suppression chamber by injecting nitrogen into the suppression chamber.

(4) When hydrogen is no longer detected in the suppression chamber, secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

PC/G-5 Reduce suppression chamber hydrogen and oxygen concentrations by one or both of the following methods:

- If adequate core cooling is not assured or the offsite radioactivity release rate is expected to remain below the offsite release rate which requires a General Emergency, vent and purge the suppression chamber as follows, defeating isolation interlocks if necessary:

If while executing the following steps adequate core cooling is assured and it has been determined that the offsite radioactivity release rate has reached the offsite release rate which requires a General Emergency, secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

(1) If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

(2) If the suppression chamber can be vented, purge the suppression chamber by injecting nitrogen into the suppression chamber at the maximum rate.

(3) When no hydrogen is detected in the suppression chamber and either suppression chamber oxygen concentration is below 5% or no hydrogen is detected in the drywell, secure vent and purge not required by other steps in the [procedures developed from the Emergency Procedure Guidelines].

PC/G-6 EMERGENCY RPV DEPRESSURIZATION IS REQUIRED; enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure; vent and purge the suppression chamber as follows, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary:

PC/G-6.1 If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

PC/G-6.2 If the suppression chamber can be vented, purge the suppression chamber by injecting air or nitrogen, whichever will more rapidly return hydrogen concentration to below 6% or oxygen concentration to below 5%, into the suppression chamber at the maximum rate.

18A.5 REACTOR BUILDING CONTROL EMERGENCY PROCEDURE GUIDELINE

PURPOSE

The purpose of this guideline is to:

- Protect equipment in the Reactor Building,
- Limit radioactivity release to the Reactor Building,
- and either:
- Maintain Reactor Building integrity, or
- Limit radioactivity release from the Reactor Building.

ENTRY CONDITIONS

The entry conditions for this guideline are any of the following Reactor Building conditions:

- Differential pressure at or above 0 in. of water
- An area temperature above the maximum normal operating temperature
- A HVAC cooler differential temperature above the maximum normal operating differential temperature
- A HVAC exhaust radiation level above the maximum normal operating radiation level
- An area radiation level above the maximum normal operating radiation level
- A floor drain sump water level above the maximum normal operating water level
- An area water level above the maximum normal operating water level

OPERATOR ACTIONS

If while executing the following steps, Primary Containment Flooding is or has been required, enter [procedure developed from the Containment and Radioactivity Release Control Severe Accident Guideline].

If while executing the following steps Reactor Building HVAC exhaust radiation level exceeds [20 mr/hr (Reactor Building HVAC isolation setpoint)]:

- Confirm or manually initiate isolation of Reactor Building HVAC

If while executing the following steps:

- Reactor Building HVAC isolates, and,
- Reactor Building HVAC exhaust radiation level is below [20 mr/hr (Reactor Building HVAC isolation setpoint)],

restart Reactor Building HVAC, defeating high drywell pressure and low RPV water level isolation interlocks if necessary.

Execute [Steps SC/T, SC/R, and SC/L] concurrently.

SC/T Monitor and control Reactor Building temperatures. ❶

SC/T-1 Operate available area coolers.

SC/T-2 If Reactor Building HVAC exhaust radiation level is below [20 mr/hr (Reactor Building HVAC isolation setpoint)], operate available Reactor Building HVAC.

SC/T-3 When an area temperature exceeds its maximum normal operating temperature, isolate all systems that are discharging into the area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Emergency Procedure Guidelines].

Execute [Steps SC/T-4 and SC/T-5] concurrently.

SC/T-4 If a primary system is discharging into Reactor Building:

SC/T-4.1 Before any area temperature reaches its maximum safe operating temperature, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

SC/T-4.2 When an area temperature exceeds its maximum safe operating temperature in more than one area, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

SC/T-5 When an area temperature exceeds its maximum safe operating temperature in more than one area, shut down the reactor.

SC/R Monitor and control Reactor Building radiation levels.

SC/R-1 When an area radiation level exceeds its maximum normal operating radiation level, isolate all systems that are discharging into the area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Emergency Procedure Guidelines].

Execute [Steps SC/R-2 and SC/R-3] concurrently.

SC/R-2 If a primary system is discharging into Reactor Building:

SC/R-2.1 Before any area radiation level reaches its maximum safe operating radiation level, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

SC/R-2.2 When an area radiation level exceeds its maximum safe operating radiation level in more than one area, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

SC/R-3 When an area radiation level exceeds its maximum safe operating radiation level in more than one area, shut down the reactor.

SC/L Monitor and control Reactor Building water levels.

SC/L-1 When a floor drain sump or area water level is above its maximum normal operating water level, operate available sump pumps to restore and maintain it below its maximum normal operating water level.

If any floor drain sump or area water level cannot be restored and maintained below its maximum normal operating water level, isolate all systems that are discharging water into the sump or area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Emergency Procedure Guidelines].

Execute [Steps SC/L-2 and SC/L-3] concurrently.

SC/L-2 If a primary system is discharging into Reactor Building:

SC/L-2.1 Before any area water level reaches its maximum safe operating water level, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute it concurrently with this procedure.

SC/L-2.2 When an area water level exceeds its maximum safe operating water level in more than one area, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED.

SC/L-3 When an area water level exceeds its maximum safe operating water level in more than one area, shut down the reactor.

18A.6 RADIOACTIVITY RELEASE CONTROL EMERGENCY PROCEDURE GUIDELINE

PURPOSE

The purpose of this guideline is to limit radioactivity release into areas outside the primary containment and Reactor Building.

ENTRY CONDITIONS

The entry condition for this guideline is:

- Offsite radioactivity release rate above the offsite release rate which requires an Alert.

OPERATOR ACTIONS

> If while executing the following steps, Primary Containment Flooding is or has been required, enter [procedure developed from the Containment and Radioactivity Release Control Severe Accident Guideline].

> If while executing the following steps HVAC in the turbine building (or any other building which may be contributing to radioactive release) is shutdown, restart the HVAC as required, defeating isolation interlocks if necessary.

RR-1 Isolate all primary systems that are discharging into areas outside the primary and Reactor Buildings except systems required to be operated by [procedures developed from the Emergency Procedure Guidelines].

RR-2 If the offsite radioactivity release rate continues to increase, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC-1] and execute concurrently

RR-3 Before offsite radioactivity release rate reaches the offsite release rate which requires a General Emergency but only if a primary system is discharging into an area outside the primary containment and Reactor Building, EMERGENCY RPV DEPRESSURIZATION IS REQUIRED

18A.7 CONTINGENCY #1 EMERGENCY RPV DEPRESSURIZATION

If while executing the following steps Primary Containment Flooding is or has been required, enter [procedure developed from the RPV and Primary Containment Flooding Severe Accident Guideline].

If while executing the following steps RPV water level cannot be determined, enter [procedure developed from Emergency Procedure Guideline Contingency #2, RPV Flooding"].

If while executing the following steps it is anticipated that primary containment water level will rise above [**m (**ft)(elevation of the inboard main steam line drain valve motor operator or elevation of the lowest SRV pneumatic solenoid, whichever is lower)], open the inboard main steam line drain valve before primary containment water level reaches [**m (**ft)(elevation of the inboard main steam line drain valve motor operator or elevation of the lowest SRV pneumatic solenoid, whichever is lower)], defeating isolation interlocks if necessary.

C1-1 When either:

- All control rods can be determined to be inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)], or
- It has been determined that the reactor will remain shutdown under all conditions without boron, or
- All injection into the RPV, except from boron injection systems and CRD purge flow, has been terminated and prevented,

C1-1.1 Initiate all Isolation Condensers.

C1-1.2 Rapidly depressurize the RPV as follows, irrespective of the resulting cooldown rate:

If suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], open all SRVs, defeating isolation interlocks if necessary.

If less than [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs are open and RPV pressure is at least [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

(1) Rapidly depressurize the RPV using one or more of the following, defeating interlocks and exceeding offsite radioactivity release rate limits if necessary, until RPV pressure is less than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

- Main condenser
- [Other steam-driven equipment]
- MSL drains
- RWCU/SDCS, only if no boron has been injected into the RPV or it has been determined that the reactor will remain shutdown under all conditions without boron.
- Head vent
- IC tube side vent

(2) Maintain RPV pressure less than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure using one or more of the systems used for depressurization.

If while executing the following step: • Less than [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs are open and RPV pressure is at least [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure, or • The reactor is not shutdown, return to [Step C1-1.2].

C1-2 When either:

- All control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)], or
- It has been determined that the reactor will remain shutdown under all conditions without boron, or
- [96.8 Kg (213 pounds) (Cold Shutdown Boron Weight)] of boron have been injected into the RPV, or

- The reactor is shutdown and no boron has been injected into the RPV,

Use the RWCU/SDCS to cool down to cold shutdown conditions [49°C (120°F) <RPV water temp <100°C (212°F) (cold shutdown conditions)]

If shutdown cooling cannot be established and further cool down is required, continue to cool down to cold shutdown conditions [49°C (120°F) <RPV water temp <100°C (212°F)] using one or more of the following, defeating interlocks if necessary:

- IC
- SRVs
- Main condenser
- [Other steam-driven equipment]
- MSL drains
- Head vent
- IC tube side vent

18A.8 CONTINGENCY #2 RPV FLOODING

If while executing the following steps RPV water level can be determined: • If any control rod cannot be determined to be inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] and it has not been determined that the reactor will remain shutdown under all conditions without boron, enter [procedure developed from Emergency Procedure Guideline Contingency #3] and [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC/P]. • If all control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] or it has been determined that the reactor will remain shutdown under all conditions without boron, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Steps RC/L and RC/P].

If while executing the following steps primary containment water level and suppression chamber pressure cannot be maintained below Primary Containment Pressure Limit, but only if RPV flooding conditions can be maintained, terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

If while executing the following steps it is determined that core damage is occurring, PRIMARY CONTAINMENT FLOODING IS REQUIRED; enter [procedure developed from the RPV and Primary Containment Flooding Severe Accident Guideline].

C2-1 If any control rod cannot be determined to be inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] and it has not been determined that the reactor will remain shutdown under all conditions without boron, flood the RPV as follows:

If while executing the following steps either all control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] or it has been determined that the reactor will remain shutdown under all conditions without boron but RPV water level cannot be determined, continue in this procedure at [Step C2-2].

C2-1.1 Terminate and prevent all injection into the RPV except from boron injection systems and CRD purge flow.

C2-1.2 If suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], open all SRVs, irrespective of the resulting RPV cooldown rate, defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary.

> If while executing the following steps it has been determined that the RPV has been flooded to the main steam lines, continue in this procedure at [Step C2-3].

C2-1.3 When RPV pressure is below the Minimum Steam Cooling Pressure or less than [1 (minimum number of SRVs for which the Minimum Steam Cooling Pressure is below the lowest SRV lift setpoint)] SRV[s] [is] open, commence and slowly increase injection into the RPV with the following systems, defeating high RPV water level interlocks if necessary, to establish and maintain at least [1 (minimum number of SRVs for which the Minimum Steam Cooling Pressure is below the lowest SRV lift setpoint)] SRV[s] open and RPV pressure above the Minimum Steam Cooling Pressure but as low as practicable: ❷

Minimum Steam Cooling Pressure

NUMBER OF OPEN SRVs	RPV PRESSURE MPa gauge (psig)
8 or more	0.801 (116.2)
7	0.929 (134.8)
6	1.1.02 (159.8)
5	1.342 (194.7)
4	1.703 (247.0)
3	2.304 (334.2)
2	3.507 (580.7)

- Condensate/Feedwater

- CRD high pressure makeup mode
- FAPCS LPCI mode

If required to open at least [1 (minimum number of SRVs for which the Minimum Steam Cooling Pressure is below the lowest SRV lift setpoint)] SRV[s] or to increase RPV pressure to above the Minimum Steam Cooling Pressure, commence and slowly increase injection with the following systems, defeating high RPV water level interlocks if necessary:

- [Fire System]
- [Interconnections with other units]

If at least [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs can be opened or if RPV pressure can be restored and maintained above the Minimum Steam Cooling Pressure, close the MSIVs, MSL drain valves and the IC isolation valves, and isolate any of the following not needed for RPV injection:

- Main steam lines
- MSL drains
- Isolation Condensers

If less than [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs can be opened and RPV pressure is at least [0.345MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure but cannot be restored and maintained above the Minimum Steam Cooling Pressure, rapidly depressurize the RPV using one or more of the following, irrespective of the resulting cooldown rate, defeating interlocks and exceeding offsite radioactivity release rate limits if necessary, until RPV pressure can be restored and maintained above the Minimum Steam Cooling Pressure or is less than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

- Isolation Condensers
- Main condenser
- [Other steam-driven equipment]
- MSL drains
- Head vent
- IC tube side vent
- RWCU/SDC

C2-2 When all control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] or it has been determined that the reactor will remain shutdown under all conditions without boron, flood the RPV as follows:

C2-2.1 If suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], open all SRVs, irrespective of the resulting RPV cooldown rate, defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary.

C2-2.2 If at least [2 (Minimum Number of SRVs Required for Decay Heat Removal)] SRVs can be opened or if CRD high pressure makeup or a feedwater pump is available, close the MSIVs, MSL drain valves, and the IC isolation valves.

If while executing the following step it can be determined that the RPV has been flooded to the main steam lines, continue in this procedure at [Step C2-3].

C2-2.3 Flood the RPV to the elevation of the main steam lines with the follo systems, defeating high RPV water level interlocks if necessary:

- Feedwater pumps
- Condensate pumps
- CRD high pressure makeup mode
- FAPCS LPCI mode
- [Interconnections with other units]
- [Fire System]

If:

- Less than [2 (Minimum Number of SRVs Required for Decay Heat Removal)] SRVs can be opened, and
- Neither CRD high pressure makeup nor a feedwater pump is available, and
- RPV pressure is more than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure,

Rapidly depressurize the RPV using one or more of the following, irrespective of the resulting cooldown rate, defeating interlocks and exceeding offsite radioactivity release rate limits if necessary, until RPV pressure is less than

[0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

- Isolation Condensers
- Main condenser
- [Other steam-driven equipment]
- MSL drains
- Head vent
- IC tube side vent
- RWCU /SDC

C2-3 When it has been determined that the RPV has been flooded to the main steam lines:

- Close the MSIVs, MSL drain valves, and the IC isolation valves.
- Control injection into the RPV to maintain the steam lines flooded with injection as low as practicable.

18A.9 CONTINGENCY #3 LEVEL/POWER CONTROL

If while executing the following steps:

- RPV water level cannot be determined, enter [procedure developed from Emergency Procedure Guideline Contingency #2].
- All control rods are inserted to or beyond position [4.2% (Maximum Subcritical Banked Withdrawal Position)] or it has been determined that the reactor will remain shutdown under all conditions without boron, enter [procedure developed from the RPV Control Emergency Procedure Guideline] at [Step RC/L].
- Primary containment water level and suppression chamber pressure cannot be maintained below Primary Containment Pressure Limit, but only if adequate core cooling can be assured, terminate injection into the RPV from sources external to the primary containment except from systems required to shut down the reactor.

C3-1 If any MSL is not isolated, commence defeating [MSL and Offgas high radiation interlocks] [and] [low RPV water level interlocks] to maintain the main condenser as a heat sink.

C3-2 If reactor power is above [6% (APRM downscale trip)] or cannot be determined and RPV water level is above [1830.5 cm (720.7 in) (60.96 cm (24 in) below the feedwater sparger nozzles)], lower RPV water level to below [1830.5 cm (720.7 in) (60.96 cm (24 in) below the feedwater sparger nozzles)] by terminating and preventing all injection into the RPV except from boron injection systems and CRD purge flow, defeating interlocks if necessary.

C3-3 If:

- Reactor power is above [6% (APRM downscale trip)] or cannot be determined, and
- RPV water level is above [745.3 cm (293.4 in) (top of active fuel)], and
- Suppression pool temperature is above the Boron Injection Initiation Temperature, and
- Either an SRV is open or opens or drywell pressure is above [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)],

Lower RPV water level, irrespective of any reactor power or RPV water level oscillations, by terminating and preventing all injection into the RPV except from boron injection systems and CRD purge flow, defeating interlocks if necessary, until either:

- Reactor power drops below [6% (APRM downscale trip)], or

- RPV water level reaches [745.3 cm (293.4 in) (top of active fuel)], or
- All SRVs remain closed and drywell pressure remains below [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)].

> If while executing the following steps Emergency RPV Depressurization is required, continue in this procedure at [Step C3-4.1].

> If while executing the following step RPV water level is above [1830.5 cm (720.7 in) (60.96 cm (24 in) below the feedwater sparger nozzles)] and reactor power is above [6% (APRM downscale trip)] or cannot be determined, return to [Step C3-2].

> If while executing the following step:
>
> - Reactor power is above [6% (APRM downscale trip)] or cannot be determined, and
> - RPV water level is above [745.3 cm (293.4 in) (top of active fuel)], and
> - Suppression pool temperature is above the Boron Injection Initiation Temperature, and
> - Either an SRV is open or opens or drywell pressure is above [0.01264 MPa (1.83 psig) (high drywell pressure scram setpoint)],
>
> return to [Step C3-2].

C3-4 Maintain RPV water level between [745.3 cm (293.4 in) (top of active fuel)] and either: ❷

- If RPV water level was deliberately lowered in [Step C3-2 or C3-3], the level to which it was lowered, or
- If RPV water level was not deliberately lowered in [Step C3-2 or C3-3], [2189 cm (861.8 in) (high level trip setpoint)],

with the following systems:

- Condensate/Feedwater
- CRD (either purge flow or high pressure makeup modes)
- FAPCS LPCI mode.

If RPV water level cannot be restored and maintained above [745.3 cm (293.4 in) (top of active fuel)], EMERGENCY RPV DEPRESSURIZATION IS REQUIRED:

C3-4.1 Terminate and prevent all injection into the RPV except from boron injection systems and CRD purge flow, defeating interlocks if necessary, until RPV pressure is below the Minimum Steam Cooling Pressure.

Minimum Steam Cooling Pressure

NUMBER OF OPEN SRVs	RPV PRESSURE MPa gauge (psig)
8 or more	0.801 (116.2)
7	0.929 (134.8)
6	1.1.02 (159.8)
5	1.342 (194.7)
4	1.703 (247.0)
3	2.304 (334.2)
2	3.507 (580.7)
1	7.116 (1032.1)

If less than [1 (minimum number of SRVs for which the Minimum Steam Cooling Pressure is below the lowest SRV lift setpoint)] SRV[s] can be opened, continue in this procedure.

C3-4.2 Commence and slowly increase injection into the RPV with the following systems to restore and maintain RPV water level above [745.3 cm (293.4 in) (top of active fuel)]: ❷

- Condensate/Feedwater
- CRD (either purge flow or high pressure makeup modes)
- FAPCS LPCI mode

If required to restore and maintain RPV water level above [745.3 cm (293.4 in) (top of active fuel)], use the following systems:

- Fire System
- [Interconnections with other units]

If RPV water level cannot be restored and maintained above [745.3 cm (293.4 in) (top of active fuel)], PRIMARY CONTAINMENT FLOODING IS REQUIRED; enter [procedure developed from the RPV and Primary Containment Flooding Severe Accident Guideline].

C3-4.3 When RPV water level can be maintained above [745.3 cm (293.4 in) (top of active fuel)], return to [Step C3-4].

18A.10 RPV AND PRIMARY CONTAINMENT FLOODING SEVERE ACCIDENT GUIDELINE

PURPOSE

The purpose of this guideline is to:

- Submerge the core (and in-vessel core debris)
- Shut down the reactor
- Depressurize the RPV and prevent it from repressurizing.
- Submerge core debris on the lower drywell floor

ENTRY CONDITIONS

This guideline is entered whenever Primary Containment Flooding is required.

OPERATOR ACTIONS

Execute [Steps RC/F, RC/P, and RC/Q] concurrently.

RC/F Monitor and control RPV and primary containment water levels ❶❸

.

If while executing the following steps: • Drywell sprays have been initiated; terminate drywell sprays before drywell pressure drops to 0 psig or if lower drywell thermocouples do not indicate the presence of core debris on the lower drywell floor.

If while executing the following steps: • DPVs have not opened, open all DPVs

- GDCS has not initiated, initiate GDCS and open all suppression pool equalization line Squib Valves (SQVs).

Flood the primary containment as follows:

If it has been determined that core debris has breached the RPV, proceed to step RC/F-1.

If it has been determined that core debris has not breached the RPV and RPV water level can be restored to [745.3 cm (293.4 in) (top of active fuel)], proceed to step RC/F-2

RC/F-1 IT HAS BEEN DETERMINED THAT CORE DEBRIS HAS BREACHED THE RPV

If while executing the following step:

- DPVs have not opened, open all DPVs
- GDCS has not initiated, initiate GDCS and open all suppression pool equalization line SQVs,
- Lower drywell thermocouples indicate core debris on the drywell floor and the GDCS Deluge Valves have not actuated, actuate GDCS Deluge Valves.

If while executing the following step primary containment water level and suppression chamber pressure approach or exceed the Primary Containment Pressure Limit, vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rates if necessary, to maintain suppression chamber pressure below the Primary

Containment Pressure Limit. If primary containment water level and suppression chamber pressure cannot be maintained below the Primary Containment Pressure Limit, but only if total injection into the RPV and drywell can be maintained greater than the Minimum Debris Retention Injection Rate, terminate injection into the RPV and primary containment from sources external to the primary containment except drywell sprays.

If while executing the following step:

- Drywell sprays are not operating, initiate drywell sprays, defeating drywell spray interlocks if necessary and irrespective of the Drywell Spray Initiation Limit or whether RPV or primary containment injection will be reduced. Use sources external to the primary containment if possible.
- Drywell sprays are operating, maintain drywell spray flowrate greater than [3840 gpm (Minimum Drywell Spray Flow)], defeating drywell spray interlocks if necessary and irrespective of whether RPV or primary containment injection will be reduced. Use sources external to the primary containment if possible.

RC/F-1.1 Flood the primary containment as follows:

Maximize injection into the RPV from sources external to the primary containment using the following systems, defeating interlocks if necessary:

- Feedwater
- Condensate
- CRD High Pressure Makeup mode
- FAPCS LPCI mode
- Fire Protection System
- [Interconnections with other units]
- SLC
- [Other primary containment fill systems]

If injection into the RPV from sources external to the primary containment will not be reduced, maximize injection into the primary containment from sources external to the primary containment.

If injection into neither the RPV nor the primary containment from sources external to the primary containment will be reduced, maximize injection into the RPV from the suppression pool.

If venting the primary containment will facilitate primary containment flooding, vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary.

If while executing the following step primary containment water level and suppression chamber pressure cannot be maintained below the Primary Containment Pressure Limit restore and maintain primary containment water level and suppression chamber pressure below the Primary Containment Pressure Limit by one or both of the following methods:

- Vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary.
- Terminate injection into the RPV and primary containment from sources external to the primary containment except drywell sprays.

If while executing the following step:

- Drywell sprays are not operating and [suppression pool water level is below [16.85 m (55.28 ft) (elevation of bottom of suppression chamber to drywell vacuum breaker openings [less vacuum breaker opening pressure in feet of water])] and] drywell temperature is below the Drywell Spray Initiation Limit, initiate drywell sprays, defeating drywell spray interlocks if necessary and irrespective of whether injection into the RPV will be reduced. Sources external to the primary containment may be used only if primary containment water level and suppression chamber pressure can be restored and maintained below the Primary Containment Pressure Limit.

- Drywell sprays are operating, continue to operate drywell sprays, defeating drywell spray interlocks if necessary and irrespective of whether injection into the RPV will be reduced. Sources external to the primary containment may be used only if primary containment water level and suppression chamber pressure can be restored and maintained below Primary Containment Pressure Limit.

RC/F-1.2 When primary containment water level reaches [7.6 m (25 ft). (Minimum Debris Submergence Level)]:

Maintain primary containment water level between [7.6 m (25 ft) (Minimum Debris Submergence Level)] and [31.2 m (103 ft) (elevation of primary containment vent)] using the following systems, defeating interlocks if necessary:

- Feedwater
- Condensate
- CRD High Pressure Makeup mode
- FAPCS LPCI mode
- Fire Protection System
- [Interconnections with other units]
- SLC
- [Other primary containment fill systems]

If necessary to maintain primary containment water level above [7.6 m (25 ft). (Minimum Debris Submergence Level)], vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary.

RC/F-2 CORE DEBRIS HAS NOT BREACHED THE RPV AND RPV WATER LEVEL CAN BE RESTORED ABOVE [745.3 cm (293.4 in) (TOP OF ACTIVE FUEL)]

If while executing the following step:

- DPVs have not opened, open all DPVs
- GDCS has not initiated, initiate GDCS and open all suppression pool equalization line SQVs,

If while executing the following step, water level cannot be maintained above [745.3 cm (293.4 in)(top of active fuel)] then maintain water level as high as can be achieved with the available systems and maximize injection into the RPV and primary containment from sources external to the primary containment.

If while executing this step primary containment water level and suppression chamber pressure approach or exceed the Primary Containment Pressure Limit:

(1) Terminate direct injection into the primary containment from sources external to the primary containment.

(2) If primary containment water level and suppression chamber pressure cannot be maintained below the Primary Containment Pressure Limit:

- Vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary, to restore and maintain suppression chamber pressure below the Primary Containment Pressure Limit.

- If primary containment water level and suppression chamber pressure reach the Primary Containment Pressure Limit, but only if RPV water level can be maintained above [745.3 cm (293.4 in) (top of active fuel)], terminate injection into the RPV from sources external to the primary containment, except boron injection from SLC.

Restore and maintain RPV water level between [745.3 cm (293.4 in) (top of active fuel)] and [2189 cm (861.8 in) (high level trip setpoint)] using the following systems, taking suction from sources external to the primary containment only when required, defeating interlocks if necessary:

- Feedwater
- Condensate
- CRD High Pressure Makeup mode
- FAPCS LPCI mode

If required to restore and maintain RPV water level above [745.3 cm (293.4 in) (top of active fuel)], use the following systems, taking suction from sources external to the primary containment only when required, defeating interlocks if necessary:

- Fire Protection System
- [Interconnections with other units]
- SLC
- [Other primary containment fill systems]

If necessary to restore and maintain RPV water level above [745.3 cm (293.4 in) (top of active fuel)]:

- Vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary.
- If no DPV can be opened, vent the RPV with one or more of the following, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary:
  - [MSIVs
  - MSL drains
  - IC tube side vents]

RC/P Monitor and control RPV pressure.

> If while executing the following steps it is anticipated that primary containment water level will rise above [**m (**ft) (elevation of the inboard main steam line drain valve motor operator or elevation of the lowest SRV pneumatic solenoid, whichever is lower)], open the inboard main steam line drain valve before primary containment water level reaches [**m (**ft) (elevation of the inboard main steam line drain valve motor operator or elevation of the lowest SRV pneumatic solenoid, whichever is lower)], defeating isolation interlocks if necessary.

RC/P-1 Initiate all Isolation Condensers.

RC/P-2 Rapidly depressurize the RPV as follows, irrespective of the resulting cooldown rate:

If suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], open all ADS SRVs and DPVs, defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary.

If any ADS valve cannot be opened, but only if suppression pool water level is above [2.13 m (6.99 ft) (elevation of top of SRV discharge device)], open other SRVs, defeating pneumatic supply isolation interlocks and restoring the pneumatic supply if necessary, until [10 (number of SRVs dedicated to ADS)] valves are open.

If DPVs did not open and less than [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs are open and RPV pressure is at least

[0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

(1) Rapidly depressurize the RPV using non-ADS SRVs and one or more of the following, defeating interlocks and exceeding offsite radioactivity release rate limits if necessary, until RPV pressure is less than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure:

- Main condenser
- [Other steam-driven equipment]
- MSL drains
- Head vent
- IC tube side vent
- RWCU/SDCS, only if no boron has been injected into the RPV or it has been determined that the reactor will remain shutdown under all conditions without boron.

(2) Maintain RPV pressure less than [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure using one or more of the systems used for depressurization.

> If while executing the following step:
>
> - Less than [8 (Minimum Number of SRVs Required for Emergency Depressurization)] SRVs are open and RPV pressure is at least [0.345 MPa (50 psig) (Decay Heat Removal Pressure)] above suppression chamber pressure, or
> - The reactor is not shutdown,
>
> return to [Step RC/P-2].

RC/P-3 If further cooldown is required, continue to cool down to cold shutdown conditions ((120°F) 49°C <RPV water temp < 100°C (212°F) (cold shutdown conditions)]) using one or more of the following, defeating interlocks if necessary:

- ICs

- RWCU/SDC
- SRVs
- Main condenser
- [Other steam-driven equipment]
- MSL drains
- Head vent
- IC tube side vent

RC/Q Monitor and control reactor power.

RC/Q-1 [Confirm or place the reactor mode switch in SHUTDOWN.]

RC/Q-2 If ARI has not initiated, initiate ARI.

Execute [Steps RC/Q-3 and RC/Q-4] concurrently.

RC/Q-3 Inject boron into the RPV with SLC

RC/Q-4 Insert control rods as follows until it has been determined that core debris has breached the RPV:

RC/Q-4.1 Reset ARI, defeating ARI logic trips if necessary.

RC/Q-4.2 Insert control rods with one or more of the following methods:

- De-energize scram solenoids.
- Vent the scram air header.
- Reset the scram, defeating RPS logic trips if necessary, drain the scram discharge volume, and initiate a manual scram.
- Open individual scram test switches.
- Drive control rods, defeating RC&IS and RWM interlocks if necessary.

18A.11 CONTAINMENT AND RADIOACTIVITY RELEASE SEVERE ACCIDENT GUIDELINE

PURPOSE

The purpose of this guideline is to:

- Protect equipment in the primary containment and Reactor Building,
- Maintain primary containment and Reactor Building integrity, and
- Limit radioactivity release into areas outside the primary containment and Reactor Building.

ENTRY CONDITIONS

This guideline is entered whenever Primary Containment Flooding is required.

OPERATOR ACTIONS

If while executing the following steps Reactor Building HVAC exhaust radiation level exceeds [20 mr/hr (Reactor Building HVAC isolation setpoint)]:

- Confirm or manually initiate isolation of Reactor Building HVAC.

If while executing the following steps:

- Reactor Building HVAC isolates, and,
- Reactor Building HVAC exhaust radiation level is below [20 mr/hr (Reactor Building HVAC isolation setpoint)],

restart Reactor Building HVAC, defeating high drywell pressure and low RPV water level isolation interlocks if necessary.

If while executing the following steps HVAC in the turbine building (or any other building which may be contributing to radioactive release) is shutdown [or isolated due to high radiation], restart the HVAC as required, defeating isolation interlocks if necessary.

Execute [Steps SP/T, DW/T, PC/P, PC/R, PC/G, SC/T, SC/R, SC/L, and RR] concurrently.

SP/T Monitor and control suppression pool temperature.

If injection into neither the RPV nor the primary containment will be reduced or if RPV water level can be maintained above [745.3 cm (293.4 in) (top of active fuel)], control suppression pool temperature below [43.3°C (110°F) (most limiting suppression pool temperature LCO)] using available suppression pool cooling.

DW/T Monitor and control drywell temperature. ❶❸

Control drywell temperature below [57.22°C (135°F) (drywell temperature LCO or maximum normal operating temperature, whichever is higher)] using available drywell cooling, defeating isolation interlocks if necessary.

Before drywell temperature reaches [171.1°C (340°F) (maximum temperature at which ADS is qualified or drywell design temperature, whichever is lower)], drywell spray can be utilized if drywell water level is less than [0.7 m] or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than [0.7 m] and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized.

PC/P Monitor and control primary containment pressure. ❶❸

When suppression chamber pressure exceeds [0.095 MPa (13.8 psig) (Drywell Spray Initiation Pressure)] drywell spray can be utilized if drywell water level is less than [0.7 m] or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than [0.7 m] and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized.
Before suppression chamber pressure reaches the Primary Containment Pressure Limit, vent the primary containment, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary, to control pressure below the Primary Containment Pressure Limit.

PC/R Monitor and control [drywell] [suppression chamber] radiation level. ❸

Before [drywell] [suppression chamber] radiation level reaches [14,000 R/hr (drywell or suppression chamber radiation level which requires a General Emergency)], drywell spray can be utilized if drywell water level is less than [0.7 m] or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than [0.7 m] and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized.

PC/G Monitor and control primary containment hydrogen and oxygen concentrations.

> If while executing the following steps the hydrogen or oxygen monitoring system is or becomes unavailable, sample the drywell and suppression chamber for hydrogen and oxygen in accordance with [sampling procedure].

Control hydrogen and oxygen concentrations in the drywell as follows:

<table>
<tr><td colspan="2" rowspan="4"></td><td colspan="4">Drywell Oxygen Concentration</td></tr>
<tr><td rowspan="3">< 5%</td><td colspan="3">≥ 5% or cannot be determined to be below 5%</td></tr>
<tr><td colspan="3">Suppression Chamber Hydrogen Concentration</td></tr>
<tr><td>None Detected</td><td>< 6%</td><td>≥ 6% or cannot be determined to be below 6%</td></tr>
<tr><td rowspan="3">Drywell Hydrogen Concentration</td><td>None Detected</td><td>No action required</td><td>No action required</td><td rowspan="2">[PC/G-2]</td><td rowspan="3">[PC/G-3]</td></tr>
<tr><td>< 6%</td><td rowspan="2">[PC/G-1]</td><td></td></tr>
<tr><td>≥ 6% or cannot be determined to be below 6%</td><td colspan="2"></td></tr>
</table>

Control hydrogen and oxygen concentrations in the suppression chamber as follows:

<table>
<tr><td colspan="2"></td><td colspan="4">Suppression Chamber Oxygen Concentration</td></tr>
<tr><td colspan="2" rowspan="3"></td><td rowspan="3">< 5%</td><td colspan="3">≥ 5% or cannot be determined to be below 5%</td></tr>
<tr><td colspan="3">Drywell Hydrogen Concentration</td></tr>
<tr><td>None Detected</td><td>< 6%</td><td>≥ 6% or cannot be determined to be below 6%</td></tr>
<tr><td rowspan="3">Suppression Chamber Hydrogen Concentration</td><td>None Detected</td><td>No action required</td><td>No action required</td><td rowspan="2">[PC/G-5]</td><td rowspan="3">[PC/G-6]</td></tr>
<tr><td>< 6%</td><td rowspan="2">[PC/G-4]</td><td></td></tr>
<tr><td>≥ 6% or cannot be determined to be below 6%</td><td colspan="2"></td></tr>
</table>

PC/G-1 Reduce drywell hydrogen and oxygen concentrations by one or both of the following methods:

- If the offsite radioactivity release rate is expected to remain below the offsite release rate specified in [Technical Specifications], vent and purge the drywell as follows, defeating isolation interlocks (except high radiation interlocks) if necessary:

> If while executing the following steps the offsite radioactivity release rate reaches the offsite release rate specified in [Technical Specifications], secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

(1) Refer to [sampling procedure].

(2) Vent the drywell.

(3) If the drywell can be vented, purge the drywell by injecting nitrogen into the drywell.

(4) When hydrogen is no longer detected in the drywell, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

PC/G-2 Reduce drywell hydrogen and oxygen concentrations by one or both of the following methods:

- If RPV water level cannot be maintained above [745.3 cm (293.4 in) (top of active fuel)] or the offsite radioactivity release rate is expected to remain below the offsite radioactivity release rate which requires a General Emergency, vent and purge the drywell as follows, defeating isolation interlocks if necessary:

> If while executing the following RPV water level can be maintained above [745.3 cm (293.4 in)(top of active fuel)] and it has been determined that the offsite radioactivity release rate has reached the offsite radioactivity release rate which requires a General Emergency, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

(1) Vent the drywell.

(2) If the drywell can be vented, purge the drywell by injecting nitrogen into the drywell at the maximum rate.

(3) When no hydrogen is detected in the drywell and either drywell oxygen concentration is below 5% or no hydrogen is detected in the suppression chamber, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

PC/G-3 Vent and purge the drywell as follows, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary:

PC/G-3.1 Vent the drywell.

PC/G-3.2 If the drywell can be vented, purge the drywell by injecting air or nitrogen, whichever will more rapidly return hydrogen concentrations to below 6% or oxygen concentration to below 5%, into the drywell at the maximum rate.

PC/G-3.3 If drywell water level is less than [0.7 m] or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor, drywell sprays can be utilized. If drywell water level is greater than [0.7 m] and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized. ❸

PC/G-4 Reduce suppression chamber hydrogen and oxygen concentrations by one or both of the following methods:

- If the offsite radioactivity release rate is expected to remain below the offsite release rate specified in [Technical Specifications], vent and purge the suppression chamber as follows, defeating isolation interlocks (except high radiation interlocks) if necessary:

If while executing the following steps the offsite radioactivity release rate reaches the offsite release rate specified in [Technical Specifications], secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

(1) Refer to [sampling procedure].

(2) If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

(3) If the suppression chamber can be vented, purge the suppression chamber by injecting nitrogen into the suppression chamber.

(4) When hydrogen is no longer detected in the suppression chamber, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

PC/G-5 Reduce suppression chamber hydrogen and oxygen concentrations by one or both of the following methods:

- If RPV water level cannot be maintained above [745.3 cm (293.4 in) (top of active fuel)] or the offsite radioactivity release rate is expected to remain below the offsite release rate which requires a General Emergency, vent and purge the suppression chamber as follows, defeating isolation interlocks if necessary:

If while executing the following steps RPV water level can be maintained above [745.3 cm (293.4 in)(top of active fuel)] and it has been determined that the offsite radioactivity release rate has reached the offsite release rate which requires a General Emergency, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

(1) If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

(2) If the suppression chamber can be vented, purge the suppression chamber by injecting nitrogen into the suppression chamber at the maximum rate.

(3) When no hydrogen is detected in the suppression chamber and either suppression chamber oxygen concentration is below 5% or no hydrogen is detected in the drywell, secure vent and purge not required by other steps in the [procedures developed from the Severe Accident Guidelines].

PC/G-6 Vent and purge the suppression chamber as follows, defeating isolation interlocks and exceeding offsite radioactivity release rate limits if necessary:

PC/G-6.1 If suppression pool water level is below [** m (** ft) (elevation of the bottom of the suppression chamber vent)], vent the suppression chamber.

PC/G-6.2 If the suppression chamber can be vented, purge the suppression chamber by injecting air or nitrogen, whichever will more rapidly return hydrogen concentration to below 6% or oxygen concentration to below 5%, into the suppression chamber at the maximum rate.

SC/T Monitor and control Reactor Building temperatures. ❶

Operate available area coolers.

If Reactor Building HVAC exhaust radiation level is below [20 mr/hr (Reactor Building HVAC isolation setpoint)], operate available Reactor Building HVAC.

When an area temperature exceeds its maximum normal operating temperature, isolate all systems that are discharging into the area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Severe Accident Guidelines].

SC/R Monitor and control Reactor Building radiation levels.

When an area radiation level exceeds its maximum normal operating radiation level, isolate all systems that are discharging into the area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Severe Accident Guidelines].

SC/L Monitor and control Reactor Building water levels.

When a floor drain sump or area water level is above its maximum normal operating water level, operate available sump pumps to restore and maintain it below its maximum normal operating water level.

If any floor drain sump or area water level cannot be restored and maintained below its maximum normal operating water level, isolate all systems that are discharging water into the sump or area except systems required to suppress a fire and systems required to be operated by [procedures developed from the Severe Accident Guidelines].

RR Isolate all primary systems that are discharging into areas outside the primary containment and Reactor Building except systems required to be operated by [procedures developed from the Severe Accident Guidelines].

Table 18A-1

EPG Abbreviations

ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
CRD	Control Rod Drive
DPV	Depressurization Valve
ECCS	Emergency Core Cooling Systems
FAPCS	Fuel and Auxiliary Pool Cooling System
GDCS	Gravity Driven Cooling System
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilating and Air Conditioning
IC	Isolation Condenser
LCO	Limiting Condition for Operation
LPCI	Low Pressure Coolant Injection
MSIV	Main Steamline Isolation Valve
RCIS	Rod Control and Information System
RCIC	Reactor Core Isolation Cooling System
RHR	Residual Heat Removal System
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWM	Rod Worth Minimizer
SLC	Standby Liquid Control
SQV	Squib Valve
SRV	Safety Relief Valve

Table 18A-2

Reactor Building Temperature Operating Values for EPGs and SAGs

Area	Maximum Normal Temperature (°C)(°F)	Maximum Safe Temperature (°C)(°F)
RWCU A Pump Room	*	*
RWCU B Pump Room	*	*
RWCU HX Room	*	*
RWCU HX Room at Hotwell Discharge	*	*
Main Steam Tunnel	*	*
Reactor Area A	*	*
Reactor Area B	*	*
	Maximum Normal delta T (°C)(°F)	**Maximum Safe delta T** (°C)(°F)
HVAC Cooler	*	*
RWCU A Pump Room	*	*
RWCU B Pump Room	*	*
RWCU HX Room at HX	*	*
RWCU HX Room at Hotwell Discharge	*	*
Main Steam Tunnel Cooler	*	*
* Plant Specific Data – COL Applicant		

Table 18A-3

Reactor Building Radiation Level Operating Values for EPGs and SAGs

Area	**Maximum Normal Radiation Level** (Sv)(Rem)	**Maximum Safe Rad Level** (Sv)(Rem)
Equipment Hatch	*	*
Drywell Personnel Lock	*	*
Clean-up System Room	*	*
Clean-up Filter Demin Area	*	*
SLC Room	*	*
Chem Lab	*	*
Fuel Pool HX	*	*
Refueling Floor Laydown Area	*	*
Refueling Floor Change Area HI	*	*
HVAC Exhaust	*	*
Safety Envelope	*	*
Refuel Floor	*	*
* Plant Specific Data – COL Applicant		

Table 18A-4

Reactor Building Water Level Operating Values for EPGs and SAGs

Area	**Maximum Normal Water Level** (m)(ft)	**Maximum Safe Water Level** (m)(ft)
Floor drain sump	*	NA
CRD Compartment	*	*
RB NE Corner Room	*	*
RB SE Corner Room	*	*
* Plant Specific Data – COL Applicant		

18B. ESBWR EPG/SAG COMPARED TO GENERIC BWR EPG

The ESBWR design has incorporated many desirable features and systems characteristic of earlier generation BWRs. Some common BWR systems have been eliminated or modified and some unique systems and configurations have been added to improve safety or operational characteristics in the ESBWR.

The purpose of this document is to provide information about the ESBWR design philosophy and features relative to other BWRs in limited, but sufficient detail to explain the rationale behind the adaptation of the generic EPG/SAG for use in the ESBWR. BWROG EPG/SAG Rev. 2 is used as the standard for comparison.

Section 18B.1 contains summary descriptions of ESBWR systems or features requiring changes from the generic EPG/SAG. Section 18B.2 describes the major changes to the ESBWR EPG/SAG and the bases for these changes. Section 18B.3 compares the ESBWR EPG/SAG step-by-step to EPG/SAG Rev. 2 and documents the basis for each modification made.

18B.1 ESBWR DESIGN FEATURES AFFECTING THE EPG/SAG

The ESBWR design is described elsewhere in detail, but a brief summary of those features, which have had significant impact on the development of the ESBWR EPG/SAG is described below.

18B.1.1 ESBWR RPV and Related Features

The ESBWR is a natural circulation plant and has neither jet pumps nor external or internal recirculation pumps. Thus, there will be no operator actions specified regarding control of recirculation flow. The reactor core is shorter in height than most of the earlier BWR designs and the reactor pressure vessel (RPV) is much taller, resulting in a much larger vessel volume to core power ratio. As a result, level transients would be slower allowing more time for operator recovery actions. The ESBWR also has fine motion control rod drives, so specific operator actions to manually scram the reactor are different from those for the hydraulic locking-piston drives in earlier BWRs. The CRD system in the ESBWR can be operated in the usual (purge) mode or can inject flow into the RPV via the feedwater line (high pressure makeup mode). Where appropriate, a distinction between modes is made in the ESBWR EPG/SAG.

18B.1.2 Isolation Condenser

Most operating plants have no isolation condenser and the relative capacities of those plants that have isolation condensers are much smaller than the ESBWR design. Thus, the ESBWR isolation condensers play a larger role for reactor pressure and level control in the ESBWR.

18B.1.3 Emergency Core Cooling Systems

Operating BWRs typically have a full compliment of automatically initiated Emergency Core Cooling Systems (ECCS) for both high and low pressure injection, plus an ADS for vessel depressurization. High-pressure systems are typically steam driven (HPCI or RCIC). ESBWR has no steam driven or other high pressure ECC related injection systems.

In contrast, ESBWR has the capacity to gravity reflood the RPV using the ADS and the Gravity Driven Cooling System (GDCS). These two systems comprise the ECCS for ESBWR. The function of ADS is to depressurize the reactor pressure vessel (RPV) sufficiently so that the GDCS can re-flood at very low RPV pressure. Gravity re-flooding of the RPV can be accomplished by draining the GDCS pools and by draining the suppression pool through the equalizer line. The RPV must be at essentially the same pressure as the containment for gravity re-flooding to be accomplished.

The ADS in ESBWR consists of 10 SRVs piped to the suppression pool and a set of 8 depressurization valves (DPVs) which discharge directly to the drywell. ESBWR ADS has no start time delay timer, but stages first the SRV openings in two groups of five with a delay in between; then after a further time delay, stages the DPV openings in groups with time delays between the groups.

Significant variations in EPG/SAG strategy arise from these basic ECC concept differences.

18B.1.4 ATWS Mitigation Systems

Operating plant SLC systems are typically pump driven with borated water taken from a supply tank. The ESBWR SLC system is accumulator driven, necessitating that somewhat different directions be specified to the operator. ESBWR has an automatic ADS inhibit feature in contrast to earlier BWRs where manual action is required to inhibit, if needed. In contrast, ESBWR has no manual inhibit feature.

18B.1.5 Containment Features

ESBWR has a unique natural circulation driven Passive Containment Cooling System (PCCS). This system condenses steam in the containment and has a vent for transport of non-condensable gases to the suppression pool which is driven by any pressure difference between the drywell and suppression pool. There are no valves or any other device requiring activation and thus no operator actions are specified in the EPG/SAG. The PCCS is mentioned here because of its inherent ability to control containment pressure and remove energy and its high likelihood of being in operation during any event without operator action.

18B.2 MAJOR DIFFERENCE BETWEEN ESBWR AND BWROG EPG/SAG REV. 2

The major strategy differences between ESBWR EPG/SAG and BWROG Rev. 2 EPG/SAG arise from the fundamental design differences described below. It is recognized that many systems would be initiated automatically and may be in operation during a given transient. However, when it is time for a system to be placed into operation during the EPG/SAG evolution, direction is given to operate the system in case it did not initiate on an automatic signal, or if the system was prevented from operation or bypassed earlier by the operator. This is the same approach generally used in BWROG EPG/SAG Rev. 2, but is noted here as a reminder because of the many automatically initiated systems in ESBWR.

18B.2.1 Level Control

The operator is directed to depressurize the RPV manually using SRVs and attempt to restore level in the event of a failure of the automatic ADS initiation on Level 1.5. If level reaches the top of the active fuel, then direction is given to enter the RPV and Containment Flooding procedure and initiate DPVs and actuate the GDCS. This provides some additional time to allow for recovery. Normally, however, the ADS would initiate automatically on Level 1.5. The Isolation Condensers will be automatically (or manually) initiated and will remove decay heat and return condensate to the RPV.

In the ESBWR EPG/SAG, Level 1.5 is chosen as the control level if RPV water level cannot be restored and maintained in the preferred range (low level scram to high level trip). Level 1.5 is chosen primarily because the GDCS is automatically initiated at Level 1.5. If Level 1.5 cannot be maintained and the GDCS fails to initiate, a further effort is provided to maintain level at top-of-active-fuel (TAF). This control method is not consistent with the BWROG EPG/SAG Rev 2, Contingency 1, Alternate Level Control, control strategy and has thus been eliminated from the ESBWR EPG/SAG. If level cannot be maintained at TAF, then the RPV and Containment Flooding procedure is entered.

ESBWR has a very limited number of ECC injection systems, depending primarily on the alternate strategy of depressurization and gravity re-flooding, thus there is no need for pump lineup and starting as directed in Contingency 1 of BWROG EPG/SAG Rev. 2. The emergency depressurization action and the final measure of containment flooding in BWROG EPG/SAG Rev. 2, Contingency 1 are incorporated in the ESBWR EPG/SAG in the level control section.

If level cannot be maintained above the preferred control point (low level scram set point or shutdown cooling RPV water level interlock, whichever is higher), then emergency depressurization is required. Once emergency depressurization has been started, the operator is directed to again attempt level recovery using the available injection systems at lower pressure.

At this point, if level cannot be maintained above TAF at the lower RPV pressure, entry into RPV and Containment Flooding is directed. The first action is to initiate GDCS.

RPV Flooding also calls for use of DPVs and GDCS. Initiation of DPVs will assure complete blow down of the RPV so that gravity re-flooding can be achieved. GDCS first drains the GDCS pools, which likely would be adequate to cover the core. The operator is also instructed to open the squib valves (SQVs) in the equalizer lines to flood the RPV from the suppression pool.

Level control steps and re-flood evolutions similar to those described above in RPV level control have been incorporated into Contingency 3, Level/Power Control of the EPG/SAG for the ESBWR.

18B.2.2 Steam Cooling and Alternate Level Control

In Contingency 1 of BWROG EPG/SAG Rev. 2, steam cooling is required if no injection or alternate injection pumps are running as noted above in Section 18B.2.1. The purpose of steam cooling is to allow the operator more time to get injection systems lined up with pumps running prior to depressurization. Once injection systems are available, depressurization can be accomplished with assurance of injection capability. For the ESBWR, gravity re-flooding is the principal mode of assuring adequate core cooling which can only be accomplished after blow down, so there is no need to delay blow down to get injection systems lined up with pumps running and thus no need for steam cooling.

As discussed in Section 18B.2.1 and directly above, the Alternate Level Control actions of injection system lineup and pump starting and potential need for steam cooling are not appropriate for the ESBWR so these steps have been eliminated. Additionally, the emergency depressurization and containment flooding steps of Contingency 1 of BWROG EPG/SAG Rev. 2 have been incorporated into the ESBWR EPG/SAG various level control sections, so the ESBWR EPG/SAG has no Alternate Level Control or Steam Cooling Contingencies.

18B.2.3 Emergency Depressurization

Emergency depressurization in BWROG EPG/SAG Rev. 2 following initiation of the isolation condenser, directs the operator to open all ADS valves or if any ADS valves can't be opened, to open an equivalent number of SRVs. For the ESBWR direction is given to open all SRVs. The operator is not instructed to take action to open the DPVs until other means of depressurization have been attempted and other level recovery actions were attempted and found to be unsuccessful.

Should this be the case, the next evolution specified in ESBWR EPG/SAG is to re-flood using the GDCS. Complete depressurization is required for re-flooding so opening the DPVs is necessary. Note that the action to open the DPVs is contained in RPV and Containment Flooding rather than in the Emergency Depressurization Contingency for the ESBWR EPG/SAG.

Delayed use of the DPVs is directed because the DPVs discharge RPV effluents directly to the drywell, whereas other paths are preferred from energy and fission product retention considerations.

18B.3 SPECIFIC DIFFERENCES BETWEEN ESBWR AND BWROG EPG/SAG REV. 2

The accompanying table 18B-1 delineates changes made to the EPG/SAG Rev. 2 for adaptation to the ESBWR EPG/SAG. Comparisons are made step-by-step for every change and the basis is provided for the differences. The table is divided into sections corresponding to the major EPG/SAG control sections and contingencies.

Table 18B-1

RPV Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
RC/L	RC/L	No reference to caution #2	ESBWR has no heated reference leg instruments.
RC/L-1	RC/L-1	Isolation condenser is added to list of equipment to be initiated	The isolation condenser provides a significant heat sink and would normally be automatically initiated at Level 2.
RC/L-2	RC/L-2	No reference to Cautions 3, 4 and 5	These cautions are no longer applicable to ESBWR. See Table 18B-9.
RC/L-2	RC/L-2	RCIC, HPCI, HPCS, LPCS and RHR were eliminated from the list of injection systems for possible use.	ESBWR has none of these injection systems.
		FAPCS LPCI mode is included in the list of systems.	ESBWR has low pressure injection capability using FAPCS in the LPCI mode which can be manually initiated.
RC/L-2	RC/L-2	Shorter list of alternate injection systems for possible use.	ESBWR has no RHR or ECCS keep full systems nor any SLC tanks.
RC/L-2	RC/L-2	Preventing automatic RPV depressurization by resetting ADS is eliminated.	ESBWR has no reset or manual inhibit of ADS.
RC/L-2	RC/L-2	Emergency depressurization and continuance in RC/L is specified; EBG/SAG Rev. 2 direction is to exit RC/L and enter the original Contingency #1, Alternate Level Control (which has been removed from the ESBWR EPGs.	The original EPG/SAGs employ a large complement of ECC injection systems which are first employed in the original Contingency #1 before requiring emergency depressurization. ESBWR has the gravity reflood system (GDCS) in place of these systems (refer to additional discussion in Section 18B.1.3).

Table 18B-1

RPV Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
RC/L-3	—	Additional action in ESBWR guideline (similar to RC/L-2 actions at lower pressure).	Since ESBWR RPV depressurization is initiated at Level 1.5 vs TAF in BWROG EPG/SAG Rev. 2, additional flexibility is given to attempt level restoration using step RC/L-2 systems but at lower RPV pressure (refer to additional discussion in Section 18B.2.
RC/L-3	—	Exit to RPV flooding contingency if no way was found to maintain RPV water level above TAF.	This is different from BWROG EPG/SAG Rev. 2 only in that it appears in RC/L rather than in Contingency #1. In both guidelines, all other level recovery possibilities are exhausted before this action is taken.
RC/P	RC/P	First override in BWROG EPG/SAG Rev. 2 calling for prevention of low pressure ECCS injection by a high drywell pressure is omitted for the ESBWR.	ESBWR has no LPCS; FAPCS LPCI mode has no automatic initiation, so no prevention action is necessary.
RC/P	RC/P	Remove reference to Caution #2 in second override.	ESBWR has no heated reference leg instruments.
RC/P-1	RC/P-1	Third override when steam cooling is required is deleted for ESBWR.	ESBWR has no Steam Cooling Contingency. Steam Cooling is an action taken in operating BWRs to delay blow down in situations when ECC pumps are not available.
RC/P-2	RC/P-2	Delete reference to Cautions 3, 4 and 5.	These Cautions are not applicable to ESBWR.

Table 18B-1

RPV Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
		HPCI, RCIC and other steam driven equipment options are deleted from the list of pressure control augmentation systems.	These equipment options are not available in ESBWR.
		RWCU/SDC high pressure shutdown cooling mode is called for rather than “RWCU (re-circulation mode)”	Alternate name and mode for this function in the ESBWR.
		No brackets enclosing “regenerative heat exchangers and”	Bypassing the heat exchangers can be done in ESBWR and is so directed.
RC/P-2	RC/P-2	A second override is added to use alternate shutdown cooling procedures, if required.	ESBWR shutdown cooling, initiated as an augmentation to pressure control in the same step, can operate over the full pressure range. The override calls for alternate shutdown cooling, if required. These changes are made along with deletion of BWROG EPG/SAG Rev. 2 Step RC/P-4 for a consistent set of instructions (refer to discussion below).
—	RC/P-4	This step is not included for the ESBWR.	RWCU/SDC high pressure shutdown cooling mode can operate at high pressure so there is no need to wait for pressure interlocks to clear. Use of alternate methods for further cool down is called for in the RC/P-2 second override (refer to discussion above).
RC/Q	RC/Q	First and second overrides for ESBWR omit the instruction to “terminate boron injection”	Accumulator driven boron injection for the ESBWR cannot be terminated once initiated.

Table 18B-1

RPV Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
RC/Q-1	RC/Q-1	Brackets encompassing the actions are deleted.	Brackets indicate system use or action may not be appropriate. Deleted because this action is the best means of providing diverse and redundant scram signals for ESBWR.
RC/Q-2	RC/Q-2	Brackets encompassing the entire step are deleted.	This step is applicable to ESBWR.
—	RC/Q-3	Deleted	Re-circulation flow runback is not an option in ESBWR since there are no re-circulation pumps.
—	RC/Q-4	Deleted	Re-circulation pump trip is not an option in ESBWR since there are no re-circulation pumps.
—	RC/Q-5	Deleted	This instruction already given in previous step, RC/Q-2.
RC/Q-2	RC/Q-5	Same instructions are given in the override; only step numbers are changed.	Required renumbering because of RC/Q-3, 4 and 5 deletions.
RC/Q-3	RC/Q-6	Same instruction; only a step number change.	Required because of step deletions.
RC/Q-3	RC/Q-6	Shorter list of alternate boron injection methods; alternate name for RWCU.	HPCS, HPCI and RCIC not available in ESBWR; RWCU/SDC is the system designation in ESBWR.
RC/Q-3	RC/Q-6	Deleted override to trip SLC pumps when SLC tank level reaches low level trip.	There are no SLC pumps or SLC tank in ESBWR.
RC/Q-3	RC/Q-6	Identical action; steps renumbered and RWCU designation changed.	Renumbered and renamed for consistency (refer to previous discussions).
RC/Q-4	RC/Q-7	Number change	Consistency with deleted step and renumbering.

Table 18B-1

RPV Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
RC/Q-4.1	RC/Q-7.1	Number change	Numbering consistency
RC/Q-4.2	RC/Q-7.2	Number change.	Numbering consistency
		The text "drain the scram discharge volume" has been deleted for ESBWR in the third method	There is no scram discharge volume feature in the ESBWR CRD.
		Deleted "Increase CRD cooling water differential pressure" and "Vent control rod drive over piston volumes".	These methods are not applicable to the FMCRDs used in the ESBWR.
		Changed "RSCS" to "RCIS"	ESBWR has no RSCS but the RCIS provides a similar function.

Table 18B-2

Primary Containment Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG EPG/SAG REV 2 EPG EPG Step	Difference	Basis For Difference
All steps in the BWROG EPG/SAG REV 2 column below refer to the Mark I and II Containment. Procedures. These more closely approximate the ESBWR design			
Entry Condition	Entry Condition	No containment temperature entry condition in ESBWR EPG.	This entry condition is applicable only to plants with Mark III containments.
Operator Action PC Overrides	Operator Action PC Overrides	Deleted 2nd, 3rd, 4th, and 5th overrides.	These overrides referred specifically to drywell or suppression pool sprays which are not included in the ESBWR design.
Branch before Step SP/T	Branch before Step SP/T	"CN/T" deleted from concurrent execution	CN/T is for Mark III containments only.
SP/T-1	SP/T-1	Brackets removed; reference to "RHR" pumps replaced by "FAPCS".	ESBWR has no RHR system but FAPCS can be used for suppression pool cooling if not needed to assure adequate core cooling.
SP/T-3	SP/T-3	Added "Before the Heat Capacity Temperature Limit is reached; operate all available Isolation Condensers to depressurize the RPV"	ICs can be used to avoid the HCTL and prevent the need for emergency depressurization. It was decided to highlight this as a separate step, even though the ICs are called out in RPV Control.
SP/T-4	SP/T-3	Separated the emergency depressurization requirement into a separate step	This is an editorial change to avoid having two actions in a single step
DW/T-2	DW/T-2	Deleted "DRYWELL SPRAY IS REQUIRED"	ESBWR does not use drywell spray in the EPGs
DW/T-2	DW/T-2	Additional direction is given to enter RPV control and execute	Entry to RPV control is given earlier because earlier entry to

Table 18B-2

Primary Containment Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG EPG/SAG REV 2 EPG EPG Step	Difference	Basis For Difference
		it concurrently.	RPV control would allow operator a chance to depressurize through the main condenser or ICs before emergency depressurization as required at Step DW/T-3
DW/T-3	DW/T-3	Direction to enter RPV control deleted.	Same direction given in earlier step (see Step DW/T-2 for discussion of the basis).
—	CN/T	This section eliminated from ESBWR EPG.	Applies only to BWRs with Mark III containments.
PC/P	PC/P	Added Containment Inerting System (CIS) and RBHVAC system.	The system called for in the ESBWR EPG to control containment pressure is the Containment Inerting System (CIS). It operates in conjunction with the RBHVAC system.
		Deleted "When primary containment pressure cannot be maintained below [13.8 KPA (2.0 psig) (high drywell pressure scram set point)]:"	No additional actions are available that are not already being done. There are no suppression pool sprays or use of drywell sprays in this mode for the ESBWR.
—	PC/P-1	Deleted Suppression Pool Spray Requirement	ESBWR does not have Suppression Pool Sprays.
—	PC/P-2	Deleted Drywell Spray Requirement	ESBWR does not use Drywell Sprays in the EPGs.
PC/P-1	PC/P-3	PC/P-3 was renumbered to PC/P-1	PC/P-1 and 2 were deleted
PC/P-2	PC/P-4	Step renumbered	Deletion of EPG/SAG steps PC/P-1 and 2 required renumbering
PC/P-3	PC/P-5	Step renumbered	Deletion of EPG/SAG steps PC/P-1 and 2 required

Table 18B-2

Primary Containment Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG EPG/SAG REV 2 EPG EPG Step	Difference	Basis For Difference
			renumbering
SP/L-1	SP/L-1	In ESBWR direction is given to use FAPCS.	No specific systems are specified in EPG/SAG because of plant variability. In ESBWR FAPCS is appropriate.
		References to SPMS have been eliminated.	ESBWR has no SPMS.
SP/L-2	SP/L-2	Eliminated override to execute following two sub steps concurrently.	Only one sub step follows in the ESBWR EPG.
SP/L-2	SP/L-2.1	Sub step number eliminated.	Sub step is not needed for ESBWR (See SP/L-2.2 discussion)
—	SP/L-2.2	Sub step eliminated.	No concern regarding HPCI exhaust since ESBWR has any HPCI.
SP/L-3	SP/L-3	Reference to SPMS deleted.	No SPMS in ESBWR.
SP/L-3.2	SP/L-3.2	"feet" changed to "meters" in two places.	Units change.
		Removed requirement to terminate drywell sprays.	ESBWR does not use Drywell Sprays in the EPGs.
PC/G-1	PC/G-1	Removed sub steps associated with hydrogen re-combiner operation	ESBWR does not utilize hydrogen re-combiners for hydrogen concentration control
PC/G-2	PC/G-2	Removed sub steps associated with hydrogen re-combiner operation	ESBWR does not utilize hydrogen re-combiners for hydrogen concentration control
PC/G-3	PC/G-3	Deleted reference to hydrogen mixing system and hydrogen re-combiners	ESBWR does not have a hydrogen mixing system or hydrogen re-combiners
-------	PC/G-3.3	Deleted this step which initiates drywell sprays	ESBWR does not use Drywell Sprays in the EPGs.
PC/G-4	PC/G-4	Removed sub steps associated	ESBWR does not utilize

Table 18B-2

Primary Containment Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG EPG/SAG REV 2 EPG EPG Step	Difference	Basis For Difference
		with hydrogen re-combiner operation	hydrogen re-combiners for hydrogen concentration control
PC/G-5	PC/G-5	Removed sub steps associated with hydrogen recombiner operation	ESBWR does not utilize hydrogen re-combiners for hydrogen concentration control
PC/G-6	PC/G-6	Removed words “secure all re-combiners taking suction on the suppression chamber “	ESBWR does not utilize hydrogen re-combiners for hydrogen concentration control
-------	PC/G-6-1	Deleted this step which requires suppression pool sprays	ESBWR does not have suppression pool sprays
PC/G-6-1	PC/G-6-2	Renumbering required	Deletion of earlier step
PC/G-6-2	PC/G-6-3	Renumbering required	Deletion of earlier step

Table 18B-3

Reactor Building Control Emergency Procedure Guidelines

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
Title	Title	“Secondary Containment” changed to “Reactor Building”	The building which surrounds the primary containment in the ESBWR is referred to as the Reactor Building
RB-1 2nd Override	RB-1 2nd Overrides	“Confirm initiation of or manually initiate SBGT” eliminated.	ESBWR has no Standby Gas Treatment System. .
Tables 18A-2 thru 18A-4	Table SC-1	This table was generalized for the ESBWR EPG/SAGs	Detailed information will be provided under the plant specific implementation of the EPGs

Table 18B-4

Radioactivity Release Control Emergency Procedure Guideline

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
Override	Override	Deleted "[or isolated due to high radiation]" and "defeating isolation interlocks if necessary".	HVAC is not isolated on high radiation in buildings other than the reactor building.
RR-2&3	RR-2	Separated BWROG step actions into two steps.	Allow attempt at normal depressurization thereby reducing containment heat load prior to emergency depressurization.

Table 18B-5

Contingency 1 - Emergency Depressurization

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
Note: Contingency 1 in EPG/SAG Rev 2 is "Alternate Level Control". This contingency involves the control of RPV level below the top of the active fuel. It is not applicable to the ESBWR and has been deleted.			
C1-1	C2-1	Reference to RCIC has been deleted and "purge flow" has been added to CRD.	ESBWR has no RCIC. ESBWR CRD can be operated in the injection mode, but operating in the purge mode characterizes this entry condition for an ATWS situation.
C1-1	C2-1	Removed original Caution 2	Original Caution 2 refereed to heated reference leg instruments which are not used in the ESBWR
—	C2-1.1	Deleted.	ESBWR has no automatic ECCS initiation on a high drywell pressure signal.
C1-1.1	C2-1.2	Step number change. Changed to "initiate all ICs".	Renumbered because of deleted step. Effects subsequent steps also. ESBWR has 4 ICs.
C1-1.2	C2-1.3	Step number change. Direction to open all SRVs vs. equivalent number dedicated to ADS. Minimum number of SRVs changed from 4 to 8	All SRVs are used to maximize depressurization. Minimum number of SRVs for ESBWR has yet to be determined. Current number is assumed to be 8.
		Cautions 3, 4 & 5 were deleted	Cautions 3, 4, 5 referred to systems not available in ESBWR (HPCI, RCIC, HPCS, LPCS, RHR).
		Shortened list of options for depressurizing.	No HPCI, RCIC or RHR systems in ESBWR.
C1-2	C2-2	Replaced Shutdown Cooling with RWCU/SDCS. Eliminated reference to pressure	ESBWR uses RWCU/SDCS to achieve cold shutdown. System operates at full reactor pressure.

Table 18B-5

Contingency 1 - Emergency Depressurization

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
		interlock	
		Removed reference to RHR	ESBWR has no RHR system.
C1-2	C2-2	Original Cautions 3, 4 & 5 were deleted	Cautions 3, 4, 5 referred to systems not available in ESBWR (HPCI, RCIC, HPCS, LPCS, RHR).

Table 18B-6

Contingency 2 - RPV Flooding

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
This was Contingency #4 in EPG/SAG Rev 2. Contingency #3 in EPG/SAG Rev 2 was "Steam Cooling" which is not applicable to the ESBWR and was eliminated. Thus, "RPV Flooding" became Contingency #2.			
C2-1	C4-1	Renumbering all steps from C4 to C2	Deletion of earlier contingencies necessitated renumbering of subsequent contingencies
C2-1.1	C4-1.1	"purge mode" has been added to CRD to specify the operation mode. RCIC was removed.	CRD flow injection through the feedwater line is not desirable when it is not certain that the reactor is or will remain shutdown. ESBWR has no RCIC system.
C2-1.2	C4-1.2	Removed discussion regarding ADS valves.	ESBWR direction is to use all SRVs for emergency manual depressurization rather than the subset of these identified as ADS valves.
C2-1.3	C4-1.3	Shortened list of injection systems.	No RCIC, HPCI, LPCI or RHR systems in ESBWR.
		Shortened list of alternate injection systems.	No HPCS, HPCS, LPCI, LPCS, or ECCS keep-full system in ESBWR.
		Shortened list of valves to close.	Fewer systems for ESBWR
		List of alternate depressurization systems shortened	ESBWR has no HPCI, RCIC or RHR systems
		Original Caution #s 3,4,5,6 were replaced with new Caution #3	Original Cautions 3, 4, 5 are not applicable to ESBWR. Original Caution #6 was renumbered to Caution #3.
C2-2.1	C4-2.1	Removed discussion regarding ADS valves.	ESBWR direction is to use all SRVs for emergency manual depressurization rather than the subset of these identified as

Table 18B-6

Contingency 2 - RPV Flooding

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
			ADS valves.
C2-2.2	C4-2.2	Removed HPCS and "motor driven". Added CRD high pressure makeup	ESBWR has no HPCS. All FW pumps are motor driven. CRD provides high pressure makeup in ESBWR.
		Shortened list of systems to isolate	No HPCI, RCIC or RHR in ESBWR
C2-2.3	C4-2.3	Shortened list of systems to use for flooding. Removed "motor drive" from feed pump. Added "high pressure makeup mode" to CRD system.	No HPCS, LPCS, LPCI, RHR or ECCS Keep-Full systems in ESBWR. All feed pumps in ESBWR are motor driven.
		Original Cautions 3, 4 & 5 were deleted	Original Cautions 3, 4, 5 referred to systems not available in ESBWR (HPCI, RCIC, HPCS, LPCS, RHR).
		Removed "motor drive" from feed pump. Removed HPCS and added "CRD high pressure makeup mode". Changed list of systems to use for depressurization if insufficient number of SRVs can be opened.	No HPCS, HPCI, RHR or RCIC systems in ESBWR. All feed pumps in ESBWR are motor driven. ESBWR has RWCU/SDC system and uses CRD in high pressure makeup mode.
C2-3	C4-3	Removed RCIC, HPCI, and RHR systems.	ESBWR has no RCIC, HPCI and RHR systems

Table 18B-7

Contingency 3 - Level/Power Control

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
The deletion of Contingencies 1 & 3 from the BWROG EPGSAG results in the renumbering of this Contingency for #5 to #3.			
-------	C5-1	The text “prevent automatic initiation of ADS” has been eliminated.	Operator prevention of ADS is not possible in ESBWR. ATWS logic automatically inhibits ADS without need for operator action.
C3-1	C5-2	Renumbering of step	Deletion of original step C5-1. This also applies to all subsequent steps.
C3-2	C5-3	Eliminated RCIC and added CRD “purge flow”	ESBWR has no RCIC. CRD has two modes, purge mode and high pressure makeup mode. Purge mode is applicable here.
C3-3	C5-4	Eliminated RCIC and added CRD “purge flow”	ESBWR has no RCIC. CRD has two modes, purge mode and high pressure makeup mode. Purge mode is applicable here.
C3-4	C5-5	Changed low level control from Minimum Steam Cooling RPV Water Level to Top of Active Fuel. Original Caution #s 3,4,5,6 were replaced with new Caution #3	Minimum Steam Cooling RPV Water Level control is not being utilized in the ESBWR EPGs. Original Cautions 3, 4, 5 are not applicable to ESBWR. Original Caution #6 was renumbered to Caution #3.
C3-4	C5-5	Shortened list of injection systems. Added, “either purge flow or high pressure makeup modes” to CRD.	RCIC LPCI, HPCI and RHR are not systems in the ESBWR; water can be injected using the FAPCS LPCI mode.
C3-4.1	C5-5.1	“purge mode” added to CRD and “RCIC” deleted.	Same basis as given in C3-2; ESBWR has no RCIC.

Table 18B-7

Contingency 3 - Level/Power Control

ESBWR EPG/SAG EPG Step	BWROG Rev. 2 EPG/SAG EPG Step	Difference	Basis For Difference
C3-4.2	C5-5.2	Shortened list of injection systems. Added, “either purge flow or high pressure makeup modes” to CRD.	RCIC LPCI, HPCI and RHR are not systems in the ESBWR; water can be injected using the FAPCS LPCI mode.
		Changed low-level control from Minimum Steam Cooling RPV Water Level to Top of Active Fuel.	Minimum Steam Cooling RPV Water Level control is not being utilized in the ESBWR EPGs.
		Original Caution #s 3,4,5,6 were replaced with new Caution #3	Original Cautions 3, 4, 5 are not applicable to ESBWR. Original Caution #6 was renumbered to Caution #3.
Override following C3-4.2	Override following C5-5.2	Changed level for entry to Primary Containment Flooding from “Minimum Steam Cooling RPV Water Level” to “Top of Active Fuel.”	Minimum Steam Cooling RPV Water Level control is not being utilized in the ESBWR EPGs. Top of active fuel is the minimum level for action.
C3-4.3	C5-5.3	Changed low level control from Minimum Steam Cooling RPV Water Level to Top of Active Fuel.	Minimum Steam Cooling RPV Water Level control is not being utilized in the ESBWR EPGs.
—	C5-6	Step C5-6 and its preceding override are eliminated.	This step and override apply to plants with SLC injection location below the core in the lower plenum. The purpose was to raise level and promote mixing into the core. ESBWR SLC injects directly into the core so this action is not required.

Table 18B-8

RPV and Primary Containment Flooding Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
Purpose	Purpose	First purpose changed to "Submerge the core (and in-vessel core debris)" Added an additional purpose "Submerge core debris on the lower drywell floor	More descriptive of purpose of this SAG
RC/F	RC/F	Removed original Caution #2 and added new Caution 3.	Original Caution #2 is not applicable to ESBWR design New Caution 3 is applicable to this step.
		Modified drywell spray override to include a shutoff of sprays unless core debris is on drywell floor.	This is directly related to new Caution 3 to avoid possible steam explosion
		Deleted suppression pool sprays from override	ESBWR has no suppression pool sprays
RC/F-1	RC/F-1	Added override to depressurize with DPVs, initiate passive cooling system and drywell deluge.	ESBWR systems used for post accident RPV and containment flooding
------	RC/F-1.1	Removed override regarding suppression pool sprays	ESBWR has no suppression pool sprays
-------	RC/F-1.1	Step deleted	ESBWR passive systems require RPV venting.
RC/F-1.1	RC/F-1.2	Renumbering of steps	Required because RC/F-1.1 was deleted

Table 18B-8

RPV and Primary Containment Flooding Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
RC/F-1.1	RC/F-1.2	Deleted "Initiate SPMS" from ESBWR EPG.	ESBWR has no SPMS.
		Modified list of systems.	ESBWR has no HPCS, LPCS, RCIC, RHR or ECCS keep-full systems. LPCI is possible using FAPCS in the LPCI mode. Other containment fill systems in the ESBWR are the Makeup Water System and the Condensate Storage and Transfer System.
		Removed original Cautions 3, 4, 5	Original Cautions 3, 4, 5 involved non – ESBWR systems
		Removed override regarding suppression pool sprays	ESBWR has no suppression pool sprays
RC/F-1.2	RC/F-1.3	Renumbering of steps	Required because RC/F-1.1 was deleted
RC/F-1.2	RC/F-1.3	Removed requirement to operate HPCS and LPCS	ESBWR has no HPCS or LPCS.
		Modified list of systems.	ESBWR RHR or ECCS keep-full systems. LPCI is possible using FAPCS in the LPCI mode. Other containment fill systems in the ESBWR are the Makeup Water System and the Condensate Storage and Transfer System.
RC/F-2	RC/F-2	Changed title	The new title reflects the condition in which water level can be restored up to and beyond TAF. This is associated with the simplification of RC/F for the ESBWR.

Table 18B-8

RPV and Primary Containment Flooding Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
		Added override to open DPVs, initiate GDCS and open SP equalization line	ESBWR passive system for RPV depressurization / venting to drywell and core cooling by gravity drain to RPV
		Added override to maintain level as high as possible if it cannot be restored and maintained above TAF and to maximize injection from external sources	All available systems are already being used to restore level and attempt to maintain it above TAF. If TAF cannot be achieved, there are no additional actions that can be taken.
		Removed override for suppression pool sprays.	ESBWR has no suppression pool sprays
		Removed override for drywell sprays	Drywell sprays will only be used if lower drywell floor thermocouples indicate core debris on the drywell floor.
		System list to restore and maintain level has been modified	ESBWR has no RCIC and RHR. Added CRD high pressure makeup mode and FAPCS LPCI mode.
		Backup system list to restore and maintain RPV level has been modified	ESBWR has no HPCS, LPCS, RHR or ECCS Keep-Full system
		Removed original Cautions 3, 4, 5	Original Cautions 3,4,5 involved non – ESBWR systems
		Shortened list of alternate methods to use to vent the RPV	ESBWR does not have flood vent valves, HPCI, RCIC or RHR. The primary means of RPV venting in the ESBWR are the DPVs.
		RPV venting with alternate systems is only required if no DPVs can be opened	One or more open DPVs is the most desirable manner of venting the RPV in this step
--------	RC/F-3	This step was deleted	Steps RC/F-3 thru RC/F-6 are not

Table 18B-8

RPV and Primary Containment Flooding Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
--------	RC/F-4	This step was deleted	directly applicable to the ESBWR because if the DPVs and GDCS function as designed then a core melt is not credible. If the GDCS fails then the ESBWR severe accident assumption that core melt is not arrested in vessel applies. Actions in earlier steps will provide in vessel core debris cooling.
--------	RC/F-5	This step was deleted	
--------	RC/F-6	This step was deleted	
RC/P-1	RC/P-1	Changed to initiate all available Isolation Condensers	ESBWR design has 4 Isolation Condensers
RC/P-2	RC/P-2	Removed original Caution 2	The original Caution 2 is not applicable to ESBWR because it does not utilize heated reference legs.
		Changed open all “ADS valves” to open all “ADS SRVs and DPVs”.	ESBWR depressurization system consists of 10 ADS SRVs and 8 DPVs.
		Modified list of systems for further depressurization	ESBWR does not have HPCI, RCIC & RHR. ESBWR has RWCU/SDC system.
RC/P-3	RC/P-3	Modified list of systems for further cool down	ESBWR does not have HPCI, RCIC & RHR. ESBWR has RWCU/SDC system.
		Removed original Cautions 3, 4, 5	These original Cautions are not applicable to ESBWR
RC/Q-3	RC/Q-3	Removed “until SLC tank water level drops to [0% (low SLC tank water level trip)]”.	ESBWR has no SLC tank or pump, only accumulators. Operator has no control of SLC level in the accumulator and there is no low SLC tank water level trip.

Table 18B-8

RPV and Primary Containment Flooding Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
RC/Q-4.2	RC/Q-4.2	Deleted “Increase CRD cooling water differential pressure” and “Vent control rod drive over piston volumes”.	These methods are not applicable to the FMCRDs used in the ESBWR.

Table 18B-9

Cautions

ESBWR EPG/SAG Caution	BWROG Rev. 2 EPG/SAG Caution	Difference	Basis For Difference
1	1	No Difference	.
-	2	Deleted for ESBWR	ESBWR has no heated reference leg instruments.
-	3	Deleted for ESBWR	ESBWR does not have these systems, so their NPSH and vortex limits are not applicable.
-	4	Deleted for ESBWR	ESBWR does not have RCIC system.
-	5	Deleted for ESBWR	ESBWR does not have HPCI or RCIC turbines.
2	6	No Difference	
-	7	Deleted for ESBWR	No NPSH issues for ESBWR because the FAPCS design requires operability without pump cavitations with conditions down to saturation at the suppression pool source.
3	-	Added a caution to identify that if core debris interacts with a water level in the lower plenum that is greater than 0.7 m, a steam explosion may occur.	This phenomenon is documented in the ESBWR severe accident analyses and is added here to prevent actions that will lead to high water level in the lower drywell prior to core melt ejection.

Table 18B-10

Containment and Radioactivity Release Control Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
OPERATOR ACTIONS	OPERATOR ACTIONS	Deleted “Confirm initiation of or manually initiate SBGT” in first override	ESBWR does not have a SBGT system
		Removed CN/T from steps to execute concurrently	CN/T refers to Mark III containments
SP/T	SP/T	Removed original Caution 3	Original Caution 3 does not apply to ESBWR
DW/T	DW/T	Removed “DRYWELL SPRAY IS REQUIRED” and added “Drywell spray can be utilized if drywell water level is less than 0.7m or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than 0.7m and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized.”	There is no requirement to initiate drywell sprays in the ESBWR. They may be used under the conditions cited.
		Added new Caution 3	New Caution 3 refers to use of drywell sprays in a manner to preclude steam explosions
---------	CN/T	Removed this step	CN/T applies only to Mark III containments
PC/P	PC/P	Removed requirement to initiate suppression pool sprays	ESBWR does not have suppression pool sprays

Table 18B-10

Containment and Radioactivity Release Control Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
PC/P	PC/P	Removed "DRYWELL SPRAY IS REQUIRED" and added "Drywell spray can be utilized if drywell water level is less than 0.7 m or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than 0.7 m and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized."	There is no requirement to initiate drywell sprays in the ESBWR. They may be used under the conditions cited.
		Added new Caution 3	New Caution 3 refers to use of drywell sprays in a manner to preclude steam explosions
		Changed "Primary Containment Pressure Limit C " to "Primary Containment Pressure Limit"	Primary Containment Pressure Limits A, B, C do not apply to ESBWR because there is one limit that encompasses the limits associated with A, B & C. The Primary Containment Pressure Limit is the ultimate pressure capability for the ESBWR.
PC/R	PC/R	Removed requirement to initiate suppression pool sprays	ESBWR does not have suppression pool sprays

Table 18B-10

Containment and Radioactivity Release Control Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
		Removed "DRYWELL SPRAY IS REQUIRED" and added "Drywell spray can be utilized if drywell water level is less than 0.7 m or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor. If drywell water level is greater than 0.7 m and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized."	There is no requirement to initiate drywell sprays in the ESBWR. They may be used under the conditions cited.
		Added new Caution 3	New Caution 3 refers to use of drywell sprays in a manner to preclude steam explosions
PC/G associated with Mark I and II containments was used since it was more applicable. PC/G associated with Mark III containments was deleted.			
PC/G-1	PC/G-1	Removed requirement for hydrogen re-combiner operation	ESBWR does not have a hydrogen re-combiner
PC/G-2	PC/G-2	Removed requirement for hydrogen re-combiner operation	ESBWR does not have a hydrogen re-combiner
PC/G-3	PC/G-3	Removed requirement for securing hydrogen re-combiner and hydrogen mixing system	ESBWR does not have a hydrogen re-combiner or mixing system

Table 18B-10

Containment and Radioactivity Release Control Severe Accident Guideline

ESBWR EPG/SAG SAG Step	BWROG Rev. 2 EPG/SAG SAG Step	Difference	Basis For Difference
PC/G-3-3	PC/G-3-3	Removed “DRYWELL SPRAY IS REQUIRED” and added “If drywell water level is less than 0.7 m or lower drywell floor thermocouples indicate the presence of core debris on the drywell floor, drywell sprays can be utilized. If drywell water level is greater than 0.7 m and lower drywell floor thermocouples do not indicate the presence of core debris on the drywell floor, drywell sprays should not be utilized.”	There is no requirement to initiate drywell sprays in the ESBWR. They may be used under the conditions cited.
PC/G-4	PC/G-4	Removed requirement for hydrogen re-combiner operation	ESBWR does not have a hydrogen re-combiner
PC/G-5	PC/G-5	Removed requirement for hydrogen re-combiner operation	ESBWR does not have a hydrogen re-combiner
PC/G-6	PC/G-6	Removed requirement for securing hydrogen re-combiner and hydrogen mixing system	ESBWR does not have a hydrogen re-combiner or mixing system
--------	PC/G-6-1	Deleted step requiring suppression pool sprays	ESBWR does not have suppression pool sprays
PC/G-6-1	PC/G-6-2	Renumbering required	Deletion of step required renumbering of subsequent steps
PC/G-6-2	PC/G-6-3	Renumbering required	Deletion of step required renumbering of subsequent steps
Tables 18A-2 thru 18A-4	Table SC-1	This table was generalized for the ESBWR	More detailed information will be provided when plant specific ESBWR EOPs are developed.

18C. ESBWR EPG/SAG INPUT DATA

18C.1 INTRODUCTION

The Emergency Procedure Guidelines (EPGs) provided in Appendix 18A refer to various limits for emergency plant operation. These operation limits are based upon plant specific design parameters. This appendix identifies the plant parameters that are used for calculation of operation limits. The input parameters provided are in accordance with those in Appendix C of the BWROG EPG/SAG Revision 2.

The parameter input values used for calculation of operation limits are given in Section 18C.2. The COL applicant is required to provide the input parameters based upon specific installation details and calculate the plant specific operation limits (Subsection 18.8.2). In addition, the EPG/SAGs in Appendix 18A shall incorporate these calculations.

18C.2 INPUT PARAMETERS

Tables 18C-1 through 18C-9 list all plant parameters that are used for calculation of operation limits. The parameter definitions are in accordance with Appendix C of the BWROG EPG/SAG, Revision 2. Generic parameter values are provided where appropriate. When detailed plant design is completed and specific plant installation details are known, all parameter values can be determined.

18C.3 CALCULATION RESULTS

Figures 18C-1 through 18C-9 contain typical limit curves and are provided for illustrative purposes only. Calculation of these limit curves would be performed at the COL stage in accordance with the methods given in Appendix C of the BWROG EPG/SAG, Revision 2 using the input data discussed above (subsection 18.8.2).

Table 18C-1
BWROG EPG/SAG Rev. 2 Appendix C: ECCS Suction Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
ECCS Suction Data		
SuctionID1	Suction identification	FAPCS
Dsuct1	Diameter of suction inlet (in.)	
Hsuct1	Elevation of center of suction inlet (m (ft))	
WsuctMax1	Flow (maximum) through suction (gpm)	
ECC Tabular Data		
Wfapcs-lpci Table	Flow rate (gpm) pf the FAPC LPCI as a function of RPV pressure (psig). Run out to shutoff.	

Table 18C-2

BWROG EPG/SAG Rev. 2 Appendix C: Primary Containment Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
Primary Containment Data		
PCDesign	BWR primary containment design (Mark I, II, or III)	ESBWR
Hhorvent	Elevation of top of horizontal vents (ft)	
HscTap	Elevation of suppression chamber pressure instrument tap (ft))	
HspRef	Elevation of suppression pool water level instrument zero (ft)	
HvbInt	Elevation of bottom of internal Mark II containment wetwell-to-drywell vacuum breakers (ft)	
HventPC	Elevation of containment vent capable of removing all decay heat and located above TAF (ft)	
PdwMaxop	Pressure (maximum normal operating), drywell (psig)	
PdwMinop	Pressure (minimum normal operating), drywell (psig)	
PdwScram	Pressure setpoint for high drywell pressure scram (psig)	
PpcVent	Pressure (maximum) in airspace at which containment vent located above TAF can be opened and closed (psig)	
PscMaxop	Pressure (maximum normal operating), suppression chamber (psig)	
PscMinop	Pressure (minimum normal operating), suppression chamber (psig)	
PspDes	Load (design), suppression pool boundary (psi)	
PspSRV	Load (maximum) on suppression pool boundary resulting from SRV actuation (psi)	
Tcst	Temperature (maximum normal operating) of condensate storage tank water (°F)	
TdwMaxop	Temperature (maximum normal operating), drywell (°F)	
TdwMinop	Temperature (minimum normal operating), drywell (°F)	
TscMax	Temperature capability (maximum) of the suppression	

Table 18C-2

BWROG EPG/SAG Rev. 2 Appendix C: Primary Containment Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
Primary Containment Data		
	chamber and equipment within the suppression chamber which may be required to operate when the RPV is pressurized (ºF)	
TscMaxop	Temperature (maximum normal operating), suppression chamber airspace (°F)	
TscMinop	Temperature (minimum normal operating), suppression chamber airspace (°F)	
TspFlood	Temperature (minimum) of suppression pool when containment flooded (°F)	
TspMinop	Temperature (minimum normal operating), suppression pool (°F)	
TspScram	Temperature of suppression pool at which reactor scram is required (°F)	
Vdw	Volume (free) of drywell and vent system (ft3)	
VscLCO	Volume (free) of suppression chamber above minimum suppression pool water level LCO (ft3)	
WLspMaxLCO	Water level LCO (maximum) of suppression pool (ft)	
WLspMinLCO	Water level LCO (minimum) of suppression pool (ft)	
Parameters for components which are limiting at high containment pressures		
CompID1	Identification	DRYWELL HEAD
Elevation1	Elevation (m (ft))	
Location1	Location (DW or WW)	DW
Material1	Material type	
Strength1	Strength type (yield or tensile)	
Pcalc1	Pressure capability (maximum) (psig)	
Tcalc1	Temperature used to determine Pcalc (°F)	
CompID2	Identification	WETWELL BOTTOM

Table 18C-2

BWROG EPG/SAG Rev. 2 Appendix C: Primary Containment Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
Primary Containment Data		
Elevation2	Elevation (m (ft))	
Location2	Location (DW or WW)	WW
Material2	Material type	
Strength2	Strength type (yield or tensile)	
Pcalc2	Pressure capability (maximum) (psig)	
Tcalc2	Temperature used to determine Pcalc (°F)	
Primary Containment Tabular Data		
 Nnzl(n) Wnzl(n)	Table: Nnzl : Wnnzl -- The minimum number of spray nozzles of size “n” through which a single Drywell Spray division can deliver flow to the drywell. (n = 1, 2, 3, etc) The minimum flow that provides a full cone spray pattern for Nnzl (n) (gpm)	
  WLsp Vsp Vsc-air	Table WLsp: Vsp: Vsc-air -- Volume of water and airspace in the suppression chamber as a function of suppression pool water level bottom to top of suppression chamber: Suppression pool water level (ft) Volume (ft3) of water in suppression pool for a given height WLsp Volume (ft3) of airspace in suppression chamber for a given water height WLsp	

Table 18C-3

BWROG EPG/SAG Rev. 2 Appendix C: Fuel Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
Fuel Data		
c-clad	Specific heat of clad and channels (BTU/lbm-°F)	
c-fuel	Specific heat of fuel (BTU/lbm-°F)	
Mclad	Mass of clad and channels (lbm)	
Mfuel	Mass of fuel (lbm)	
Nbuns	Number of fuel bundles	1132
Qrx-rated	Power (rated) (MWt)	4500
Lfuel	Length of active fuel (in)	120
	Fuel Type	GE-14
Fafl-15	Minimum active fuel length fraction, which must be covered to maintain PCT<1500°F with injection (%)	
Fafl-18	Minimum active fuel length fraction which must be covered to maintain PCT<1800°F without injection (%)	
FQdh-10	Decay heat fraction 10 minutes after shutdown	0.0221
Fuel Data Tables		
Wg-1500 (Wg-1500Table)	Fuel-n: Wg-1500-n -- Minimum bundle steam flow required to maintain PCT<1500°F at peak LHGR (lbm/hr) Fuel-n Wg-1500-n (lbm/hr) 1 -------- 2 -------- 3 --------	

Table 18C-3
BWROG EPG/SAG Rev. 2 Appendix C: Fuel Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
Qdh (Table)	Time (min): Qdh (%) -- Decay heat after reactor shutdown: Time (min.) Qdh-% Time Qdh 2 3.038 120 1.057 4 2.658 150 0.9901 6 2.458 200 0.9661 8 2.320 300 0.8673 10 2.212 450 0.7863 15 2.009 600 0.7255 20 1.862 900 0.6509 25 1.747 1200 0.6002 30 1.653 1500 0.5622 35 1.574 1800 0.5331 40 1.507 45 1.450 50 1.400 60 1.318 90 1.155	

Table 18C-4

BWROG EPG/SAG Rev. 2 Appendix C: RPV Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
RPV Data		
dPvent-rpv	Differential pneumatic pressure (minimum) required to operate RPV vent valve(s) (psid)	
Hvent-rpv	Elevation of lowest RPV vent valve pneumatic solenoid (ft)	
Mf-rpv-cld	Mass of water in shutdown cooling, and RWCU loops and in RPV with water level at high level trip setpoint and water temperature at 68°F (lbm)	
Mf-rpv-hot	Mass of water in RWCU loops and in RPV with water level at high level trip setpoint and water temperature at saturation temperature for minimum pressure at which an SRV is set to lift (lbm)	
Mg-rpv-hot	Mass of saturated steam in RPV and main steam lines inboard of outboard MSIVs with water level at high level trip setpoint and pressure at minimum at which an SRV is set to lift (lbm)	
Mrpv	Mass of RPV, internals and main steam lines inboard of outboard MSIVs (lbm)	
Psup-rpv	Pressure (minimum normal operating), pneumatic supply system for RPV vent valve(s) (psig)	
WLrpv-baf	Water level at bottom of active fuel (in.)	

Table 18C-5

BWROG EPG/SAG Rev. 2 Appendix C: RPV Level Instrument Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
RPV Water Level Instrument Data		
Provide the following information for each RPV water level instrument range		
Fhtc	Heat transfer coefficient (zero unless instrument has heated reference leg) (dimensionless)	
Hrange-lo	Elevation of instrument range low end (ft)	
Href-dw	Elevation of reference leg drywell penetration (ft)	
Href-surf	Elevation of condensing chamber water surface (ft)	
Hvar-dw	Elevation of variable leg drywell penetration (ft)	
Hvar-tap	Elevation of variable leg RPV tap (ft)	
Prpv-cal	Pressure in RPV at calibration (psig)	
Tdw-cal	Temperature in drywell at calibration (°F)	
Trb-cal	Temperature in reactor building or containment at calibration (°F)	
WLrpv-hi	Water level at instrument range high end (in.)	
WLrpv-lo	Water level at instrument range low end (in.)	

Table 18C-6

BWROG EPG/SAG Rev. 2 Appendix C: SLC System Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
	SLC System Data	
FB10-slc	Atomic abundance of B10 isotope (i.e., boron enrichment) as a fraction of all boron in the SLC Accumulator (0.1978 if naturally occurring boron is used)	
Tslc	Temperature (maximum normal operating) of water in SLC Accumulator (°F)	
Wslc	SLC injection flowrate for boron injection under 10CFR50.62, the ATWS rule (gpm)	
XB-cld-nat	Cold shutdown boron concentration requirement for naturally occurring boron (ppm)	
XB-hot-nat	Hot shutdown boron concentration requirement for naturally occurring boron (ppm)	
XB-slc	Concentration (minimum normal operating) of boron in SLC Accumulator (ppm)	

Table 18C-7

BWROG EPG/SAG Rev. 2 Appendix C: SRV System Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
dPsrv	Differential pneumatic pressure (minimum) required to open SRVs (psid)	
Hsrv	Elevation of lowest SRV pneumatic solenoid (ft)	
Nsrv-ads	Number of SRVs dedicated to ADS	10
Pq-code	Stress (code allowable) for SRV quencher (kpsi)	
Pq-des	Stress (design basis) for SRV quencher (kpsi)	
Pqs-code	Stress (code allowable) for SRV quencher support (kpsi)	
Pqs-des	Stress (design basis) for SRV quencher support (kpsi)	
Prpv-tp	Pressure in RPV used for SRV tail pipe design calculations (psig)	
Psrv-lift	Pressure (minimum) in RPV at which an SRV is set to lift (psig)	
Psrv-name	Pressure for SRV per nameplate (psig)	
Psup-srv	Pressure (minimum normal operating), pneumatic supply system for SRVs (psig)	
Ptp-code	Stress (code allowable) for SRV tail pipe (kpsi)	
Ptp-des	Stress (design basis) for SRV tail pipe (kpsi)	
Ptps-code	Stress (code allowable) for SRV tail pipe support (kpsi)	
Ptps-des	Stress (design basis) for SRV tail pipe support (kpsi)	
Type of SRV (Name, Model#)	Dresser Crosby Target Rock Dikkers Sebim	
WLsp-srv	Water level of suppression pool used to determine maximum suppression pool boundary load resulting from SRV actuation (ft)	
WLsp-tp	Water level of suppression pool used for SRV tail pipe	

Table 18C-7

BWROG EPG/SAG Rev. 2 Appendix C: SRV System Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
	design calculations (ft)	
Wsrv-name	Flowrate through SRV per nameplate (lbm/hr)	
WLL-n (Tables)	WLsp: WLL-n -- Water leg length (*WLL-n*) in tail pipe (for discharge line "n") as a function of suppression pool water level (*WLsp*) referenced to 0 for *WLsp* at level used for SRV tail pipe design calculations. One table for each SRV discharge line (SRVDL 1…n)).	

Table 18C-8

BWROG EPG/SAG Rev. 2 Appendix C: Generic Input Data

PARAMETER	PARAMETER DEFINITION	VALUE
c-steel	Specific heat of steel (BTU/lbm-°F)	0.128
dPtp-dPrpv	SRV tail pipe system load variation with RPV pressure (%/psig)	0.10
dPtp-dWLL	SRV tail pipe system load variation with SRVDL water leg length (%/ft)	5.00
F-vortex	Air entrainment threshold Froude Number	0.80
FB10-nat	Nuclidic abundance of B10 isotope as a fraction of naturally occurring boron	0.1978
FB11-nat	Nuclidic abundance of B11 isotope as a fraction of naturally occurring boron	0.8022
K1	Conversion constant (BTU/hr-MWt)	3,412,000
K2	Conversion constant (BTU/min.-MWt)	56,868
K3	Conversion constant (gal/ft3)	7.48
M-B10	Nuclidic mass of B10 isotope	10.01
M-B11	Nuclidic mass of B11 isotope	11.01
M-B-nat	Atomic mass of natural boron (B)	10.81
M-H	Atomic mass of hydrogen (H)	1.01
M-Na	Atomic mass of sodium (Na)	22.99
M-O	Atomic mass of oxygen (O)	16
Material Strength	Temp (°F): Fyield –n: Ftens –n -- Normalized material strengths. Information to be provided for each material (1…n). Yield Tensile Temp Strength Strength (°F) Fyield-1 Ftens-n	
SRV Re-opening Pressure Table:	SRV Type: Re-opening Pressure (Psid)	

Table 18C-9

BWROG EPG/SAG Rev. 2 Appendix C: Assumed and Supplemental Data

PARAMETER	PARAMETER DEFINITION	VALUE
Assumed Data		
TdwMinX	Minimum drywell temperature (°F)	
TdwMaxX	Maximum drywell temperature (°F)	
TscMinX	Minimum suppression chamber temperature (°F)	
TscMaxX	Maximum suppression chamber temperature (°F)	
Supplemental Data		
Primary Containment Supplemental Data		
Hsc0	Elevation of bottom of suppression chamber (ft)	
HpcRef	Elevation of primary containment water level instrument zero (ft)	
nSRV	Total number of SRVs	18
TdwMaxInd	Maximum indicated drywell temperature (°C)	
SLC Supplemental Data		
FB10_spb	Atomic abundance of B10 isotope (boron enrichment) as a fraction of all boron in enriched boron used for alternate boron injection (needed only if enriched boron is used for alternate boron injection)	
Kslc_wl	SLC Accumulator volume conversion factor (gal/unit)	
MbxUnit	Borax container capacity (lbm borax) (needed only if borax and boric acid are used for alternate boron injection)	
MbaUnit	Boric acid container capacity (lbm boric acid) (needed only if borax and boric acid are used for alternate boron injection)	
MspbUnit	Sodium pentaborate container capacity (lbm sodium pentaborate (needed only if enriched sodium pentaborate is used for alternate boron injection)	
rhoCal	Density of solution in the SLC Accumulator assumed for calibration of SLC Accumulator level instrument (lbm/ft3) (needed only if SLC Accumulator level indication is affected by solution density)	

Table 18C-9

BWROG EPG/SAG Rev. 2 Appendix C: Assumed and Supplemental Data

PARAMETER	PARAMETER DEFINITION	VALUE
rhoSLC	Density of solution in the SLC Accumulator with boron concentration at XB-slc and temperature at Tslc (lbm/ft3)	
WLslcMin	Water level (minimum normal operating) in SLC tank	
WLslcUnits	SLC Accumulator level indicator units	
WLslcErr	Is the SLC Accumulator level indication affected by changes in solution density?	
AltBoronMethod	Alternate boron injection method:	
Pump NPSH Supplemental Data		
	Provide information for all pumps (1….n)	
Pump-n	Pump identification	
HminNPSH-n	Elevation of minimum suppression pool water level for pump operation within NPSH and vortex limits	
WloopMax-n	Maximum loop flow (gpm)	
hLRefTable-n	Suction piping reference head loss: % Loop Flow In Segment / Head Loss (ft) / Reference Flow (gpm) Strainer: Segment 1: Segment 2:	
NPSHTable-n	Required net positive suction head (ft) as a function of pump flow—maximum ten data points, from zero to maximum flow: Wpump (gpm) / NPSHreq (ft)	

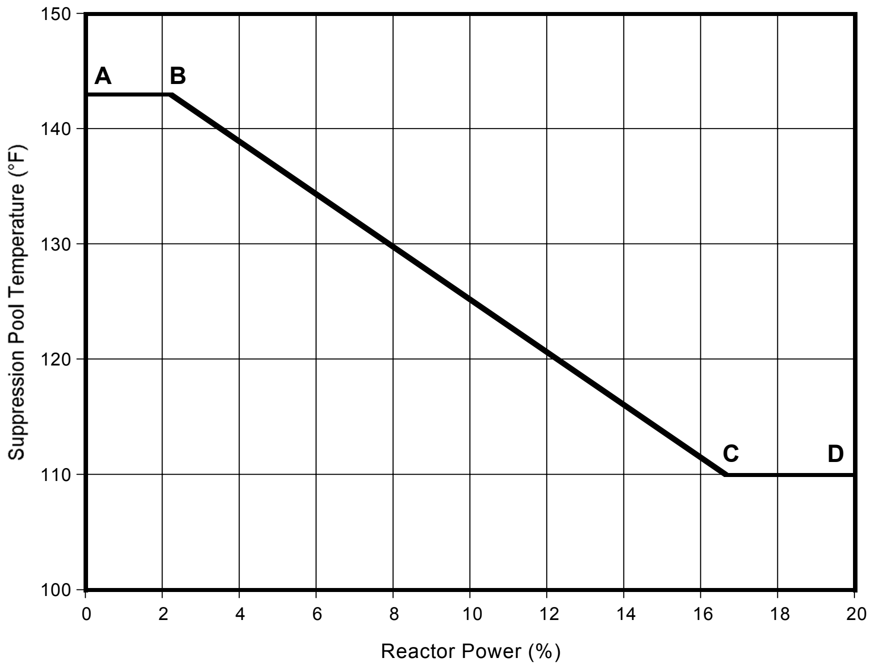

Figure 18C-1. Typical Boron Injection Initiation Temperature
(Plant Specific Operating Limit to be provided at COL)

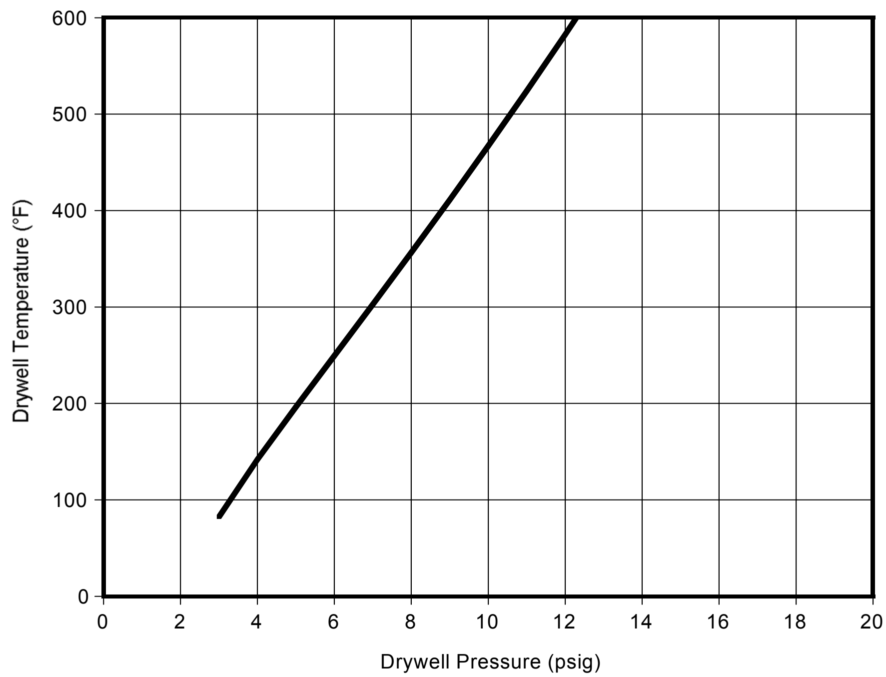

Figure 18C-2. Typical Drywell Spray Initiation Limit
(Plant Specific Operating Limit to Be Provided at COL)

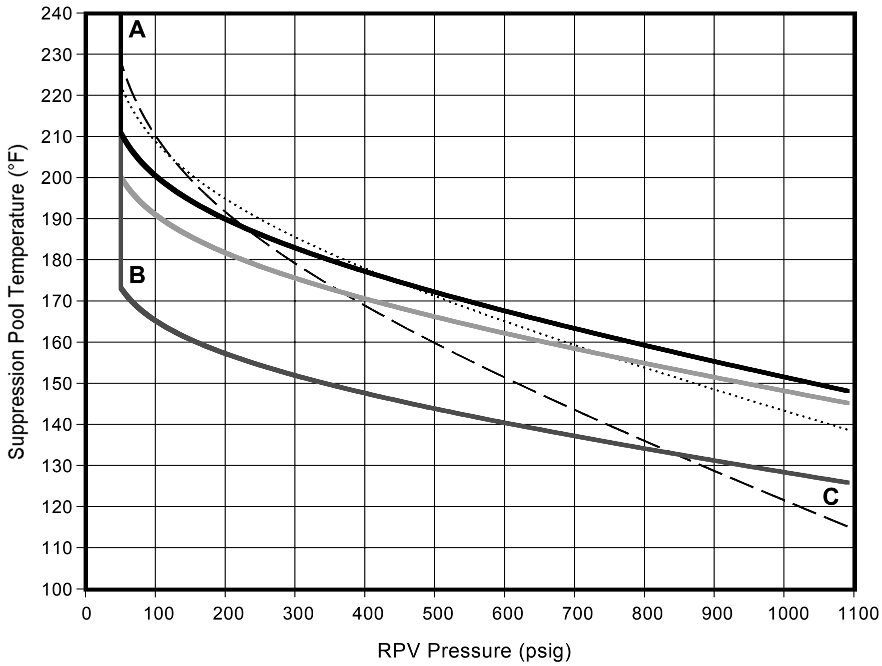

Figure 18C-3. Typical Heat Capacity Temperature Limit

(Plant Specific Operating Limit to Be Provided at COL)

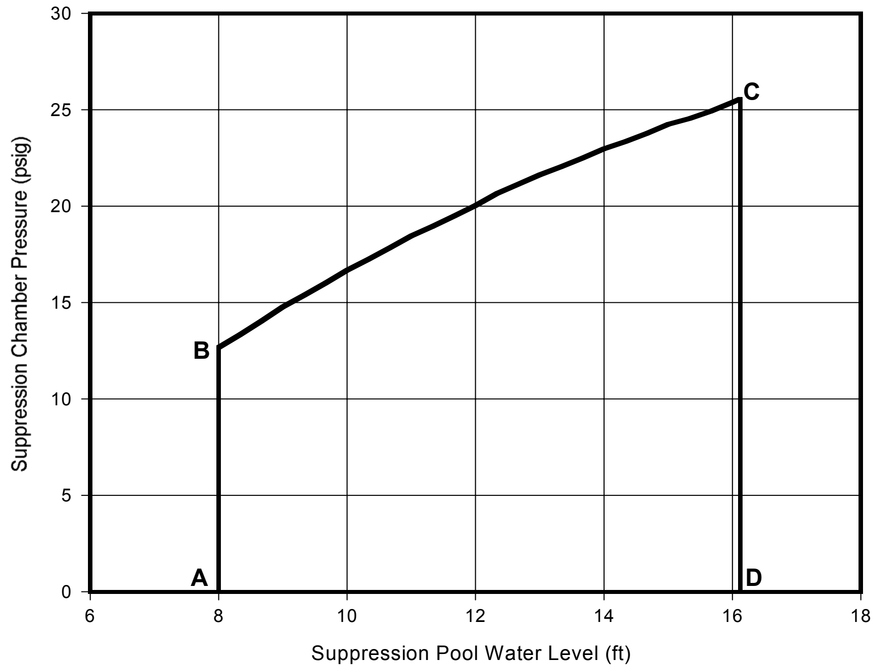

Figure 18C-4. Typical Pressure Suppression Pressure Curve
(Plant Specific Operating Limit to Be Provided at COL)

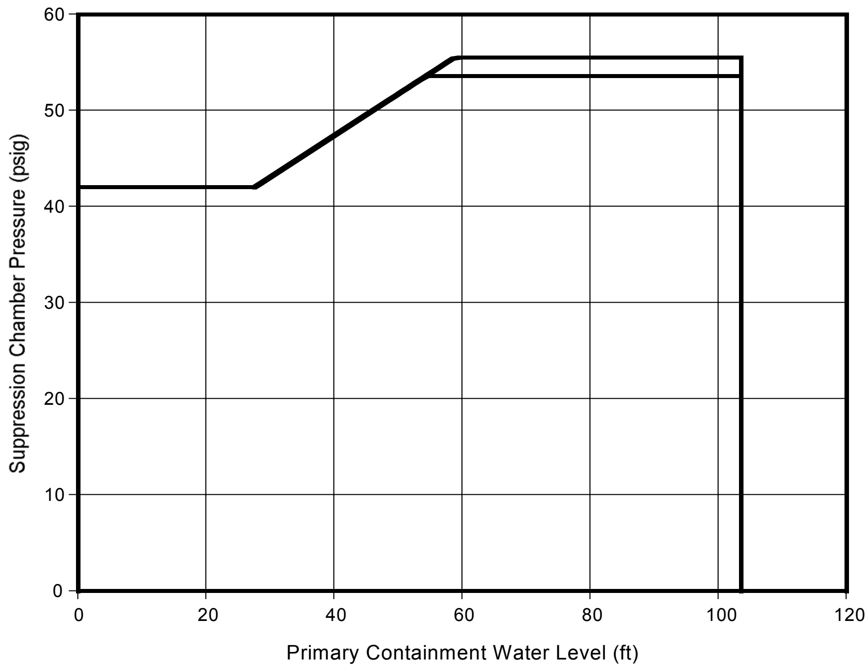

Figure 18C-5. Typical Containment Pressure Limit
(Plant Specific Operating Limit to Be Provided at COL)

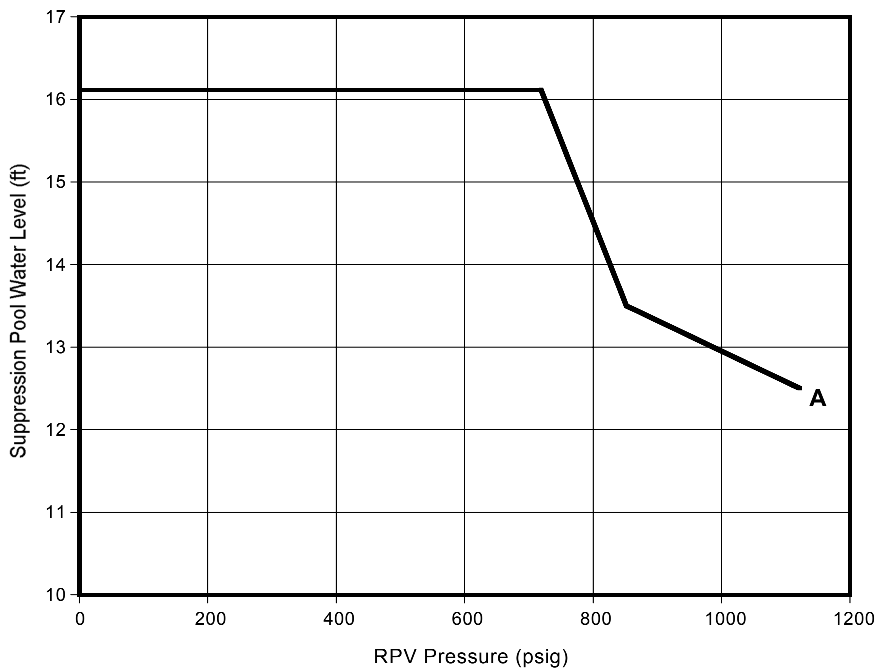

Figure 18C-6. Typical SRV Tail Pipe Level Limit
(Plant Specific Operating Limit to Be Provided at COL)

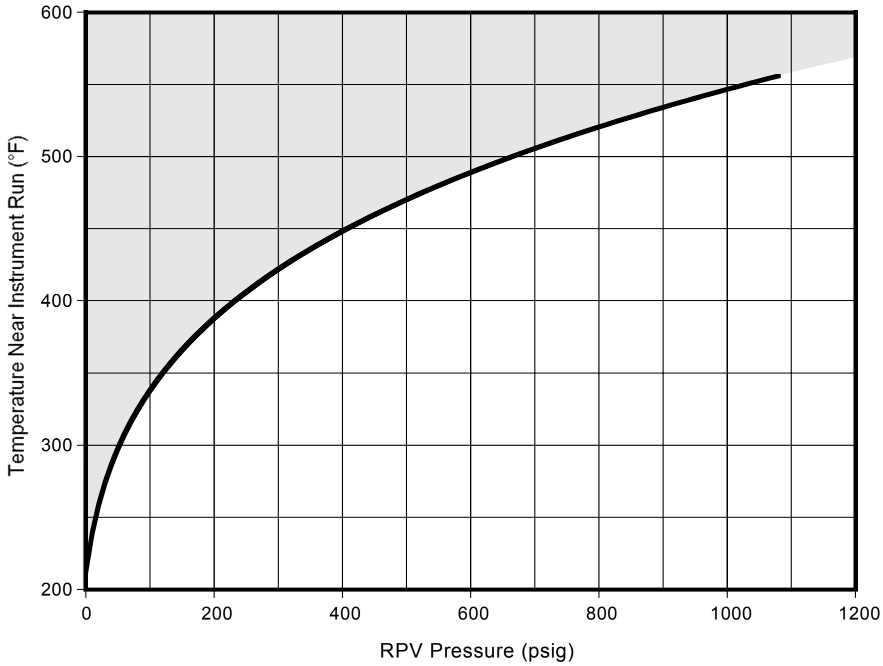

Figure 18C-7. Typical RPV Saturation Temperature

18D. OPERATOR INTERFACE EQUIPMENT CHARACTERIZATION

This appendix contains a characterization of a main control room operator interface that incorporates the standard design features presented in Subsection 18.3.2. The purpose of presenting this design characterization is to provide an illustration of an implemented ESBWR main control room operator interface and is not meant to impose any limits on the ESBWR design.

18D.1 CONTROL ROOM ARRANGEMENT

The conceptual main control room contains the main control console, the large display panel, the supervisor's console, the assistant shift supervisor's desk, a large table and various other desks, peripheral equipment and storage space. The arrangement of these items of equipment and furniture is shown in Figure 18D-1. The spatial arrangement of the main control console, large display panel and supervisor's console is a standard design feature, as discussed in Subsection 18.4.2.15. Figure 18D-1 illustrates this standard arrangement.

18D.2 MAIN CONTROL CONSOLE CONFIGURATION

The conceptual main control console is configured as shown in a plan view in Figure 18D-2. As shown in Figure 18D-2, the configuration is that of a shallow, truncated V with desk space attachments at the ends of both wings. The dimensions are such that two operators can comfortably work at the console at all times.

A cross-sectional view of the main console is shown in Figure 18D-3. This is a cross-section at points A-A, indicated in Figure 18D-2. This view gives an indication of the console height and the depth of the console desk surface. The dashed lines indicate the position of the computer driven VDUs, which, in this concept, are CRTs.

A second cross-sectional view, at points B-B, as indicated in Figure 18D-2, is shown in Figure 18D-4. This view shows the cross-sectional shape of the main console in the desk areas.

The details in Figure 18D-2 include the identification and arrangement of the equipment installed on the main control console. This equipment includes computer-driven CRTs, flat panel display devices, panels of dedicated function switches and analog displays for selected equipment (e.g., the main generator). The flat panel display devices are driven by dedicated microprocessors and, thus, are independent of the NE-DCIS.

In general, the conceptual equipment arrangement on the main console (plan view) is (1) safety-related and NSS on the left, (2) overall plant supervision in the center and (3) balance of plant (BOP) on the right.

The flat panel displays on the left side of the console are divisionally dedicated. These flat panels are qualified to Class 1E standards and are used to monitor and control the divisional safety-related systems.

The flat panels on the center and right panels of the main console support monitoring and control of nonsafety-related NSS and BOP systems.

The CRTs and flat panel display devices on the main control console are fitted with touch screens. In addition to the control capabilities provided by the touch screens on the CRTs and flat panels, there are panels of dedicated switches implemented in hardware and located on the main control console. Dedicated switches are discussed in Subsection 18.4.2.5.

18D.3 LARGE DISPLAY PANEL CONFIGURATION

The conceptual large display panel is approximately 3.2 meters high (10 ft. 6 in.) and 10.5 (34 ft. 5 in.) meters wide. In conformance with the standard design feature discussed in Subsection 18.4.2.7, it has three major components; the fixed-position (mimic) display, the top-level alarm display and the large VDU. There are also fixed-position alarm tiles positioned in the top portion of the fixed-mimic display. All displays on the large display panel are positioned to be viewed by the operators from a sitting position behind the main control console as shown in Figure 18D-5.

The fixed-position displays occupy the central portion of the large display panel and are discussed in Subsection 18.4.2.8. The fixed-position displays are driven by controllers that are independent of the NE-DCIS so that the fixed-position displays will continue to function normally in the event of NE-DCIS failure. [v40]The plant-level alarm display panel is at the left of the fixed-position displays as you face the large display panel. To the right of the fixed-position displays on the panel is the large VDU.

18D.4 SYSTEMS INTEGRATION

A characterization of the plant instrumentation and control systems architecture which supports the control room operator interface is illustrated in Figure 18D-6. As shown in Figure 18D-6, display and control capability for both safety-related and nonsafety-related systems are driven by microprocessors that are independent of the redundant NE-DCIS . This assures the ability to safely shut down the plant in the unlikely event of computer failure. In the case of the safety systems, the microprocessors are divisionally dedicated and are each electrically isolated from the rest of the system.

The plant-wide, fiber-optic essential distributed control and instrumentation system (E-DCIS) provides the communications network for the system. This multiplexing system is actually a series of data acquisition and control networks; separate networks being provided for safety-related and nonsafety-related plant systems.

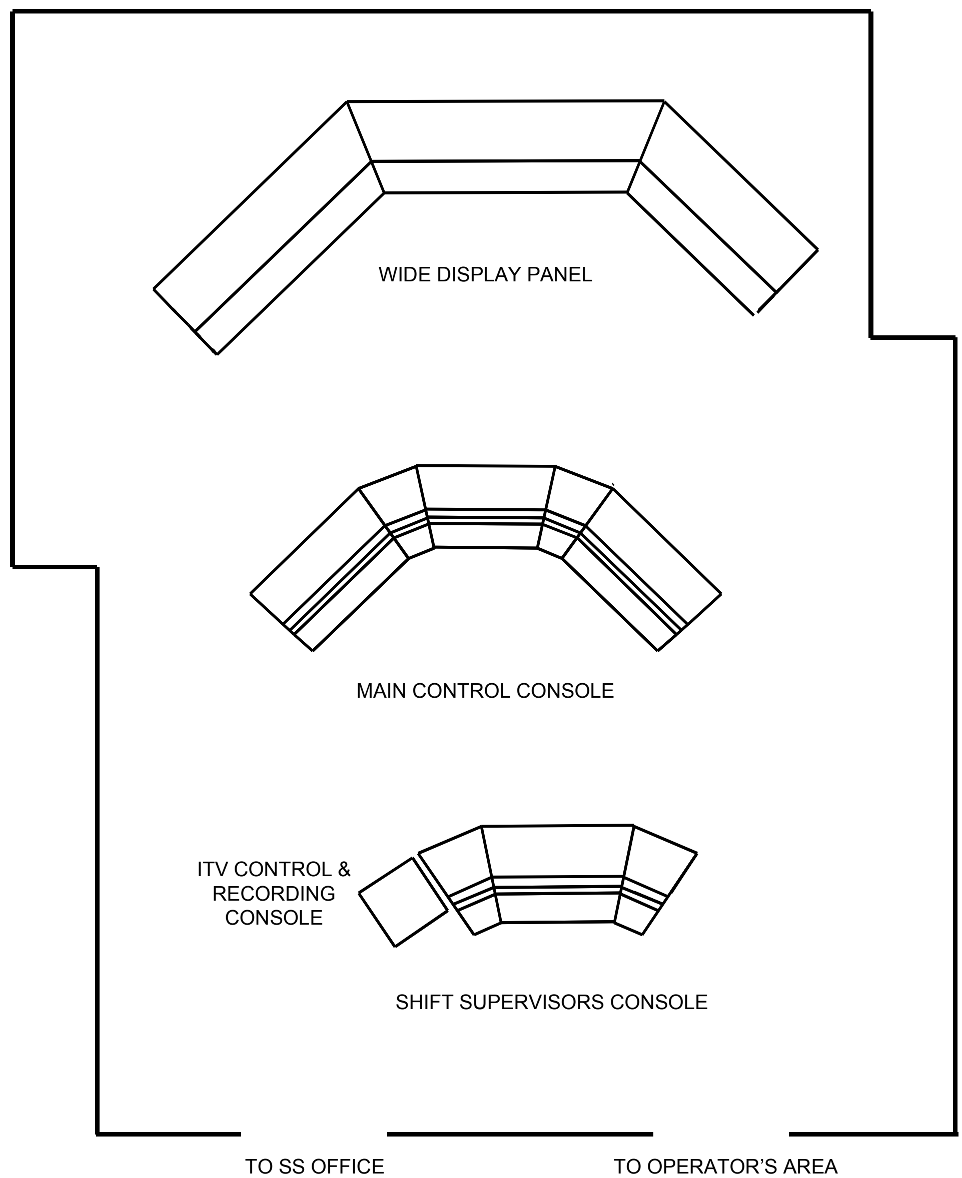

Figure 18D-1. Control Room Arrangement

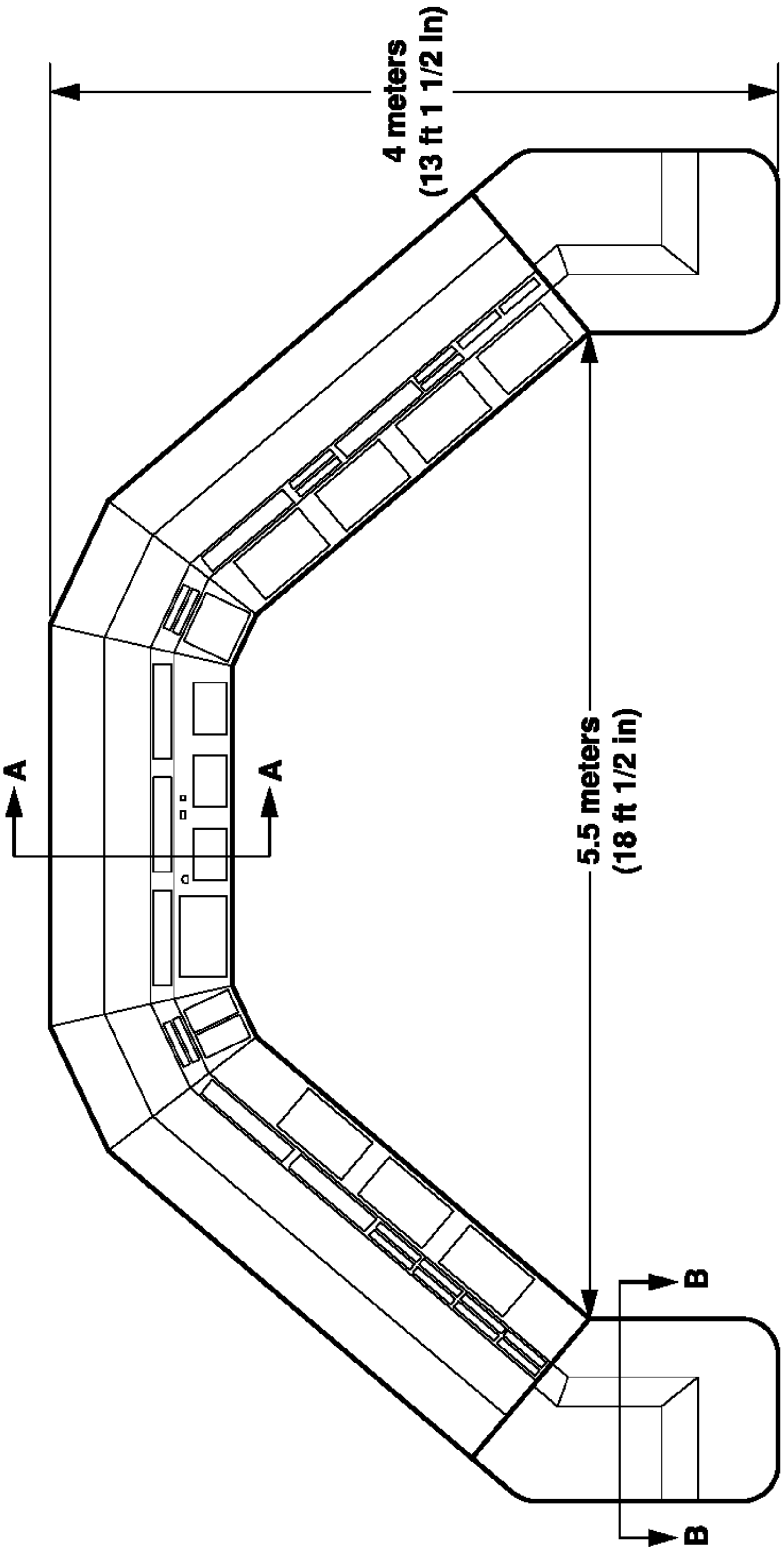

Figure 18D-2. Main Control Console Configuration

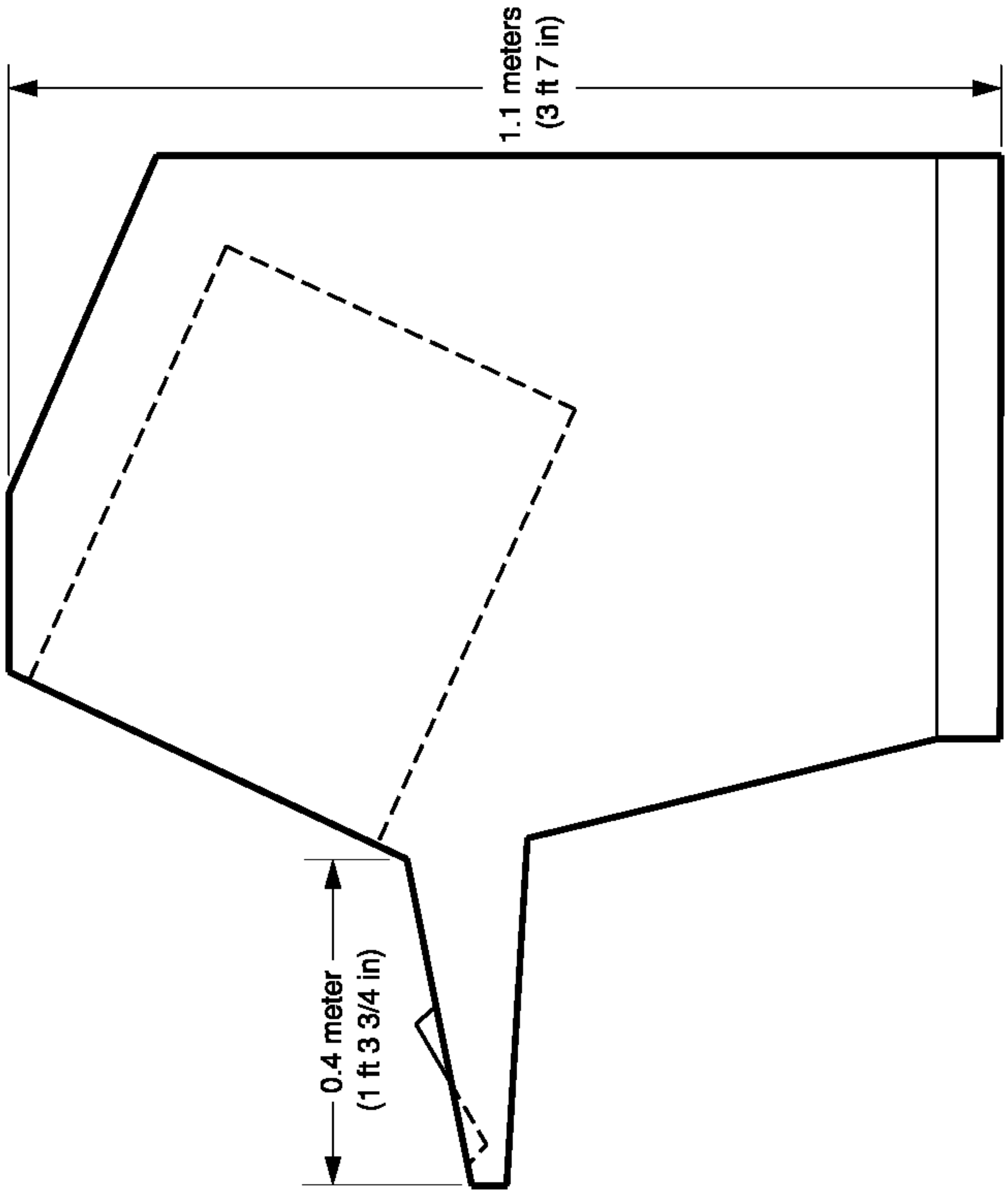

Figure 18D-3. Main Control Console Cross-Section A-A

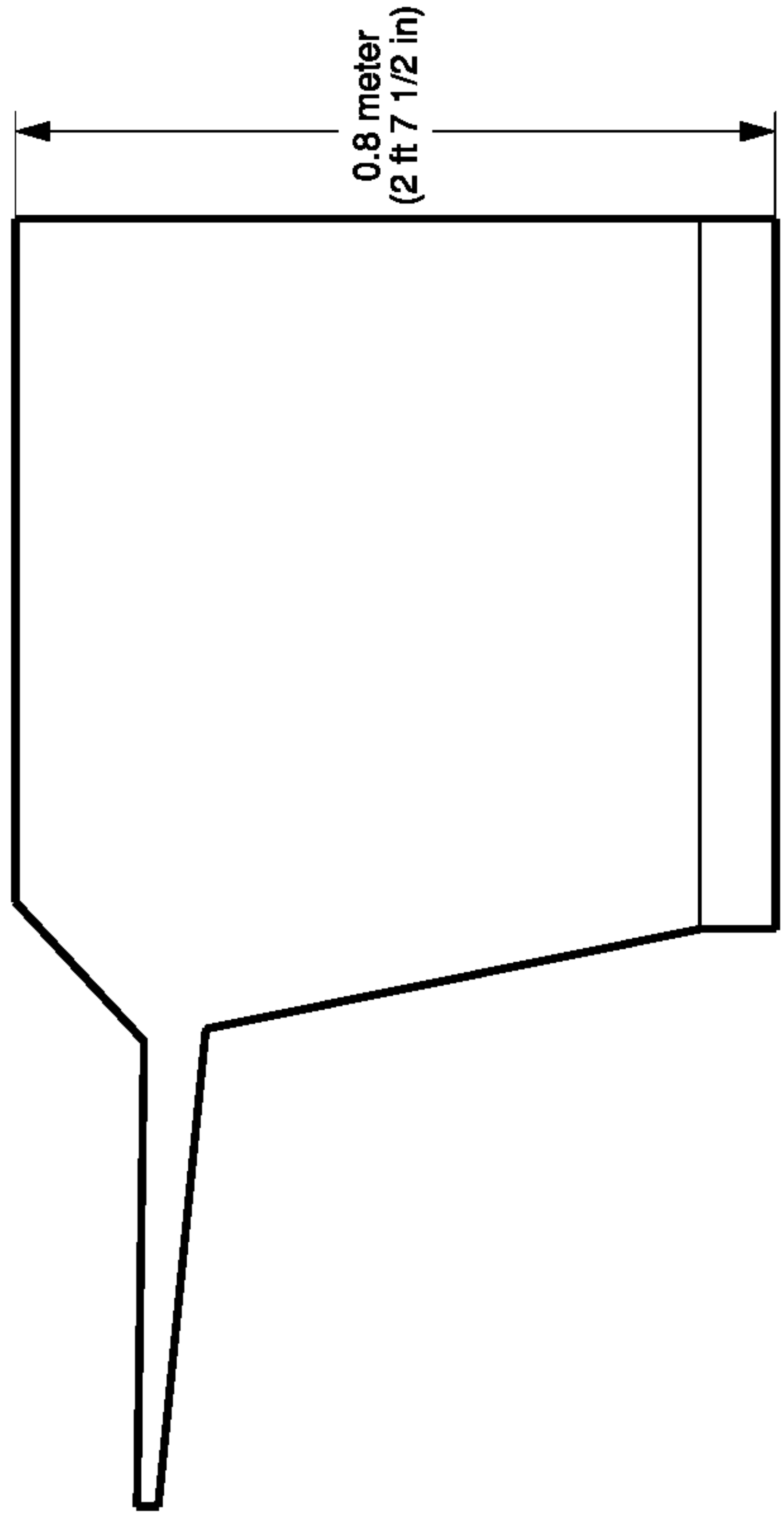

Figure 18D-4. Main Control Console Cross-Section B-B

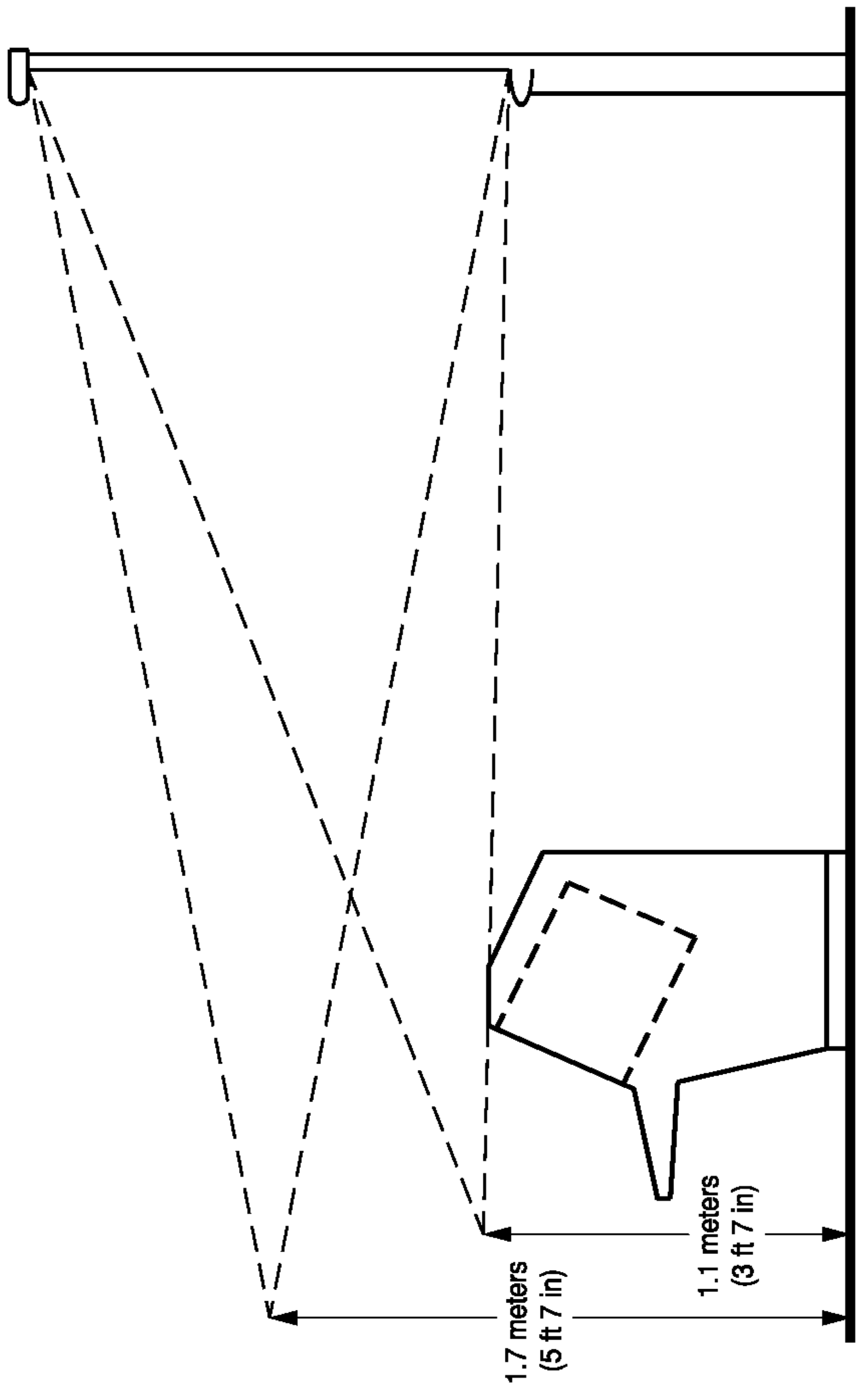

Figure 18D-5. Side View of Relative Positions of Main Console and Wide Display Device

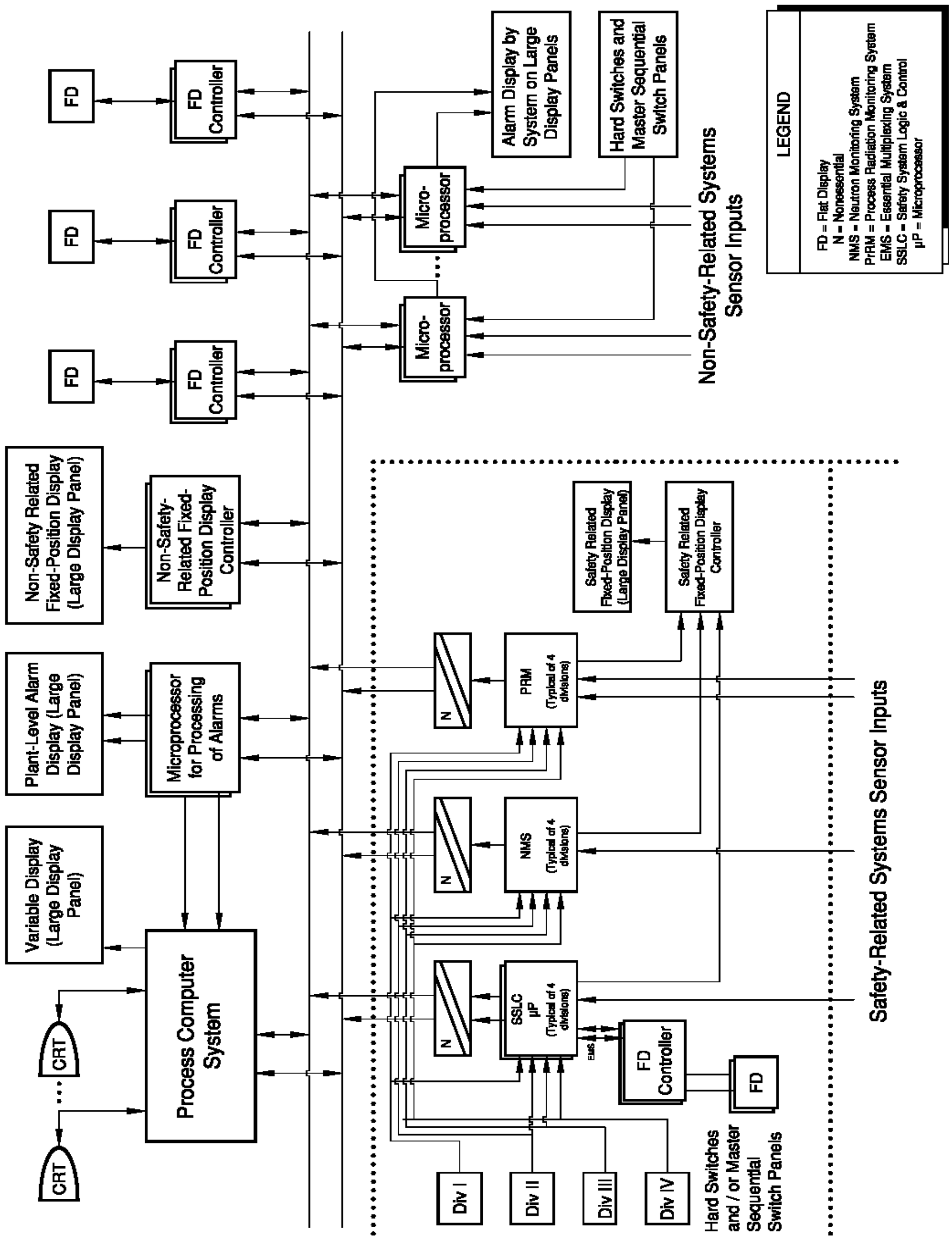

Figure 18D-6. Overall Configuration of Operator Interface System

18E. ESBWR HUMAN-SYSTEM INTERFACE DESIGN IMPLEMENTATION PROCESS

18E.1 INTRODUCTION

Section 18.3 discusses the program of human factors related activities conducted throughout the development of the ESBWR plant system designs, including the development of the Main Control Room (MCR) and Remote Shutdown System (RSS) designs. This appendix describes the process through which the MCR and RSS human system interface (HSI) design implementations are conducted and evaluated through the application of accepted Human Factors Engineering (HFE) practices and principles. Section 18E.2 discusses the basic elements of this HFE design implementation process and includes identification of where in the process the results are presented for NRC review. The review criteria to be used the review of the design implementation (i.e., the Design Acceptance Criteria (DAC)) are presented in Section 18E.3.

18E.2 HSI DESIGN IMPLEMENTATION PROCESS

The designs of the MCR and RSS areas of operator interface, for the execution of normal plant operation and emergency operation, will be implemented and evaluated in accordance with the process illustrated in Figure 18E-1. As shown in Figure 18E-1, the implementation process begins with the establishment of the Human Factors Engineering (HFE) Design Team which prepares the HFE Program and Implementation Plans and guides the process through the remaining steps to the final validation of the implemented design. Figure 18E-1 also identifies the relative timing of the planned NRC conformance reviews along with the corresponding table in Section 18E.3 that defines the acceptance criteria applicable to the individual reviews.

18E.2.1 The HFE Design Team

The HFE Design Team is composed of experienced individuals whose collective expertise covers a broad range of disciplines relevant to the design and implementation process. These disciplines include technical project management, control and instrument engineering, plant operations and architect engineering, as well as human factors engineering.

The duties of the HFE Design Team are to establish the HFE Program and Implementation Plans, to guide and oversee the design implementation process and to assure that the execution and documentation of each step in the process is carried out in accordance with the established program and procedures. The team will have the authority to insure that all its areas of responsibility are accomplished and to identify problems in the implementation of the HSI design. The team has the authority to determine where its input is required and to access work areas and design documentation. The team also has the authority to control further processing, delivery, installation or use of HFE/HSI products until the disposition of a non-conformance, deficiency or unsatisfactory condition has been achieved.

18E.2.2 The HFE Program and Implementation Plans

The HFE Design Team establishes the HFE Program and Implementation Plans that provide overall direction and integration of the HFE-related design implementation and evaluation activities for the specific HSI scope which includes the RSS and MCR areas of operational interface. The HFE Program Plan identifies the individuals who comprise the HFE Design Team and establishes the processes through which the HFE Design Team performs its functions. Included in the HFE Program Plan is a system for documenting human factors issues that may be identified throughout the implementation of the designs, and the actions taken to resolve those issues. The HFE Design Team also establishes the Implementation Plans for conducting each of the following HFE-related activities:

(1) System functional requirements analysis.

(2) Allocation of functions.

(3) Task analysis.

(4) Human-system interface design.

(5) Human factors verification and validation.

The Implementation Plans establish methods and criteria for the conduct of each of these HFE-related activities, which are consistent with accepted HFE practices and principles. (For additional detailed information regarding the scope and content of the HFE Program and Implementation Plans, refer to the acceptance criteria presented in Table 18E-1).

18E.2.3 System Functional Requirements Analysis

Analyses of the system functional requirements are conducted through application of the methods and criteria established by the HFE Design Team in the System Functional Requirements Analysis Implementation Plan. The system functional analysis determines the performance requirements and constraints of the HSI design and establishes the functions, which must be accomplished to meet these requirements. Safety functions are specifically identified along with any functional interrelationship that those safety functions may have with non-safety systems. In addition, critical functions (i.e., functions required to achieve major system performance requirements or functions which, if failed, could degrade system performance or pose a safety hazard to plant personnel or the general public) are identified. Detailed narrative descriptions are developed for each of the identified functions.

18E.2.4 Allocation of Functions

The functions defined through the function analysis are then allocated (i.e., defined as a function to be performed by the human, the system equipment or by a combination of the human and system equipment) per the methods and criteria established by the HFE Design Team in the Allocation of Functions Implementation Plan. The allocation of functions is done to take advantage of areas of human strengths and avoid allocating functions to personnel, which would be impacted by human limitations. The allocation of functions to personnel, systems or personnel-system combinations are made to reflect: sensitivity, precision, time and safety requirements, required reliability of system performance, and the number and level of skills of personnel required to operate and maintain the system.

As alternative allocation concepts are developed, analyses and trade-off studies are conducted to determine adequate configurations of personnel and system-performed functions. Analyses are done to confirm that the personnel elements can properly perform tasks that are allocated to them while maintaining proper operator situational awareness, workload and vigilance.

18E.2.5 Task Analyses

Following the completion of the function allocation step, task analysis is performed on those functions that are allocated to personnel. These task analyses are performed per the methods and criteria established by the HFE Design Team Task Analysis Implementation Plan. The task analysis identifies the behavioral requirements of the tasks associated with individual functions. Tasks are defined as groups of activities that have a common purpose, often occurring in temporal proximity, and which utilize the same displays and controls. The task analysis

(1) provides one of the bases for making design decisions (e.g., determining before hardware fabrication, to the extent practicable, whether system performance requirements can be met by combinations of anticipated equipment, software and personnel);

(2) assure that human performance requirements do not exceed human capabilities;

(3) are used as basic information for developing manning, skill, training and communications requirements of the system; and

(4) form the basis for specifying the requirements for the displays, data processing and controls needed to carry out the tasks.

The scope of the task analysis includes all operations performed at the operator interface in the main control room and at the Remote Shutdown System. The analysis is directed to the full range of plant operating modes, including startup, normal operations, abnormal operations, transient conditions, low power and shutdown conditions. The analysis also addresses operator interface operations during periods of maintenance test and inspection of plant systems and equipment and of the HSI equipment.

18E.2.6 Human-System Interface Design

As established by the HFE Design Team in their development of the HSI Design Implementation Plan, human engineering criteria are applied along with all other design requirements to select and design the particular equipment for application to the MCR and RSS HSI. The HSI design implements the information and control requirements that have been developed in the task analysis, including the displays, control and alarms necessary for the execution of those tasks identified in the task analyses as being critical tasks. The equipment design configuration satisfies the functional and technical design requirements and insures that the HSI is consistent with applicable HFE principles.

18E.2.7 Procedure Development

Plant and emergency operating procedures are developed to support and guide human interactions with plant systems and to control plant-related events and activities. Plant procedure development is discussed in Section 13.5.

18E.2.8 Human Factors Verification and Validation

Following the methods and criteria established by the HFE Design Team in the Human Factors Verification and Validation Plan, the successful incorporation of human factors engineering into the implemented HSI design and the acceptability of the resulting HSI is thoroughly evaluated as an integrated system.

The evaluations includes consideration of the HSI, the plant and emergency operating technical procedures and the overall work environment (e.g., lighting, ventilation, etc.). Individual HSI elements are evaluated in a static mode to assure that all controls, displays and data processing that were identified in the task analyses are available and that they are designed according to accepted HFE principles, practices, and criteria. In addition, the integration of HSI elements with each other and with personnel are evaluated and validated through dynamic task performance evaluation using evaluation tools such as a dynamic HSI prototype driven by real-time plant simulation models. The dynamic task performance evaluation is conducted over the full range of operational conditions and plant maintenance activities including: normal plant operation; plant system and equipment failures; HSI equipment failures; plant transients and postulated plant emergency conditions.

18E.2.9 HSI Implementation Requirements

Section 18E.2 describes the process through which the ESBWR Main Control Room (MCR) and Remote Shutdown System (RSS) areas of operator interface will be implemented and evaluated. Figure 18E-1 presents the relative timing of the NRC conformance reviews, which are planned throughout the MCR and RSS Human-System Interface (HSI) design implementation. Tables 18E-1 through 18E-5 of this section define the requirements that are to be met by the HSI design implementation activities that are to be made available for review by the NRC. The HSI design implementation-related Design Acceptance Criteria (DAC), which are established through Rulemaking, (refer to Section 3.3 of the Tier 1 Design Certification material for the GE ESBWR design) are defined such that there exists a direct correspondence between the DAC entries and requirements imposed herein on those design activities whose results are to be made available for the NRC conformance reviews, as identified in Figure 18E-1 Satisfaction of the specific requirements outlined in the five tables in Appendix E shall result in full compliance with the Certified Design Commitment and the corresponding Acceptance Criteria presented in the Tier 1 (Rulemaking) DAC established for the HSI design implementation.

Table 18E-1

Human Factors Engineering Design Team and Plans

18E.2.10 HFE Design Team Composition

(Satisfaction of the requirements presented herein shall result in the creation of an HFE Design Team which is in full compliance with the Item 1a Acceptance Criteria presented in Table 3.3 of the Tier 1 Design Certification Material for the GE ESBWR design).

(1) The composition of the Human Factor Engineering (HFE) Design Team shall include, as a minimum, the technical skills presented in Article (4), below.

(2) The education and related professional experience of the HFE Design Team personnel shall satisfy the minimum personal qualification requirements specified in Article (4), below, for each of the areas of required skills. In those skill areas where related professional experience is specified, qualifying experience of the individual HFE Design Team personnel shall include experience with previous plants in the main control room and Remote Shutdown System (RSS) Human System Interface (HSI) designs and design implementation activities. The required professional experiences presented in those personal qualifications of Article (4) are to be satisfied by the HFE Design Team as a collective whole. Therefore, satisfaction of the professional experience requirements associated with a particular skill area may be realized through the combination of the professional experience of two or more members of the HFE Design Team who each, individually, satisfy the other defined credentials of the particular skill area but who do not possess all of the specified professional experience. Similarly, an individual member of the HFE Design Team may possess all of the credentials sufficient to satisfy the HFE Design Team qualification requirements for two or more of the defined skill areas.

(3) Alternative personal credentials may be accepted as the basis for satisfying the minimum personal qualification requirements specified in Article (4), below. Acceptance of such alternative personal credentials shall be evaluated on a case-by-case basis and approved, documented and retained in auditable plant construction files by the COL applicant. The following factors are examples of alternative credentials which are considered acceptable:

(a) Professional Engineer's license in the required skill area may be substituted for the required Bachelor's degree.

(b) Related experience may substitute for education at the rate of six semester credit hours for each year of experience up to a maximum of 60 hours credit.

(c) Where course work is related to job assignments, post-secondary education may be substituted for experience at the rate of two years of education for one year experience. Total credit for post-secondary education shall not exceed two years experience credit.

(4) Required Skill Area Personal Qualification

(a)	Technical Project Management	Bachelor's degree, and five years experience in nuclear power plant design operations, and three years management experience.
(b)	Systems Engineering	Bachelor of Science degree, and four years cumulative experience in at least three of the following areas of systems engineering; design, development, integration, operation, and test and evaluation.
(c)	Nuclear Engineering	Bachelor of Science degree, and four years nuclear design, development, test or operations experience.
(d)	Instrumentation and Control (I&C) Engineering	Bachelor of Science degree, and four years experience in design of hardware and software aspects of process control systems, and experience in at least one of the following areas of I&C engineering; development, power plant operations, and test and evaluation, and familiarity with the theory and practice of software quality assurance and control.
(e)	Architect Engineering	Bachelor of Science degree, and four years power plant control room design experience.
(f)	Human Factors Engineering	Bachelor's degree in Human Factors Engineering, Engineering Psychology or related science, and four years cumulative experience related to the human factors aspects of human-computer interfaces. Qualifying experience should include at least the following activities within the context of large-scale human-machine systems (e.g. process control): design, development, and test and evaluation, and four years cumulative experience related to the human factors field of ergonomics. Qualifying experience shall include experience in at least two of the following areas of human factors activities; design, development, and test and evaluation.
(g)	Plant Operations	Have or have held a Senior Reactor Operator license; two years experience in BWR nuclear power plant operations.
(h)	Computer System Engineering	Bachelor's degree in Electrical Engineering or Computer Science, or graduate degree in other engineering discipline (e.g., Mechanical

		Engineering or Chemical Engineering), and four years experience in the design of digital computer systems and real time systems applications.
(i)	Plant Procedure Development	Bachelor's degree, and four years experience in developing nuclear power plant operating procedures.
(j)	Personnel Training	Bachelor's degree, and four years experience in the development of personnel training programs for power plants, and experience in the application of systematic training development methods.
(k)	System Safety Engineering	Bachelor's degree, and four years of experience in system safety engineering.
(l)	Maintainability/Inspectability Engineering	Bachelor's degree, and four years cumulative experience in at least two of the following areas of power plant maintainability and inspectability engineering activity: design, development, integration, and test and evaluation.
(m)	Reliability/Availability Engineering	Bachelor's degree, and four years cumulative experience in at least two of the following areas of power plant reliability engineering activity: design, development, integration, and test and evaluation, and knowledge of computer-based human interface systems.

(II) Human Factors Engineering Program Plan

(1) (Satisfaction of the requirement is presented herein shall result in the creation of a Human Factors Engineering Program Plan which is in full compliance with the Item 1.b. Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design.) The Human Factors Engineering (HFE) Program Plan shall establish:

(a) Methods and criteria, for the development and evaluation of the Main Control Room (MCR) and Remote Shutdown System (RSS) HSI, which are consistent with accepted HFE practices and principles. Within the defined scope and content of the HFE Program Plan, accepted HFE methods and criteria are presented in the following documents:

(i) AR 602-1, Human Factors Engineering Program. (Dept. of Defense)

(ii) DI-HFAC-80740, Human Engineering Program Plan. (Dept. of Defense)

(iii) DOD-HDBK-763, Human Engineering Procedures Guide, Chapters 5-7 and Appendices A and B. (Dept. of Defense)

(iv) EPRI NP-3659, Human Factors Guide for Nuclear Power Plant Control Room Development, 1984. (Electric Power Research Institute)

(v) IEEE-1023, IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations, 1988, (IEEE)

(vi) MIL-H-46855B, Human Engineering Requirements for Military Systems, Equipment and Facilities (Dept. of Defense)

(vii) NUREG-0700, Guidelines for Control Room Design Reviews 2002. (US Nuclear Regulatory Commission)

(viii) NUREG-0737, Clarification of TMI Action Plan Requirements (Item I.C.5, "Feedback of Operating Experience to Plant Staff"), 1983 (US Nuclear Regulatory Commission)

(ix) NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures, 1982, (US Nuclear Regulatory Commission)

(x) NUREG/CR-3331, A Methodology for Allocating Nuclear Power Plant Control Functions to Human and Automated Control, 1983 (US Nuclear Regulatory Commission)

(xi) TOP 1-2-610, Test Operating Procedure Part 1. (Dept. of Defense)

IEEE-566, Recommended Practices for the Design of Display and Control Facilities for Central Control Room for Nuclear Power Generating Stations.

Note that within the set of documents listed above, differences may exist regarding specific methods and criteria applicable to the HFE Program Plan. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those documents which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for how that particular issue is addressed in the HFE Program.

(b) The methods for addressing:

(i) The ability of the operating personnel to accomplish assigned tasks

(ii) Operator workload levels and vigilance

(iii) Operating personnel "situation awareness

(iv) The operator's information processing requirements

(v) Operator memory requirements

(vi) The potential for operator error

(c) HSI design and evaluation scope that applies to the Main Control Room (MCR) and Remote Shutdown System (RSS).

The HSI scope shall address normal, abnormal and emergency plant operations and test and maintenance interfaces that impact the function of the operations personnel. The HSI scope shall also address the development of operating technical procedures for normal, abnormal and emergency plant operations and the identification of personnel training needs applicable to the HSI design. The developments of operating technical procedures are a COL action item (see Section 13.5). The establishment of an operator-training program, which meets the requirements of 10 CFR 50, is also a COL license information requirement (see Subsection 18.8.8).

(d) The HFE Design Team as being responsible for:

(i) The development of HFE plans and procedures

(ii) The oversight and review of HFE design, development, test, and evaluation activities

(iii) The initiation, recommendation, and provision of solutions through designated channels for problems identified in the implementation of the HFE activities

(iv) Verification of implementation of solutions to problems

(v) Assurance that HFE activities comply to the HFE plans and procedures

(vi) Phasing of activities

(e) The methods for identification, closure and documentation of human factors issues.

(f) The HSI design configuration control procedures.

(2) The HFE Program Plan shall also establish:

(a) That each HFE issue/concern shall be entered on the HFE Issue Tracking System log when first identified, and each action taken to eliminate or reduce the issue/concern should be documented. The final resolution of the issue/concern, as accepted by the HFE Design Team, shall be documented along with information regarding HFE Design Team acceptance (e.g., person accepting, date, etc.) the individual responsibilities of the HFE Design Team members when an HFE issue/concern is identified, including definition of who should log the item, who is responsible for tracking the resolution efforts, who is responsible for acceptance of a resolution, and who shall enter the necessary closeout data.

(b) That the HFE Issue Tracking System shall address human factors issues that are identified throughout the development and evaluations of the Main Control Room and Remote Shutdown System HSI design implementation.

(c) That the MCR and RSS designs shall be implemented using HSI equipment technologies that are consistent with those defined in Section 18.4.3.

(d) That in the event other HSI equipment technologies are alternatively selected for application in the MCR and RSS design implementations:

(i) A review of the industry experience with the operation of those selected new HSI equipment technologies shall be conducted.

(ii) The Operating Experience Review (OER) of those new HSI equipment technologies shall include both a review of literature pertaining to the human factors issues related to similar system applications of those new HSI equipment technologies and interviews with personnel experienced with the operation of those systems.

(iii) Any relevant HFE issues/concerns associated with those selected new HSI equipment technologies, identified through the conduct of the OER, shall be entered into the HFE Issue Tracking System for closure.

(e) That a review of HSI operating experience shall be conducted as follows:

(i) For the first implementation of the ESBWR Certified Design:

(a) That the lessons learned from the review of previous nuclear plant HSI designs, as defined by Attachment 1 to this Table 18E-1, shall be entered into the HFE Issue Tracking System to assure that problems observed in previous designs have been adequately addressed in the ESBWR design implementation.

(b) Reviews of operating experience with the following ESBWR HSI design areas, in which further development of the industry's experience base can be expected, shall be completed

— Use of flat panel and CRT displays.

— Use of electronic on-screen controls.

— Use of wide display panels.

— Use of prioritized alarm systems.

— Automation of process systems.

— Operator workstation design integration.

Those operating experience reviews shall include review of reports provided by industry organizations (i.e., EPRI, etc.); review of applicable research in these design areas, as may be documented in reports from universities, national laboratories and the NRC, and in proceedings published by HFE professional societies; and review of applicable research and experience reports published by the HSI equipment vendors. Further, the review of operating experience in each of the six above identified areas shall include feedback obtained from actual users. Therefore, if the documents selected for the conduct of the operating experience review for a particular area do not include the results of user feedback, then interviews with users of at least two applications of that particular technology area shall also be conducted. Finally, the results from all these operating experience review activities shall be entered into the HFE Issue Tracking System to assure that the ESBWR implementation reflects the experience gained by the resolution of design problems in operating plants.

(ii) For all subsequent implementations of the ESBWR design:

(a) If a previously implemented ESBWR HSI design is utilized directly and without change, then no further review of operating experience is required.

(b) If a previously implemented ESBWR HSI design is not being utilized directly, then the operating experience of the most recent implementations, up to three, shall be reviewed through the conduct of operator interviews and surveys and the evaluation of Licensing Event Reports and the results of these reviews shall be entered into the HFE Issue Tracking System to assure that previous design problems have been adequately addressed in the ESBWR design implementation.

(3) The HFE Program Management Plan document shall include:

(a) The purpose and organization of the plan.

(b) The relationship between the HFE program and the overall plant equipment procurement and construction program (organization and phasing).

(c) Definition of the HFE Design Team and their activities, including:

(i) Description of the HFE Design Team function within the broader scope of the plant equipment procurement and construction program, including charts to show organizational and functional relationships, reporting relationships, and lines of communication.

(ii) Description of the responsibility, authority and accountability of the HFE Design Team organization.

(iii) Description of the process through which management decisions will be made regarding HFE.

(iv) Description of the process through which the HFE Design Team will make technical decisions.

(v) Description of the tools and techniques (e.g., review forms, documentation) to be utilized by the HFE Design Team in fulfilling their responsibilities.

(vi) Description of the HFE Design Team staffing, job descriptions of the individual HFE Design Team personnel and their personal qualifications.

(vii) Definition of the procedures that will govern the internal management of the HFE Design Team.

(d) Definition of the HFE Issue Tracking System and its implementation, including:

(i) Individual HFE Design Team member responsibilities regarding HFE issue identification, logging, issue resolution, and issue closeout.

(ii) Procedures and documentation requirements regarding HFE issue identification. These shall include description of the HFE issue, effects of the issue if no design change action is taken and an assessment of the criticality and likelihood of the identified HFE issue manifesting itself into unacceptable HSI performance.

(iii) Procedures and documentation requirements regarding HFE issue resolution. These procedures shall include evaluation and documentation of proposed solutions, implemented solutions, evaluated residual effects of the implemented solution and the evaluated criticality and likelihood of the implemented resolution of the HFE issue manifesting itself into unacceptable HSI performance.

(e) Identification and description of the following implementation plans to be developed:

(i) System Functional Requirements Development.

(ii) Allocation of Function.

(iii) Task Analysis.

(iv) Human-System Interface Design.

(v) Human Factors Verification and Validation.

(f) Definition of the phasing of HFE program activities, including:

(i) The plan for completion of HFE tasks which addresses the relationships between HFE elements and activities, the development of HFE reports and the conduct of HFE reviews

(ii) Identification of other plant equipment procurement and construction activities that are related to HFE Design Team activities but outside the scope of the team (e.g., I&C equipment manufacture)

(g) Definition of HFE documentation requirements and procedures for retention and retrieval.

(h) Description of the manner in which HFE Program requirements will be communicated to applicable personnel and organizations, including those which may be subcontracted, who are responsible for the performance of work associated with the Main Control Room and Remote Shutdown System design implementation.

(III) System Functional Requirements Analysis Implementation Plan

(1) (Satisfaction of the requirements presented herein shall result in the creation of a System Functional Requirements Analysis Implementation Plan which is in full compliance with the Item 2.a acceptance criteria presented in Table 3.1 of the Tier 1

Design Certification material for the GE ESBWR design). The System Functional Requirements Analysis Implementation Plan shall establish:

(a) Methods and criteria for conducting the System Functional Requirements Analysis which are consistent with accepted HFE practices and principles. Within the context of system functional requirements analysis, accepted HFE methods and criteria are presented in the following documents:

(i) AD/A233 168, System Engineering Management Guide, (Dept. of Defense, Defense Systems Management College, Kockler, F., et al)

(ii) AR602-1, Human Factors Engineering Program, (Dept. of Defense)

(iii) EPRI NP-3659, Human Factors Guide for Nuclear Power Plant Control Room Development, 1984, (Electric Power Research Institute)

(iv) IEC 964, Design for Control Rooms of Nuclear Power Plants, 1989, (Bureau Central de la Commission Electro technique International)

(v) IEEE-1023, IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations, 1988, (IEEE)

(vi) MIL-H-46855B, Human Engineering Requirements for Military Systems, Equipment and Facilities, (Dept. of Defense)

(vii) NUREG-0700, Guidelines for Control Room Design Reviews, 2002, (US Nuclear Regulatory Commission)

(viii) NUREG/CR-3331, A Methodology for Allocating Nuclear Power Plant Control Functions to Human and Automated Control, 1983, (US Nuclear Regulatory Commission)

Note that within the set of documents listed above, differences may exist regarding the specific methods and criteria applicable to the conduct of system functional requirements analysis. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those document which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for defining how that particular issue is addressed in the system functional requirements analysis.

(b) That system requirements shall define the system functions and those system functions shall provide the basis for determining the associated HSI performance requirements.

(c) That functions critical to safety shall be defined (i.e., those functions required to achieve safety system performance requirements; or those functions which, if failed, could pose a safety hazard to plant personnel or to the general public).

(d) That descriptions shall be developed for each of the identified functions and for the overall system configuration design itself. Each function shall be identified and described in terms of inputs (observable parameters which will indicate systems status) functional processing (control process and performance measures required to achieve the function), functional operations (including detecting signals, measuring information, comparing one measurement with another, processing information, and acting upon decisions to produce a desired condition or result such as a system or component operation actuation or trip) outputs, feedback (how to determine correct discharge of function), and interface requirements so that sub functions are related to larger functional elements.

(2) The System Functional Requirements Analysis Implementation Plan shall include:

(a) The methods for identification of system level functions based upon system performance requirements. The functions shall be defined as the most general, yet differentiable means whereby the system requirements are met, discharged, or satisfied. Functions shall be arranged in a logical sequence so that any specified operational usage of the system can be traced in an end-to-end path.

(b) The methods for developing graphic function descriptions (e.g., Functional Flow Block Diagrams and Time Line Diagrams). The functions shall be described initially in graphic form. Function diagramming shall be done starting at a "top level", where major functions are described, and continuing to decompose major functions to lower levels until a specific critical end-item requirement emerges (e.g., a piece of equipment, software, or an operator).

(c) The method for developing detailed function narrative descriptions, which encompass:

(i) Observable parameters that indicate system status.

(ii) Control process and data required to achieve the function.

(iii) How to determine the manner in which proper discharge of function is to be determined.

(d) Analysis methods that define the integration of closely related sub functions so that they can be treated as a unit.

(e) Analysis methods that divide identified sub functions into two groups according to whether:

(i) Common achievement of the sub function is an essential condition for the accomplishment of a higher level function.

(ii) The sub function is an alternative supporting function to a higher-level function or the sub function accomplishment is not necessarily a requisite for a higher level function.

(f) Requirements to identify for each integrated sub function:

(i) The basis for why accomplishment of the sub function is required.

(ii) The control actions necessary for accomplishment of the sub functions.

(iii) The parameters necessary for the sub function control actions.

(iv) The criteria for evaluating the results of the sub function control actions.

(v) The parameters necessary for evaluation of the sub function.

(vi) The criteria to be used to evaluate the sub function.

(vii) The criteria for selecting alternative function assignments if the evaluation criteria b is not satisfied

(IV) Allocation of Function Implementation Plan

(1) (Satisfaction of the requirements presented herein shall result in the creation of an Allocation of Function Implementation Plan which is in full compliance with the Item 3.a Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The Allocation of Function Implementation Plan shall establish:

(a) The methods and criteria for the execution of function allocation, which are consistent with accepted HFE practices and principles. Within the context of function allocation, accepted HFE practices and principles are presented in the following documents:

(i) AD/A223 168, System Engineering Management Guide, (Dept. of Defense, Defense Systems Management College, Kockler, F., et al)

(ii) AR 602-1, Human Factors Engineering Program, (Dept. of Defense)

(iii) EPRI NP-3659, Human Factors Guide for Nuclear Power Plant Control Room Development, 1984, (Electric Power Research Institute)

(iv) IEC 964, Design for Control Rooms of Nuclear Power Plants, 1989, (Bureau Central de la Commission Electro technique International)

(v) NUREG-0700, Guidelines for Control Room Design Reviews, 2002, (US Nuclear Regulatory Commission)

(vi) NUREG/CR-3331, A Methodology for Allocating Nuclear Power Plant Control Functions to Human and Automated Control, 1983, (US Nuclear Regulatory Commission)

Note that within the set of documents listed above, differences may exist regarding the specific methods and criteria applicable to the conduct and analysis of function allocation. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those documents which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for defining how the particular issue is to be addressed in the conduct of the function allocation and analysis.

(b) That aspects of system and functions definition shall be analyzed in terms of resulting human performance requirements based on the expected user population.

(c) That the allocation of functions to personnel, system elements, and personnel system combinations shall reflect:

(i) Areas of human strengths and limitations

(ii) Sensitivity, precision, time, and safety requirements

(iii) Reliability of system performance

(iv) The number and the necessary skills of the personnel required to operate and maintain the system

(d) That the allocation criteria, rationale, analyses, and procedures shall be documented.

(e) Analyses shall confirm that the personnel can perform tasks allocated to them while maintaining operator situation awareness, acceptable personnel workload, and personnel vigilance.

(2) The Allocation of Function Implementation Plan shall include:

(a) Establishment of a structured basis and criteria for function allocation.

(b) Definition of function allocation analyses requirements, including:

(i) Definition of the objectives and requirements for the evaluation of function allocations

(ii) Development of alternative function allocations for use in the conduct of comparative evaluations

(iii) Development of criteria to be used as the basis for selecting between alternative function allocations

(iv) Development of evaluation criteria weighing factors

(v) Development of test and analysis methods for evaluating function allocation alternatives

(vi) Definition of the methods to be used in conducting assessments of the sensitivity of the comparative function allocation alternatives analyses results to the individual analysis inputs and criteria

(vii) Definition of the methods to be employed in selecting individual function allocation for incorporation into the implemented design

(V) Task Analysis Implementation Plan

(1) (Satisfaction of the requirements presented herein shall result in the creation of a Task Analysis Implementation Plan which is in full compliance with the Item 4.a Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The Task Analysis Implementation Plan shall establish:

(a) The methods and criteria for conduct of the task analyses which are consistent with accepted HFE practices and principles. Within the context of performing task analysis, accepted HFE methods and criteria are presented in the following documents:

(i) AD/A223 168, System Engineering Management Guide, (Dept. of Defense, Defense Systems Management College, Kockler, F., et al)

(ii) DOD-HDBK-763, Human Engineering Procedures Guide, Chapters 5-7 and Appendices A and B, 1991, (Dept. of Defense)

(iii) EPRI NP-3659, Human Factors Guide for Nuclear Power Plant Control Room Development, 1984, (Electric Power Research Institute)

(iv) IEC 964, Design for Control Rooms of Nuclear Power Plants, 1989, (Bureau Central de la Commission Electro technique International)

(v) IEEE-1023, IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations, 1988, (IEEE)

(vi) MIL-H-46855B, Human Engineering Requirements for Military Systems, Equipment and Facilities, (Dept. of Defense)

(vii) MIL-STD-1478, Task Performance Analysis, (Dept. of Defense)

(viii) NUREG-0700, Guidelines for Control Room Design Reviews, 2002, (US Nuclear Regulatory Commission)

(ix) NUREG/CR-3331, A Methodology for Allocating Nuclear Power Plant Control Functions to Human and Automated Control, 1983, (US Nuclear Regulatory Commission)

(x) NUREG/CR-3371, Task Analysis of Nuclear Power Plant Control Room Crews (Vol. 1), 1983, (US Nuclear Regulatory Commission)

Note that within the set of documents listed above, differences may exist regarding the specific methods and criteria applicable to the conduct of HFE task analysis. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those documents which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for defining how that particular issue is addressed in the task analysis.

(b) The scope of the task analysis, which shall include operations performed at the operator interface in the Main Control Room and at the Remote Shutdown System. The analyses shall be directed to the full range of plant operating modes, including startup, normal operations, abnormal operations, transient conditions, low power and shutdown conditions. The analyses shall also address operator interface operations during periods of maintenance, test and inspection of plant systems and equipment, including the HSI equipment.

(c) That the analysis shall link the identified and described tasks in operational sequence diagrams. The task descriptions and operational sequence diagrams shall be used to identify which tasks are critical to safety in terms of importance for function achievement, potential for human error, and impact of task failure. Human actions, which are identified through PRA sensitivity analyses to have significant impact on safety, shall also be considered “critical” tasks. Where critical functions are automated, the analyses shall address the associated human tasks including the monitoring of the automated function and the backup manual actions which may be required if the automated function fails.

(d) Task analysis shall develop narrative descriptions of the personnel activities required for successful completion of the task. A task shall be a group of activities, often occurring in temporal proximity, which utilize a common set of displays and controls. Task analyses shall define the input, process, and output required by and of personnel.

(e) The task analysis shall identify requirements for alarms, displays, data processing, and controls.

(f) The task analysis results shall be made available as input to the personnel training programs.

(2) The Task Analysis Implementation Plan shall include:

(a) The methods and data sources to be used in the conduct of the task analysis.

(b) The methods for conducting the initial (high level) task analysis, including:

(i) Converting functions to tasks.

(ii) Developing narrative task descriptions.

(iii) Developing the basic statement of the task functions.

(iv) Decomposition of tasks to individual activities.

(v) Development of operational sequence diagrams.

(c) The methods for developing detailed task descriptions that address:

(i) Information requirements. (i.e., information required to execute a task, including cues for task initiation)

(ii) Decision-making requirements (i.e., decisions that are probably based on the evaluations, description of the decisions to be made and the evaluations to be performed)

(iii) Response requirements (i.e., actions to be taken, frequency of action, speed/time line requirements, any tolerance/accuracy requirements associated with the action, consideration of any operational limits of personnel performance or of equipment body movements required by an action taken, and any overlap of task requirements such as serial vs. parallel task elements)

(iv) Feedback requirements (i.e., feedback required to indicate adequacy of actions taken)

(v) Personnel workload (i.e., both cognitive and physical workload and the estimation of the level of difficulty associated with a particular workload condition)

(vi) Any associated task support requirements (i.e., special/protective clothing, job aids or reference materials required; any tools and equipment required, or any computer processing support aids)

(vii) Workplace factors (i.e., the workspace envelope required by the action taken, workspace environmental conditions, location that the work is to be performed, the physical/mental attributes of the work)

(viii) Staffing and communication requirements (i.e., the number of personnel, their technical specialty, and specific skills, the form and content of communications and other personnel interaction required when more than one person is involved)

(ix) The identification of any hazards involved in execution of the task

(d) The methods for identification of critical tasks. The identified critical tasks shall include, at the minimum, those operator actions which have significant impact on the PRA results, as presented in Section 19D.7, and the operator actions to isolate the reactor and inject water for the postulated event scenarios of a common mode failure of the Safety System Logic and Control System and/or the Essential Distributed Control and Instrumentation System concurrent with a design basis main steam line, feed water line or shutdown cooling line break LOCA.

(e) The methods for establishing information and control requirements.

(f) The methods for conducting alarm, display, processing, and control requirements analysis.

(g) The methods through which the application of task analysis results are assembled and documented to provide input to the development of personnel training programs.

(h) The methods to be used to evaluate the results of the task analysis.

(VI) HSI Design Implementation Plan

(1) (Satisfaction of the requirements presented herein shall result in the creation of an HSI Design Implementation Plan which is in full compliance with the Item 5.a Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The HSI Design Implementation Plan shall establish:

(a) The methods and criteria for HSI equipment design and evaluation of HSI human performance, equipment design and associated work place factors, such as illumination in the MCR and in the RSS area, which are consistent with accepted HFE practices and principles. Within the context of performing these HSI design evaluations, accepted HFE methods and criteria are presented in the following documents:

(i) AD/A223 168, System Engineering Management Guide, (Dept. of Defense, Defense Systems Management College, Kockler, F., et al)

(ii) ANSI HFS-100, American National Standard for Human Factors Engineering of Visual Display Terminal Workstations, (Am. Nat'l. Standards Institute)

(iii) EPRI NP-3659, Human Factors Guide for Nuclear Power Plant Control Room Development, 1984, (Electric Power Research Institute)

(iv) EPRI NP-3701, Computer-Generated Display System Guidelines, 1984, (Electric Power Research Institute)

(v) ESD-TR-86-278, Guidelines for Designing User Interface Software, (Department of Defense)

(vi) IEC 964, Design for Control Rooms of Nuclear Power Plants, 1989, (Bureau Central de la Commission Electro technique International)

(vii) MIL-H-46855B, Human Engineering Requirements for Military Systems, Equipment and Facilities, (Dept. of Defense)

(viii) MIL-HDBK-759A, Human Factors Engineering Design for Army, Material (Dept. of Defense)

(ix) DOD-HDBK-761A, Human Engineering Guidelines for Management Information Systems, (Dept. of Defense)

(x) MIL-STD-1472D, Human Engineering Design Criteria for Military Systems, Equipment and Facilities, (Dept. of Defense)

(xi) NUREG-0696, Functional Criteria for Emergency Response Facilities, 1980, (US Nuclear Regulatory Commission)

(xii) NUREG-0700, Guidelines for Control Room Design Reviews, 2002, (US Nuclear Regulatory Commission)

(xiii) NUREG-0800, Standard Review Plan, 2004, (US Nuclear Regulatory Commission)

(xiv) NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures, 1982, (US Nuclear Regulatory Commission)

(xv) NUREG/CR-5228, Techniques for Preparing Flowchart Format Emergency Operating Procedures (Vols. 1 & 2), 1989, (US Nuclear Regulatory Commission)

(xvi) NUREG/CR-4227, Human Engineering Guidelines for the Evaluation and Assessment of Video Display Units, 1985, (US Nuclear Regulatory Commission)

(xvii) Gilmore, et. al. (1989), User-Computer interface in process control: A human factors engineering handbook. San Diego, CA: Academic Press, Inc.

Note that within the set of documents listed above, differences may exist regarding the specific methods and criteria applicable to the conduct of HSI design evaluations. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those documents, which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for defining how that particular issue is addressed in the HSI design evaluations.

(b) That the HSI design shall implement the information and control requirements developed through the task analyses, including the displays, controls and alarms necessary for the execution of those tasks identified in the task analyses as being critical tasks (see paragraph V.2.d of this table).

(c) The methods for comparing the consistency of the HSI human performance equipment, design and associated workplace factors with that modeled and evaluated in the completed task analysis.

(d) That the HSI design shall not incorporate equipment (i.e., hardware or software function) which has not been specifically evaluated in the task analysis.

(e) The HSI design criteria and guidance for control room operations during periods of maintenance, test and inspection of control room HSI equipment and of other plant equipment that has control room personnel interface.

(f) The test and evaluation methods for resolving HFE/HSI design issues. These test and evaluation methods shall include the criteria to be used in selecting HFE/HSI design and evaluation tools which:

(i) May incorporate the use of static mockups and models for evaluating access and workspace-related HFE issues

(ii) Shall require dynamic simulations and HSI prototypes for conducting evaluations of the human performance associated with the activities in the critical tasks identified in the task analysis

(2) The Human System Interface Design Implementation Plan shall include:

(a) Identification of the specific HFE standards and guidelines documents that substantiate that the selected HSI Design Evaluation Methods and Criteria are based upon accepted HFE practices and principles.

(b) Definition of standardized HFE design conventions.

(c) Definition that the standard design features (Section 18.4.2), the standard HSI equipment technologies (Section 18.4.3), and the displays, controls and alarms shall be incorporated as requirements on the HSI design.

(d) Definition of the design/evaluation tools (e.g., prototypes) which are to be used in the conduct of the HSI design analyses, the specific scope of evaluations for which those tools are to be applied and the rationale for the selection of those specific tools and their associated scope of application.

(VII) Human Factors Verification and Validation Implementation Plan

(1) (Satisfaction of the requirements presented herein shall result in the creation of a Human Factors Verification and Validation Implementation Plan which is in full compliance with the Item 7.a Acceptance Criteria presented in Table 3.1 of the Tier 1

Design Certification material for the ESBWR design). The Human Factors Verification and Validation (V&V) Implementation Plan shall establish:

(a) Human factors V&V methods and criteria that are consistent with accepted HFE practices and principles. Within the context of performing human factors V&V, accepted HFE methods and criteria are presented in the following documents:

(i) AD/A223 168, System Engineering Management Guide, (Dept. of Defense, Defense Systems Management College, Kockler, F., et al)

(ii) DOD-HDBK-763, Human Engineering Procedures Guide, Chapters 5-7 and Appendices A and B, (Dept. of Defense)

(iii) DOD 5000.2, Defense Acquisition Management Policies and Procedures, (Dept. of Defense)

(iv) EPRI NP-3701, Computer-Generated Display System Guidelines, 1984, (Electric Power Research Institute)

(v) IEC 964, Design for Control Rooms of Nuclear Power Plants, 1989, (Bureau Central de la Commission Electro technique International)

(vi) IEEE-845, IEEE Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Station Control Rooms and Other Peripheries, 1999, (IEEE)

(vii) MIL-H-46855B, Human Engineering Requirements for Military Systems, Equipment and Facilities, (Dept. of Defense)

(viii) DOD-HDBK-761A, Human Engineering Guidelines for Management Information Systems, (Dept. of Defense)

(ix) NUREG-0700, Guidelines for Control Room Design Reviews, 2002, (US Nuclear Regulatory Commission)

(x) NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures, 1982, (US Nuclear Regulatory Commission)

(xi) TOP 1-2-610, Test Operating Procedure Part 1, (Dept. of Defense)

(xii) NSAC-39, Verification and Validation for Safety Parameter Display Systems, (Electric Power Research Institute)

(xiii) NUREG/CR-4227, Human Engineering Guidelines for the Evaluation and Assessment of Video Display Units, 1985, (US Nuclear Regulatory Commission)

Note that within the set of documents listed above, differences may exist regarding the specific methods and criteria applicable to the conduct of human factors V&V. In situations that such differences exist, for a particular issue, all of the methods and criteria presented within those documents which address that particular issue are considered to be equally appropriate and valid and, therefore, any of those documents may be selected as the basis for defining how that particular issue is addressed in the human factors V&V.

(b) That the scope of the evaluations of the integrated HSI shall include:

(i) The Human-System Interface (including both the interface of the operator with the HSI equipment hardware and the interface of the operator with the HSI equipment's software-driven functions)

(ii) The plant normal and emergency operating procedures

(iii) HSI work environment

(c) That static and/or "part-task" mode evaluations of the HSI equipment shall be conducted to confirm that the controls, displays, and data processing functions identified in the task analyses are designed per accepted HFE guidelines and principles.

(d) The integration of HSI equipment with each other, with the operating personnel and with the plant normal and emergency operating procedures shall be evaluated through the conduct of dynamic task performance testing. The dynamic task performance testing and evaluations shall be performed over the full scope of the integrated HSI design using dynamic HSI prototypes (i.e., prototypical HSI equipment which is dynamically-driven using real time plant simulation computer models). In the event that the particular HSI design implementation under consideration is referenced to a previous HSI design for which dynamic task performance test and evaluation results are available, those existing results, along with the results of limited scope dynamic task performance tests which address the areas of difference between the two subject HSI designs, may be used to satisfy this requirement. The methods for defining the scope and application of the dynamic HSI prototype, past test results and other evaluation tools shall be documented in the implementation plan. The dynamic task performance tests and evaluations shall have as their objectives:

(i) Confirmation that the identified critical functions can be achieved using integrated HSI design.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(A) Control Room Design

(1) The large size of the control room and console and their configuration contributed to operator dissatisfaction.

(2) Traffic flows should not be impeded by placement of consoles.

(3) Adequate levels of illumination are necessary to ensure that visual effectiveness is sufficient for task performance. Emergency lighting should be available.

(4) Noise levels in the main control room should be maintained within acceptable industry levels.

(5) The climate control system in the control room should be capable of continuously maintaining temperature and humidity within the human comfort zone.

(6) Convenient storage should be provided so that procedures, logs, and drawings needed for routine job performance are conveniently available. Storage should also be provided for equipment needed for emergency operation.

(B) Control Board Design

(1) Control boards should be optimized for minimum manning.

(2) Panels in the control rooms were observed to have large arrays of identical controls and displays and repetitive labels. The systems, subsystems, and components should be separated by appropriate demarcation methods.

(3) Controls and related displays should be located in close proximity so that the two items are readily associated and can be used conveniently with one another. Controls should be placed in an obvious and consistent order. The displays and controls used to monitor major system functions should be assigned to and arranged in functional groups.

(4) Flow arrangements between Video display unit formats and controls on panels should not differ.

(5) Flow mimics should be used to aid (and not mislead) the operators.

(6) Panel arrangements for similar systems should be the same.

(7) Location of controls in areas and orientations that render them vulnerable to accidental contact and disturbance should be avoided.

(8) Unclear, illogical, overly complex, or mirror-imaged control board or panel layout arrangements have been observed to promote operational mishaps and should be avoided.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(C) Computer

(1) Computer data should be available on display devices and hard copy output.

(2) Computer audible alarms should not be distracting.

(D) Display Devices

(1) The nomenclature, labeling, and arrangement of systems on the display devices should be similar to the panels.

(2) Display devices should be comprehensible with a minimum of visual search. When data is presented in lines and columns, the lines of data should be separated by a space (blank line), one character high, every 4-5 lines.

(3) Display access should be efficient and require a minimum of key strokes.

(4) Display devices should have convenient brightness, focus, and degauss controls.

(5) The character height should be the appropriate height for the viewing distance during normal and emergency conditions.

(6) Visibility of display devices should not be affected by glare.

(E) Anthropometrics

(1) Panel dimensions should accommodate the 5 to 95 percentile range of the user population to ensure that personnel can see and reach the displays and controls or the front and back panels. Displays should not be placed beyond the visual range of the operators.

(2) Controls should not be located in the control panels that require the operator to lean into the panel. This is a potential health risk to the operator and to the equipment.

(F) Controls

(1) Large controls were observed to have been used in place of preferred smaller controls. Larger controls impact panel size and should be avoided.

(2) Labeling or coding techniques should be used to differentiate controls and indicator lights of similar appearance.

(3) Control configurations should not introduce parallax problems.

(4) Control switches that must be held by the operator for operation should be avoided unless necessary.

(5) Projecting control handles should not cover or obstruct labels.

(6) Key lock switches require administrative control and should be avoided if possible.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(7) Control handles should not be difficult to operate and should not cause the operators to resort to using unauthorized mechanical leveraging devices (i.e., “cheaters”) so as to achieve reduced difficulty in operation.

(8) Controls should be built and installed following standard conventions for OPEN/CLOSE and INCREASE/DECREASE. Setpoint scales should not move up in response to a downward movement of the controller thumbwheel.

(9) Inadvertent operation of adjacent controls may be reduced through the use of shape coding such as using similar shaped handles for similar functions (i.e., pistol grips for pumps and round handles for valves).

(G) Indicator Lights

(1) Instances of improper use of qualitative indicators were observed where quantitative displays such as meters would be more effective.

(2) Light status (on/off) should be visible to the operator. Extinguished bulbs should be obvious and a test method provided. Lamp designs should allow for easy access for lamp removal.

(3) The use of so-called negative indications (the absence of an indication) should not be used to convey information to the operator.

(4) Indicator design selection and layout should be standardized to conserve panel space.

(5) A color code standard should be established for indicating lights.

(H) Display and Information Processing

(1) Plant parameter validity should not have to be inferred. In addition to secondary information, the quality or validity of the displayed parameter should be available to allow operators to readily identify improper ESF or other safety equipment status under various operating modes.

(2) Necessary information should be available during events such as SBO and LOOP. Systems and indications such as Neutron Monitoring System, control rod position indication, and drywell area radiation indication should all be available during these events.

(3) The main control room should contain an integrating overview display. The overview display should provide a limited number of key operating parameters.

(4) The same displays that are used during normal operation should be used by the operators during accident conditions to ensure their familiarity with the interface.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(I) Meters

(1) Proper use of minor, intermediate, and major scale markings in association with scale numerals should be made. Formats should be customized to take into account identification of normal operating values and limits. Scale numerical progressions and formats should be selected for the process parameter being presented.

(2) Placement of meters above and below eye level, making the upper and lower segment of the scale difficult to read (especially with curved scales), can present parallax problems.

(3) Meters were observed that fail with the pointer reading in the normal operating band of the scale. The instrument design should allow the operator to determine a valid indication from a failed indication.

(4) Placement of meters on panels should prevent glare and reflections caused by overhead illumination.

(5) Where redundant channels of instrumentation exist, software-based displays should provide for easy inspection of the source data and intermediate results without the need to display them continuously.

(6) Data presented to the operator should be in a usable form and not require the operator to calculate its value. Scale graduations should be consistent and easily readable. Zone markings should be provided to aid in data interpretation.

(7) Meter pointers should not obscure the scale on meters.

(8) Process units between the control room instruments and the operating procedures should be consistent.

(J) Chart Recorders

(1) Recorders should not be used in place of meters. Recorders should be selected with consideration given to minimizing required maintenance and high reliability.

(2) A recorder designed to monitor 24 parameters was observed to have 42 parameters assigned to it. This makes it extremely difficult to read the numerical outputs on the chart paper. The inputs assigned should be consistent with the design of the recorder.

(3) Operational limits should be defined on recorders. Proper selection of recorder scales will eliminate the need for overlays. The units for the process should be labeled on the recorder.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(4) Monitored inputs should be assigned to recorder pens in alphabetical order. The correlation of pen color to input parameter should be clearly defined by multi-pen recorder labels.

(5) The change of chart speed should also be noted on the chart paper when the paper is changed. The paper scales should match the fixed scales.

(6) Recorders should have fast speed and point select capability.

(7) Proper placement of recorders and adequate illumination should prevent glare and parallax problems with recorder faces.

(8) The pointers should not cover the graduation marks.

(9) When upper and lower pens coincide, the printout of the upper scale should still be visible.

(K) Annunciator Warning Systems

(1) Annunciators should be located near the control board panel elements to which they are related. Divisional arrangements should be consistent. Annunciators should be functionally located near the applicable system.

(2) "Advisory alarms" reporting expected conditions should not be grouped with true alarms. The audio and visual warning system signal should be prioritized to reduce the audio and visual burden placed on the operators during an event.

(3) Some alarms were observed to lack specificity. Multi-input alarms (e.g. xyz pressure/levels, hi/lo) frustrate, rather than inform the operator.

(4) Excessive alarms were observed during emergency conditions. Auditory signals should be coded to aid the operator in determining the panel location.

(5) Alarm operating sequence controls should be placed at specific locations to encourage operator acknowledgment.

(6) For standing and sit-down workstations, window size and lettering height should be consistent with the viewing distance.

(7) The labels should use consistent abbreviations and nomenclature and not be ambiguous.

(8) For traceability to response procedures, the windows should be identified with a location reference code.

(9) A consistent color coding convention should be employed.

(10) A "First Out" feature should be provided that presents prioritized parameters important to safety parameters for immediate operator response.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(11) Means should be provided for identification of out-of-service annunciators.

(12) Annunciators for conditions that signal an EOP entry condition should be located based on the functional analysis.

(L) Coding of Displays and Controls

(1) The color codes for the control boards should be systematically applied. Effective color coding should be used to aid in differentiating between identical controls placed in close proximity.

(2) The coding of indicators should inform the operator whether a valve is open or closed.

(3) Systematic approach to color and shape coding of controls should be taken.

(M) Labeling

(1) Label abbreviations, numbering, and nomenclature should be consistent. A label placement standard for the control room should be established. Labels should be placed consistently above or below the panel elements being identified and not placed between two components.

(2) Hierarchical labeling schemes, including size coding or differentiation of labels, should be used to identify major console panels, sub-panels, and panel elements. Hierarchical labeling will eliminate the need to place redundant labels on control or display devices.

(3) The content of the labels should be consistent with the procedures used by the operators.

(4) The labels should meet the readability guidelines and should not be obscured by the equipment that they are mounted near. A control room standard for labels should be established that addresses label character size and font.

(5) Maintenance tags should not obscure labels or panel components such as displays.

(6) To minimize the mispositioning of valves and other equipment, the controls and displays should be labeled with the unique number or name of the valve or piece of equipment.

(N) Communications

(1) Communications in the control room should consider the ambient noise levels in the control room and plant. The control room operator should be able to communicate with necessary personnel in the plant. Communication equipment should also be provided at the remote shutdown panel.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(2) Communications equipment design should not limit the operator's access to the controls or displays.

(3) The communication system should be accessible from the operator's workstations.

(O) Task Analysis

(1) Controls and displays should be located for effective operator response to postulated events. Information needed by the operator in the control room should be readily available and not located at remote panels in the plant.

(2) In addition to normal and emergency conditions, plant displays and controls should also consider low power and shutdown scenario information requirements.

(P) Procedures

(1) The measurement units in the procedure and the values indicated on display scales should be consistent.

(2) Control board designs should make provisions for the operator's simultaneous referral to the procedures and the operation of the control boards.

(3) The parameters displayed on electronic information systems or on the control boards should be designed to support the EOPs as well as other required monitoring tasks.

(4) The safety function parameter status should be presented in an organized, readily accessible format compatible with the EOPs.

(5) A procedure should address operator action in the event of computer, display device, or printer problems or complete failure.

(Q) Operator Errors

(1) Operator mishaps were observed to be caused by the absence of a timely, attention-getting indication (either qualitative or quantitative) that informs the operator that some element of the system is not operating properly.

(2) Operator mishaps were also observed to result from incorrect lineup of valves.

(R) Maintenance and Testing

(1) The main control room should be designed in such a way that minimizes the need for maintenance and test personnel to work, or at least limit their presence, in the control room.

(2) Control room displays should be designed and installed for easy calibration and replacement.

Table 18E-2

Results of Operating Experience Review of Previous Nuclear Power Plant HSI Designs

(3) Access for inspection, operation, and routine maintenance of components should not be restrictive.

Table 18E-3

HFE Analysis

(I) System Functional Requirements Analyses

(1) (Satisfaction of the requirements presented herein shall result in the conduct of system functional requirements analyses which are in full compliance with the Item 2.b Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The system functional requirements analyses shall be conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the System Functional Requirements Analysis Implementation Plan.

(2) The results of the system functional requirements analyses shall be documented in a report that includes the following:

(a) Objectives of the system functional requirements analyses

(b) Description of the methods employed in the conduct of system functional requirements analyses

(c) Identification of deviations from the System Functional Requirements Analysis Implementation Plan

(d) Presentation and discussion of the results of the system functional requirements analysis, including a discussion of design change recommendations derived from these analyses and/or negative implications that the current design may have on safe plant operations

(e) Conclusions regarding the conduct of the analyses and the analyses results

(3) The results of the HFE Design Team's evaluation of the conduct and results of the system functional requirements analyses shall be documented in a report that includes the following:

(a) The methods and procedures used by the HFE Design Team in their review of the system functional requirements analyses

(b) The HFE Design Team's evaluation of the completed system functional requirements analyses, including an evaluation of the compliance with the System Functional Requirements Analysis Implementation Plan and the HFE Program Plan

(c) Presentation and discussion of the HFE Design Team's Review findings

Table 18E-3

HFE Analysis

(II) Function Allocation Analyses

(1) (Satisfaction of the requirements presented herein shall result in the conduct of function allocation analyses which are in full compliance with the Item 3.b Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The function allocation analyses shall be conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Allocation of Functions Implementation Plan.

(2) The results of the function allocation analysis shall be documented in a report that includes the following:

(a) Objectives of the function allocation analyses

(b) Description of the methods employed in the conduct of the function allocation analyses

(c) Identification of deviations from the Allocation of Function Implementation Plan

(d) Presentation and discussion of the results of the function allocation analyses, including a discussion of design change recommendations derived from these analyses and/or negative implications that the current design may have on safe plant operations

(e) Conclusions regarding the conduct of the analyses and analysis results

(3) The results of the HFE Design Team's evaluation of the conduct and results of the function allocation analyses shall be documented in a report that includes the following:

(a) The methods and procedures used by the HFE Design Team in their review of the function allocation analyses

(b) The HFE Design Team's evaluation of the completed function allocation analyses, including an evaluation of the compliance with the Allocation of Function Implementation Plan and the HFE Program Plan

(c) Presentation and discussion of the HFE Design Team's review findings

Table 18E-3

HFE Analysis

(III) Task Analyses

(1) (Satisfaction of the requirements presented herein shall result in the conduct of task analyses which are in full compliance with the Item 4.b Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR Design). The task analyses shall be conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Task Analysis Implementation Plan.

(2) The results of the task analyses shall be documented in a report that includes the following:

(a) Objectives of the task analyses

(b) Description of the methods employed in the conduct of the task analyses

(c) Identification of deviations from the Task Analyses Implementation Plan

(d) Presentation and discussion of the results of the task analyses, including discussion of design change recommendations derived from these analyses and/or negative implications that the current design may have on safe plant operations

(e) Conclusions regarding the conduct of the analyses and the analyses results

(3) The results of the HFE Design Team's evaluation of the conduct and results of the task analyses shall be documented in a report that includes the following:

(a) The methods and procedures used by the HFE Design Team in their review of the completed task analyses

(b) The HFE Design Team's evaluation of the completed task analyses including an evaluation of the compliance with the Task Analysis Implementation Plan and the HFE Program Plan

(c) Presentation and discussion of the HFE Design Team's review findings

Table 18E-4

Human System Interface Design

(I) HSI Design Analyses

(1) (Satisfaction of the requirements presented herein shall result in the conduct of HSI design analyses which are in full compliance with the Item 5.b Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The Human System Interface (HSI) design implementation and analyses shall be conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the HSI Design Implementation Plan.

(2) The results of the HSI design analyses shall be documented in a report that includes the following:

(a) Objectives of the HSI design analyses

(b) Description of the methods employed in the conduct of the HSI design analyses

(c) Identification of deviations from the HSI Design Implementation Plan

(d) Presentation and discussion of the results of the HSI design analyses, including discussion of design change recommendations derived from these analyses and/or negative implications that the current design may have on safe plant operations

(e) Conclusions regarding the conduct of the analyses and the analysis results

(3) The results of the HFE Design Team's evaluation of the conduct and results of the HSI design analyses shall be documented in a report that includes the following:

(a) The methods and procedures used by the HFE Design Team in their review of the HSI design analyses

(b) The HFE Design Team's evaluation of the completed HSI design analyses, including an evaluation of the compliance with the HSI Design Implementation Plan and HFE Program Plan

(c) Presentation and discussion of the HFE Design Team's review findings

Table 18E-5

Human Factors Verification and Validation

(I) Human Factors Verification and Validation

(1) (Satisfaction of the requirements presented herein shall result in the conduct of human factors verification and validation activities which are in full compliance with the Item6.b Acceptance Criteria presented in Table 3.1 of the Tier 1 Design Certification material for the GE ESBWR design). The human factors verification and validation (V&V) of the human system interface (HSI) design shall be conducted in accordance with the requirements of the Human Factors Engineering Program Plan and the Human Factors V&V Implementation Plan.

(2) The results of the human factor verification and validation (V&V) activities shall be documented in a report that includes the following:

(a) Objectives of the human factors V&V

(b) Description of the methods employed in the conduct of the human factors V&V

(c) Identification of deviations from the Human Factors V&V Implementation Plan

(d) Presentation and discussion of the human factors V&V results, including discussion of design change recommendations derived from the human factors V&V tests and evaluations and/or significant negative implications that the current HSI design may have on safe plant operations which may have been identified

(e) Conclusions regarding the conduct of the human factors V&V and the results

(3) The results of the HFE Design Team's evaluation of the conduct and results of the human factor verification and validation (V&V) shall be documented in a report that includes the following:

(a) The review methodology and procedures used by the HFE Design Team in their review of the human factor V&V

(b) The HFE Design Team's evaluation of the completed human factors V&V, including an evaluation of the compliance with the Human Factors V&V Implementation Plan and HFE Program Plan

(c) The HFE Design Team's evaluation of the completed human factors V&V, including an evaluation of the presentation and discussion of the HFE Design Team's Human Factors review findings

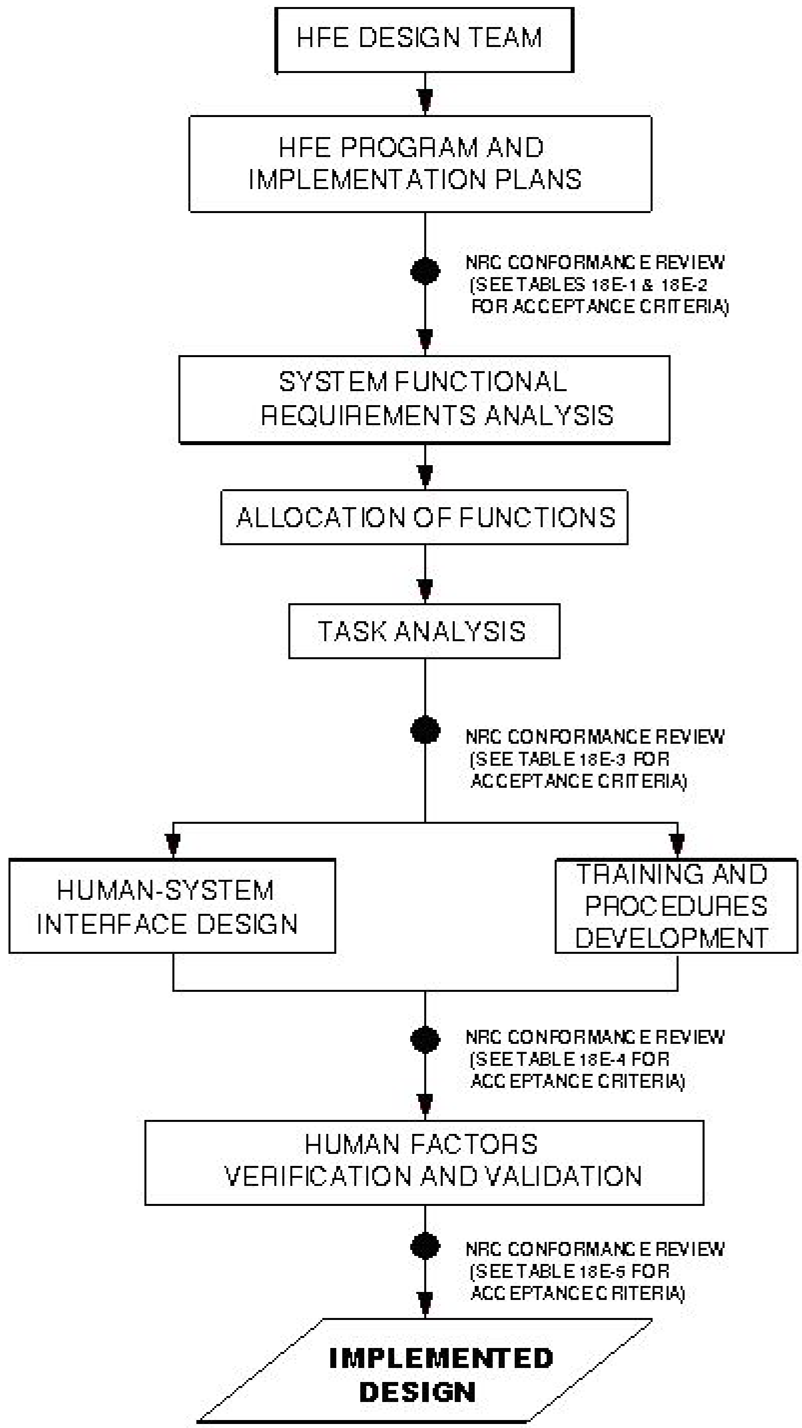

Figure 18E-1. ESBWR Human-System Interface Design Implementation Process

18F. EMERGENCY OPERATION INFORMATION AND CONTROLS

This appendix is COL applicant scope. It will contain the results of an analysis of information and control needs of the main control room operators. The analysis will be developed as a COL item. Please refer to Section 18.8. The analysis will be based upon the operation strategies given in the ESBWR Emergency Procedure Guidelines (EPGs) as presented in Appendix 18A and the significant operator actions determined by the Probabilistic Risk Assessment (PRA) described in Chapter 19. The minimum inventory of controls, displays and alarms from this analysis will be presented herein. The information and controls identified from this analysis do not necessarily include those from other design requirements (such as those from Subsection 18.4.2.11, SPDS). The results of this evaluation will be placed in the HFE issue Tracking System. Supporting information will be provided in Appendix 18H.

18G. DESIGN DEVELOPMENT AND VALIDATION TESTING

18G.1.1 Introduction

As part of the ABWR design development, a five-year program was undertaken for the purpose of "studying the application of man-machine technologies to enhance the efficiency of operation control of nuclear power plants". During the course of this program, a variety of tests, studies and evaluations were performed in a number of areas of control room equipment design. These studies and evaluations culminated in the fabrication and testing of prototype control room human-system interface (HSI) equipment designs at two separate facilities. The results of this development program form the basis for the ESBWR control room HSI design.

The purpose of this report is to provide summary descriptions of the studies, evaluations and validation testing performed during the joint study program. The studies, evaluations and testing, other than the prototype validation testing, are described in Subsection 18G.2 and the validation testing of the control room equipment prototypes is discussed in Subsection 18G.3.

18G.1.2 Design Development

18G.1.3 General

The program research described in this section is discussed under the following subtitles:

(1) Standard control room design features

(2) Allocation of functions

(3) Operator work load

(4) Other Areas of Interest

18G.1.4 Standard Control Room Design Features

There are eighteen standard design features for the ESBWR control room HSI which are listed as follows:

(1) A single, integrated control console staffed by two operators; the console has a low profile such that the operators can see over the console from a seated position.

(2) The use of plant process computer system-driven on-screen control video display units (VDUs) for safety system monitoring and non-safety system control and monitoring (E-DCIS and NE-DCIS).

(3) The use of a separate set of on-screen control VDUs for safety system control and monitoring and separate on-screen control VDUs for non-safety system control and monitoring; the operation of these two sets of VDUs is entirely independent of the process computer system. Further, the first set of VDUs and all equipment associated with their functions of safety system control and monitoring are divisionally separate and qualified to Class l-E standards.

(4) The use of dedicated function switches on the control console.

(5) Operator selectable automation of pre-defined plant operation sequences.

(6) The incorporation of an operator selectable semi-automated mode of plant operations, which provides procedural guidance on the control console VDUs.

(7) The capability to conduct these pre-defined plant operation sequences in manual mode.

(8) The incorporation of a large display panel which presents information for use by the entire control room operating staff.

(9) The inclusion on the large display panel of fixed-position displays of key plant parameters and major equipment status.

(10) The inclusion in the fixed-position displays of both 1E-qualified and non-1E qualified display elements.

(11) The independence of the fixed-position displays from the plant process computer.

(12) The inclusion within the large display panel of a large video display unit which is driven by the plant process computer system.

(13) The incorporation of a “monitoring only” supervisor’s console which includes VDUs on which display formats available to the operators on the main control console are also available to the supervisor.

(14) The incorporation of the safety parameter display system (SPDS) function as part of the plant status summary information, which is continuously, displayed on the fixed-position displays a portion of the large display panel.

(15) The use of fixed-position alarm tiles on the large display panel.

(16) The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms.

(17) A spatial arrangement between the large display panel, the main control console and the shift supervisors’ console which allows the control room operating crew to view the information presented on the large display panel from the seated position at their respective consoles.

(18) The use of VDUs to provide alarm information in addition to the alarm information provided via the fixed-position alarm tiles on the large display panel.

Specific studies and evaluations done in support of these standard design features, prior to the validation testing of the two prototype designs, are discussed in the following paragraphs of this subsection.

18G.1.4.1 Control Console

[v62]The results of operating crew task analyses were used to aid in control panel layout studies and support evaluations of the operating crew size. Studies of the feasibility of main control panel size reduction and consolidation of two or more panel functions into one panel were completed. Other studies were made of alternate arrangements of control and monitoring equipment on the console.

18G.1.4.2 Video Display Units (VDUs)

Studies of trends in VDU uses in control rooms were made. Because VDUs have been used previously, not only in control rooms of other industrial processing plants but in nuclear plants as well, the issue of concern was not whether to use these devices but how to use them most efficiently. Studies of VDU control devices were completed and development testing of on-screen control devices was done.

[v63]Means of enhancing the reliability of certain types of VDUs (e.g., CRTs) were also studied. An evaluation was made of the merits of different types of computer input devices (i.e., light pens, trackballs, mice and touch-screens). Methods of optimization of VDU screen systems for aiding operator responses (a) during normal plant operation, (b) after discovery of an anomaly, (c) after a scram and (d) in the event of an accident were examined. Screen selection methods and methods of integrating the use of the console VDUs with that of the large display panel were also reviewed. As part of this study, operator decision-making processes during each of the above-mentioned plant conditions were analyzed.

18G.1.4.3 Plant Operations

Automation of normal plant operations (e.g., startup, shutdown, power maneuvers) was evaluated for potential ABWR application to enhance operability and minimize the burden on the operating staff. Automation in ESBWR is more extensive based on SBWR experience. The extent of automated operations was carefully selected to ensure that the primary control of plant operations remains in the hands of the operators. The general approach, which was followed in selecting the operations to be automated, consisted of the following steps:

(1) Task analyses were performed which defined the type and sequence of tasks that are required for accomplishing normal plant operations. These analyses were done assuming completely manual operations (i.e., no automation).

(2) From these task analyses, assessments were made of the operator workload, the complexity of the operation, the degree of repetitiveness and tedium in the operation and the feasibility of automating the operation.

(3) From these evaluations, it was concluded that, for a given plant system, many tasks are conducted for the purpose of changing the operational status of that system. For many of these normal system operations, sequence master control functions were defined. This approach was applied to both safety-related and nonsafety-related system operations. Dedicated sequence master control switches were incorporated into the main control console design for initiating these automation sequences. The sequence master control switches are located in the hard switch panels on the horizontal desk area of the main control panel.

(4) Operational changes in safety systems that require only minimal operator action (e.g., changing the position of the reactor mode switch) were not automated. Safety system mode changes are performed on the ESBWR in a manner similar to conventional BWR operations.

(5) For tedious or repetitive operations, special plant level automation functions were incorporated into the control room design. Examples of such automated functions include control rod operations during startup (e.g., rod selection and movement of control rods to

maintain the vessel heat up rate), changes in reactor power controller set points to accomplish daily load following maneuvers and changes in the reactor pressure set point during normal heat up and cool down of the reactor.

In accordance with the above studies, the semi-automatic mode was provided to give the operator(s) automatic guidance for accomplishing the desired normal changes in plant status. In the semi-automatic mode of operation, the plant computer provides no control actions. The operator must activate all necessary system and equipment controls for the semi-automatic sequence to proceed.

A manual mode of operation was also retained in the design. The manual operating mode is equivalent to the manual operation of conventional BWR designs and is available at the operator's discretion or when automatic operation is terminated due to abnormal plant conditions.

18G.1.4.4 Large Display Panel

Studies of trends in control room information presentation methods (e.g., the Halden IPSO project as described in Report No. HWR-184) were completed. An analysis of questionnaire survey responses from the manufacturing sector of Japanese industry, including fossil fuels and other process plants, indicated that large displays or combinations of large displays and console-mounted VDUs are regarded as an effective means of providing information to operators in a broad spectrum of industrial plant environments.

18G.1.4.5 Independence of Fixed-Position Displays

This feature was adopted as a result of evaluations of plant operations during periods of postulated equipment failures and studies of trends in fixed-position display designs throughout industry. The incorporation of the fixed-position displays provides redundancy in display modes and contributes to the ability to safely shut down the plant if the process computer system is lost.

18G.1.4.6 Large Video Display

Studies of large screen utilization trends and of how large screens can best be integrated with VDUs in the control room were performed. The large screen study topics are summarized in Table 18G-1.

Traditionally, during planned outages and at other times, large blackboards have been used, on a temporary basis, in control rooms to display information regarding the status of important ongoing processes. As summarized in Table 18G-1, large screens can be utilized as substitutes for these blackboards. Also, large screens can be used as industrial television monitors (ITV) to make local checks during normal plant operation. In addition, large screens can be used to display CRT screen formats to the entire control room crew simultaneously.

The technology for large screens was also reviewed. Video projectors and displays using liquid crystal projection, luminous source, liquid crystal transmission, LED and CRT technology were compared from the standpoint of screen size, optimum viewing distance, resolution, brightness and update speed.

18G.1.4.7 Alarms

Studies of alarm system technologies, the uses of alarms in control rooms and alarm prioritization and suppression methods to reduce the information load on the operating crew in times of upsets were carried out. Improvements in methods of distributing alarm functions between fixed-position indicators and VDUs were examined. Evaluations of VDU alarm presentation methods and formats were also done.

18G.1.4.8 Control Room Spatial Arrangement

Control room functions and arrangements were studied. Evaluations were made of the free space requirements of the control room. Regulations and legislation affecting areas of control room design such as comfort, human factors engineering and control room habitability considerations were reviewed.

18G.1.5 Allocation of Functions

Studies of trends in automation of operator functions in nuclear power plants were done. In the early days of the industry, improvements in the level of automation were related chiefly to maintenance of safety. Later, attention has come to be focused on the goal of reducing the burden on the operators. Accordingly, the task analyses discussed in Subsection 18G.2.2.3 "Plant Operations", were performed and allocation of functions were made as described in that subsection.

18G.1.6 Operator Work Load

Studies of operator workload were performed as part of the automation studies. The task analyses performed on both system and plant operations were used to develop time histories of operator workload for both normal and abnormal operations. As discussed previously, the operator workload information was an important part of the basis for decisions regarding automation.

18G.1.7 Other Areas of Interest

Other study areas, not directly contributing to the development of the standard design features discussed in Subsection 18G.2.2, were also pursued. These other study areas included the following:

(1) Configuration of the HSI system:

- (a) Relation of process computer to other components.
- (b) Data highway (Network) design.

(2) Functions and configurations of plant data management systems:

- (a) Core management.
- (b) Operation management.
- (c) Maintenance management.
- (d) Security controls.

(e) Management of documentary information.

(f) Management of site operation records.

(3) Operator assistance technologies:

(a) Diagnosis.

(b) Planning.

(c) Routine maintenance assistance.

(4) Audio response systems:

(a) Speech recognition.

(b) Speech synthesis.

18G.2 VALIDATION TESTING

18G.2.1 General

During the summer of 1990, a systematic program of testing on two separate prototypes was performed to validate the key features of the ABWR main control room equipment design. Three teams of three licensed plant operators each were used for the tests. The prototypes were fabricated according to a conceptual design prepared during the development program discussed in Section 18G.2.

18G.2.1.1 General Test Description

Test Teams:

The test teams were composed of a utility's operations personnel from operating boiling water reactors. Each team consisted of three individuals; one senior operator, one operator and one shift supervisor. The test sequences were conducted with the two operators at the main control console and the supervisor located at a desk about 4.5m behind the operators.

Data:

Test data were collected in the form of (a) performance measures: these consisted of observations of operator actions and behavior during the tests and examination of video tapes taken during the tests, and (b) opinions, which took the form of post-test operator interviews and questionnaires filled out by the operators during the post-test debriefing sessions. The objective of the data collection methods was to obtain as complete a record as possible of the operator's interactions with the HSI model and each other as they reacted to the simulated plant conditions and of the operator's reactions to the equipment design.

Equipment Configuration:

The control room prototypes included the following key design features, which are correlated to the ESBWR standard design features listed in Subsection 18G.2.2:

(1) A single, integrated control console staffed by two operators; the console has a low profile such that the operators can see over the console from a seated position [Standard Design Feature (1)].

(2) The use of on-screen controls video display units (VDUs) for safety system monitoring and nonsafety-related system control and monitoring [Standard Design Feature (2)].

(3) The use of a separate set of on-screen control VDUs for safety system control and monitoring [Standard Design Feature (3)].

(4) The use of dedicated function hardware switches on the control console [Standard Design Feature (4)].

(5) Operator-selectable automation of predefined plant operation sequences [Standard Design Feature (5)].

(6) The incorporation of an operator-selectable semi-automated mode of plant operations, which provides procedural guidance on the control console VDUs [Standard Design Feature (6)].

(7) The capability to conduct these pre-defined plant operation sequences in an operator manual mode [Standard Design Feature (7)].

(8) The incorporation of a Large Display Panel, located behind the main control console, which presents information for use by the entire control room operating staff [Standard Design Feature (8)].

(9) The inclusion on the Large Display Panel of key plant parameters and major equipment status displayed at fixed positions on the panel [Standard Design Feature (9)].

(10) The inclusion on the Large Display Panel of critical plant parameters which are continuously displayed at fixed positions on the panel [Standard Design Features (10) and 14)].

(11) The independence of the fixed-position displays from the process computer [Standard Design Feature (11)].

(12) The inclusion on the Large Display Panel of a large video display unit to supplement the information presented on the fixed displays [Standard Design Feature (12)].

(13) The use of fixed-position alarm tiles on the Large Display Panel [(Standard Design Feature (15)].

(14) The application of alarm processing logic to prioritize alarm indications and to filter unnecessary alarms [Standard Design Feature (16)].

(15) Spatial arrangement between the large display panels, the main control console and the supervisors' console that allow the entire control room crew to monitor the information presented on the large display panel [Standard Design Feature (17)].

(16) The use of VDUs to provide low-level alarm indications and detailed information on the alarms, which are presented on fixed-position alarm, tiles [Standard Design Feature (18)].

Standard Design Feature (13) was not specifically incorporated in the prototype-testing program. The essential feature of the supervisors' console, which was modeled, was its lack of control capability.

Test Items:

The testing consisted of exercising a variety of plant operation scenarios, which were generated by a dynamic simulator. The operating crew would respond to the changing plant conditions during the scenarios by monitoring the information, which was available to them and manipulating the appropriate controls. The scenarios included in the test program are summarized in Table 18G-2. Video tapes were taken of the operators as the scenarios were being run.

18G.2.1.2 Test Results

The operator interviews and questionnaire data collected during debriefing sessions at both test facilities, plus the video tape evaluations, fully supported the key design features summarized previously in Section 18G.3.1.1. Specific, relevant results regarding those key features are summarized below.

(1) Under normal, abnormal and accident conditions it is easy to comprehend the plant condition from the information presented on the Large Display Panel (LDP).

(2) The LDP contributes much to improve overall monitor ability of the plant.

(3) The fixed position displays and large alarm tiles on the LDP are very effective in promoting understanding of the conditions during accidents.

(4) The large variable display on the LDP is watched more often during normal operation than during abnormal operation. However, in general, the large variable display is not watched as often as the fixed position displays.

(5) Suggested formats for display on the large variable screen include trends, plant summary and Safety Parameter Display System (SPDS) information during abnormal/ accident events and alarms.

(6) Work space and sitting posture at the Main Control Console are good, in general.

(7) Work space on the tested Main Control Console arrangement would be crowded if more than two operators were required at the console.

(8) The presentation of top-level, fixed position alarms on the LDP is very effective.

(9) The display of individual alarms on the VDUs is good.

(10) The size of the CRT screens was preferred over that of the flat panel devices. However, the touch screen operations on the flat panels were rated as better than on the CRTS.

The operator's comments focused on those design features that were new to them. Hence, the comments dealt with the LDP, alarm presentation scheme, main console configuration and operations with touch screens. There were little or no comments regarding the automated, semi-automated or manual operating modes or the hardware switches because of the particular operator's experience with such features.

The direct operator feedback and evaluation of the test sequence video tapes also provided much additional data which was relevant to the specific details of the equipment employed and the scope of the prototype control console dynamic simulation. Although these additional test results are not pertinent to the evaluation of the ESBWR key features described in Subsection 18G.2.2, they are being addressed and incorporated into the ongoing control room.

Table 18G-1

Large Screen Utilization Topics

		During Planned Outages	During Startup Shutdown	During Normal Operation	During Abnormalities/ Accidents
BLACK-BOARD	SUB-STITUTES	• Status of system isolation • Status of refueling • Status of incoming power	• Startup/ shutdown curves • Schedule tables • Load-following curves	• Status of equipment repair	• Weather / environment • Instrumentation during accidents
USED	ASITV	• Local Equipment status • MSIV room • Turbine operating floor, etc.	• Status of startup and shutdown auxiliaries • Drywell inspection status	• Radwaste control room, etc.	• Local equipment status
CON-CURRENTLY	ASVDU	• Plant summary • Ordinary alarms • Trend displays	• Plant summary • Ordinary alarms • Trend displays • Summary display of core status	• Plant summary • Ordinary alarms • Trend displays • Summary display of core status	• Emergency alarms • Plant summary • ECCS summary • Trend displays

Table 18G-2

Test Scenarios and Evaluations

Test Case Classification	Test Scenarios	Key Items of Evaluation
Normal Startup/Shut down and Surveillance	Reactor Critical Reactor Temperature Increase Power Adjustment to Support Drywell Inspection Generator in Parallel Condensate/ Feedwater System Alignment Power Increase Generator Trip/Turbine Trip	1) Transition to manual from automated operations. 2) Hardware switch operation capability. 3) Functions identified for implementation using hardware switches. 4) Adequacy and effectiveness of displayed information for monitoring. 5) Location of displayed information. 6) Adequacy and need for expansion of operator support function(s).
Equipment Trips	Reactor Feed Pump Trip Condensate Pump Trip	Items 2) through 5), plus 7) Large display panel effectiveness. 8) Alarm system effectiveness.
Scrams and Accidents	MSIV Closure Loss of Preferred Power Loss-of-Coolant Accident	Items 2) through 5) and 7) through 8), plus 9) Appropriateness and effectiveness of level of automation after a scram.

Computer Failure	Loss of Automated Operating Mode Loss of all Process Computer supported VDUs	Items 1) through 5), plus 10) Operability by sub-loop automation.

18H. SUPPORTING ANALYSIS FOR EMERGENCY OPERATION INFORMATION AND CONTROLS

This appendix is COL applicant scope. It will discuss the supporting analysis of information and control needs of the main control room operators. The discussion is based upon the operation strategies given in the ESBWR Emergency Procedure Guidelines (EPGs) as presented in Appendix 18A and the significant operator actions determined by the Probabilistic Risk Assessment (PRA) described in Chapter 19. The minimum inventory of controls, displays and alarms from this analysis will be presented herein. EPG considerations will be included to address issues associated with ATWS stability changes.