GE Nuclear Energy

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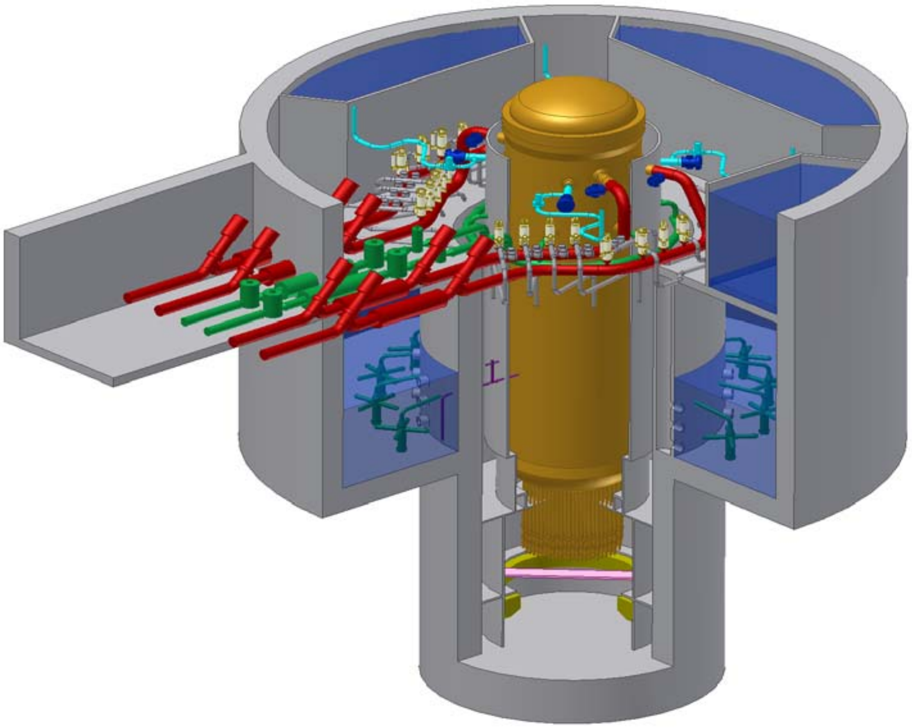

ESBWR Design Control Document
Tier 2
Chapter 17
Quality Assurance

Contents

List of Tables

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 Head 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

Global Abbreviations And Acronyms List

Term	Definition
ASTM	American Society of Testing Methods
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

Global Abbreviations And Acronyms List

Term	Definition
CPS	Condensate Purification System
CPU	Central Processing Unit
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
DS	Independent Spent Fuel Storage Installation
DTM	Digital Trip Module

Global Abbreviations And Acronyms List

Term	**Definition**
DW	Drywell
EB	Electrical Building
EBAS	Emergency Breathing Air System
EBHV	Electrical Building HVAC
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

Global Abbreviations And Acronyms List

Term	Definition
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
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
GENE	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

Global Abbreviations And Acronyms List

Term	**Definition**
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
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

Global Abbreviations And Acronyms List

Term	**Definition**
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
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

Global Abbreviations And Acronyms List

Term	Definition
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
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
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

Global Abbreviations And Acronyms List

Term	**Definition**
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
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
QAPD	Quality Assurance Program Document
RACS	Rod Action Control Subsystem
RAM	Reliability, Availability and Maintainability
RAPI	Rod Action and Position Information
RAT	Reserve Auxiliary Transformer
RB	Reactor Building

Global Abbreviations And Acronyms List

Term	Definition
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
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
RTNSS	Regulatory Treatment of Non-Safety Systems
RTP	Reactor Thermal Power
RW	Radwaste Building
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling
RWE	Rod Withdrawal Error

Global Abbreviations And Acronyms List

Term	**Definition**
RWM	Rod Worth Minimizer
SA	Severe Accident
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
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
SR	Surveillance Requirement
SRM	Source Range Monitor
SRNM	Startup Range Neutron Monitor

Global Abbreviations And Acronyms List

Term	**Definition**
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
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

Global Abbreviations And Acronyms List

Term	Definition
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.
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

17. QUALITY ASSURANCE

17.0 INTRODUCTION

Section 17.1 describes the Quality Assurance (QA) Program used by GE Nuclear Energy (GENE) for the ESBWR. The program is based on the standard GENE QA Program, documented in topical report NEDO-11209-04A (Reference 17.0-1) and the additional information in this chapter, which describes and clarifies GE interfaces and responsibilities with its ESBWR project participants. The ESBWR project participants are domestic and international organizations that have extensive independent experience in the design, development, construction and operation of nuclear power plants.

The standard GENE QA Program is used on all other nuclear power plant work, and is Nuclear Regulatory Commission (NRC) accepted. The GENE QA Program complies with 10 CFR 50, Appendix B, the implementing ANSI/ASME N45.2 series daughter standards and the Regulatory Guides shown in NEDO-11209-04A, Table 2-1 with some NRC-accepted GE alternate positions. The QA Program meets Regulatory Guide 1.28, Revision 3, and is organized to show its relationship to ANSI/ASME NQA-1-1983 and NQA-1a-1983 and GE's interfaces with its project participants.

GENE ESBWR work is controlled through the NP2010 COL Demonstration Project Quality Assurance Plan, NEDO-33181 (Reference 17.0-2). NEDO-33181 provides the description of the quality assurance plan scope, which GENE, as supplier for ESBWR engineering services, will implement in support of DOE NP-2010 COL Demonstration Project.

Suppliers' and sub-tier suppliers' work is controlled through the SQAR – ESBWR QA Requirements for Procurement of Engineering Services and Equipment, NEDC-33260 (Reference 17.0-3). NEDC-33260 defines relationships, responsibilities, and requirements for the Supplier's quality program. All safety-related suppliers and sub-tier suppliers must have QA plans to meet the applicable requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983.

The evolution of the ESBWR Design and the use of SBWR test programs conducted at supplier test facilities for the GIRAFFE, PANTHERS and PANDA tests are discussed in detail in DCD Section 1.5. Each of these test programs was conducted under the appropriate provisions of NEDG-31831 (Reference 17.0-4), and implemented using GE approved supplier QA Plans. It was required that all of these supplier QA plans either met the requirements of ANSI/ASME NQA-1-1983 and NQA-1a-1983 addenda as endorsed by the NRC in Regulatory Guide 1.28, Revision 3, or the intent of these requirements by reference to equivalent national standards (such as the use of Japanese standard JEAG 4101-1990 for GIRAFFE). Additionally, NEDG-31831 provides that design and testing work performed by international technical associates will be performed to their internal QA programs acceptable to the regulatory authorities of their respective countries as evaluated by GENE for compliance with the provisions of ANSI/ASME NQA-1-1983 and NQA-1a-1983. The NRC participated in oversight activities related to the testing as documented in NRC Inspection Report (Reference 17.0-5). The NRC staff has conducted QA inspections of all of GE's major design certification test programs (GIST, Panthers/PCC, Panthers/IC, GIRAFFE, and PANDA) and has concluded that for GIST, Panthers, and Giraffe, NQA-1 standards were met, or that appropriate remedial actions were taken to correct deficiencies found during those inspections" (Reference 17.0-6).

17.0.1 References

17.0-1 GE Nuclear Energy, "GE Nuclear Energy Quality Assurance Program Description," NEDO 11209 04A (NRC accepted), March 1989.

17.0-2 GE Nuclear Energy, "NP2010 COL Demonstration Project Quality Assurance Plan," NEDO-33181.

17.0-3 GE Nuclear Energy, "NP2010 COL Demonstration Project SQAR – ESBWR QA Requirements for Procurement of Engineering Services and Equipment," NEDC-33260.

17.0-4 GE Nuclear Energy, "SBWR Design and Certification Program Quality Assurance Plan," NEDG-31831, May 1990.

17.0-5 USNRC, "NRC Inspection Report No. 99900404/95-02," MFN-196-95, September 25, 1995.

17.0-6 USNRC, "Staff Evaluation of General Electric's (GE's) Test and Analysis Program Description, NEDC-32391 Rev. C", MFN-119-96, July 11, 1996.

Table 17.0-1

Compliance With Quality Assurance Related Regulatory Guides

Regulatory Guide No.	Revision	Comments
1.8	3	COL holder scope
1.26	3	No exception
1.28	3	Except for NRC-accepted alternate positions in Table 2-1 of Reference 17.0-1
1.29	3	No exception
1.30	0	No exception
1.37	0	Except for NRC-accepted alternate positions in Table 2-1 of Reference 17.0-1
1.38	2	Except for NRC-accepted alternate positions in Table 2-1 of Reference 17.0-1
1.39	2	No exception
1.58	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.64	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3, except for NRC-accepted alternate positions in Table 2-1 of Reference 17.0-1
1.74	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.88	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.94	1	COL holder scope
1.116	0-R	No exception
1.123	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.144	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.146	Withdrawn	Superceded by Regulatory Guide 1.28, Rev. 3
1.176	0	COL holder scope

17.1 QUALITY ASSURANCE DURING DESIGN

The QA Program described in Section 17.1 is applicable to the ESBWR design activities supporting the standard design certification. Quality assurance is the responsibility of the DCD applicant for these design activities. The COL applicant is responsible for design activities related to a specific plant as defined in Section 17.2.

17.1.1 Organization

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 1, establishes the Organization structure used during design of the ESBWR.

17.1.2 Quality Assurance Program

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 2, establishes the Quality Assurance Program used during design of the ESBWR.

The identification of safety-related structures, systems and components (Q list) to be controlled by the GENE QA Program is shown in Table 3.2-1.

17.1.3 Design Control

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 3, establishes Design Control used during design of ESBWR. Minimum design requirements are identified in Table 3.2-2.

17.1.4 Procurement Document Control

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 4, establishes Procurement Document Control used during design of ESBWR.

17.1.5 Instructions, Procedures, and Drawings

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 5, establishes the Instructions, Procedures, and Drawings used during design of ESBWR.

17.1.6 Document Control

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 6, establishes Document Control used during design of ESBWR.

17.1.7 Control of Purchased Material, Equipment, and Services

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 7, establishes Control of Purchased Material, Equipment, and Services used during design of ESBWR.

17.1.8 Identification and Control of Materials, Parts, and Components

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 8, establishes Identification and Control of Materials, Parts, and Components during design of ESBWR.

17.1.9 Control of Special Processes

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 9, establishes Control of Special Processes used during design of ESBWR.

17.1.10 Inspection

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 10, establishes Inspection used during design of ESBWR.

17.1.11 Test Control

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 11, establishes Test Control used during design of ESBWR.

17.1.12 Control of Measuring and Test Equipment

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 12, establishes Control of Measuring and Test Equipment during design of ESBWR.

17.1.13 Handling, Storage and Shipping

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 13, establishes Handling, Storage and Shipping used during design of ESBWR.

17.1.14 Inspection, Test, and Operating Status

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 14, establishes the control of Inspection, Test, and Operating Status used during design of ESBWR.

17.1.15 Nonconforming Materials, Parts, or Components

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 15, establishes the control of Nonconforming Materials, Parts, or Components used during design of ESBWR.

17.1.16 Corrective Action

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 16, establishes the Corrective Action program used during design of ESBWR.

17.1.17 Quality Assurance Records

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 17, establishes control of Quality Assurance Records used during design of ESBWR.

17.1.18 Audits

GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 18, establishes a comprehensive system of QA Audits used during design of ESBWR.

17.1.19 Training and Qualification Criteria – Quality Assurance

In accordance with GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 2, Training and Qualification of QA personnel is procedurally established and maintained.

17.1.20 Training and Qualification – Inspection and Test

In accordance with GENE QA Program Description, NEDO-11209-04A (Reference 17.1-1) Section 2, Training and Qualification of Inspection and Test personnel is procedurally established and maintained.

17.1.21 QA Program Commitments

Regulatory Guides and Standards and their respective revisions, including exceptions, are addressed in the appropriate DCD section.

17.1.22 10 CFR Part 21 and 10 CFR 50.55(e) Programs

GENE has an established 10 CFR Part 21 notification and posting system which is procedurally controlled. The requirement of 10 CFR Part 21 is imposed on all safety-related purchase documents. The construction permit holder is required to adopt procedures to identify and evaluate defects and failures to comply associated with substantial safety hazards under the requirements of 10 CFR 50.55 (e).

17.1.23 Commercial Grade Dedication

GENE has procedurally established a Commercial Grade dedication program which meets the requirements of 10 CFR Part 21.

17.1.24 Digital Equipment Software Verification and Validation Quality Controls

GENE has established an ESBWR I&C Software QA Plan, NEDE-33245 (Reference 17.1-3) which addresses Software Verification and Validation Quality Controls. Software Design Verification and Validation is discussed in Section 7.8.2.1 and Appendix 7B.

17.1.25 Nonsafety-Related SSC Quality Controls

Nonsafety-related SSC(s) that perform safety significant functions have QA requirements applied commensurate with the importance of the item's function. The identification of nonsafety-related structures, systems and components is shown in Table 3.2 1.

17.1.26 Independent Review

This section will be addressed by and is the responsibility of the COL applicant.

17.1.27 References

17.1-1 GE Nuclear Energy, "GE Nuclear Energy Quality Assurance Program Description," NEDO-11209-04A (NRC accepted), March 1989.

17.1-2 GE Nuclear Energy, "NP-2010 COL Demonstration Project Quality Assurance Plan," NEDO-33181.

17.1-3 GE Nuclear Energy, “ESBWR I&C Software QA Plan,” NEDE-33245.

17.2 QUALITY ASSURANCE DURING CONSTRUCTION AND OPERATIONS

QA responsibilities during construction and operations are Combined Operating License (COL) applicant scope. Quality assurance during design activities necessary to adapt the certified standard plant design to the specific plant implementation is the responsibility of the COL applicant.

17.3 QUALITY ASSURANCE PROGRAM DOCUMENT

The project overall Quality Assurance Program Document (QAPD) from the applicant is a COL applicant/holder responsibility. The QAPD applied by the Design Team during the design certification phase is described in Section 17.1.

17.4 RELIABILITY ASSURANCE PROGRAM DURING DESIGN PHASE

This section presents the ESBWR Design Reliability Assurance Program (D-RAP).

17.4.1 Introduction

The GE ESBWR D-RAP is a program utilized during detailed design and specific equipment selection phases to assure that the important ESBWR reliability assumptions of the probabilistic risk assessment (PRA) are considered throughout the plant life. The PRA is used to evaluate the plant response to anticipated operational occurrence (AOO) initiating events and mitigation to ensure potential plant damage scenarios pose a very low risk to the public.

The D-RAP identifies relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limiting risk to the public. An example is provided to demonstrate how the D-RAP applies to the Isolation Condenser System (ICS). The ICS example shows how the principles of D-RAP are applied to other systems identified by the PRA as being risk-significant.

17.4.2 Scope

The scope of the ESBWR D-RAP includes risk-significant SSCs, both safety-related and nonsafety-related, that provide defense-in-depth or result in significant improvement in the PRA evaluations.

A preliminary list of risk-significant SSCs within the scope of the D-RAP is developed in the design phase.

The list is updated, using a blended approach and an Expert Panel when plant-specific information is available. This information forms the basis for the Maintenance Rule program, which ensures that risk-significant SSCs operate throughout plant life with reliable performance that is consistent with the PRA. The PRA for the ESBWR, and other sources, are used to identify and prioritize those SSCs that are important to prevent or mitigate plant AOOs or other events that could present a risk to the public.

17.4.3 Purpose

The purpose of the D-RAP is to ensure that the plant safety, as estimated by the PRA, is maintained as the detailed design evolves through the implementation and procurement phases, and that pertinent information is provided in the design documentation to the future owner/operator so that equipment reliability, as it affects plant safety, is maintained through operation and maintenance during the entire plant life.

17.4.4 Objective

The objective of the D-RAP is to identify those plant SSCs that are significant contributors to risk, as shown by the PRA or other sources, and to assure that, during the implementation phase, the plant design continues to utilize risk-significant SSCs whose reliability is commensurate with the PRA assumptions. Reliability includes ensuring that risk-significant SSCs do not degrade to an unacceptable level during plant operations, and that the frequency of AOOs posing challenges to risk-significant SSCs is minimized. The D-RAP also identifies key assumptions regarding any operation, maintenance and monitoring activities that the owner/operator should consider in developing its O-RAP to assure that such SSCs function reliably when challenged throughout plant life with reliability consistent with that assumed in the PRA.

17.4.5 GE Organization for D-RAP

The GE ESBWR Engineering Section is an integrated design and engineering organization that is responsible for formulating and implementing the D-RAP. The Manager, ESBWR Engineering is responsible for the design and licensing of the ESBWR, and for development of the D-RAP. The COL applicant is responsible for implementing the operations phase of the RAP.

The ESBWR Engineering organization is responsible for the design analysis and PRA engineering that is necessary to support the development of the D-RAP. PRA personnel and design engineering personnel report to the Manager of ESBWR Engineering. As such, the PRA personnel are directly involved with the design organization and keep the design staff cognizant of risk-significant items, program needs, and project status. PRA personnel participate in the design change control process, which includes providing D-RAP related inputs in the design process.

GE ESBWR engineering design procedural controls are applied to the D-RAP. Specific procedures provide guidance on the design process, control of design changes, and storage and retrieval controls.

The design control procedure defines the process for performing, documenting, and verifying design activities. This includes developing or modifying the design of systems, engineering evaluations, analyses, calculations and document preparation, (e.g., specifications, drawings, reports.)

The procedure for design change control defines the process for evaluating design changes in engineering controlled documents to ensure that the total effect is considered before a change is approved, and the affected documents are identified and changed accordingly. The procedure identifies interfaces and organizations responsible for these interfaces, including PRA review. If a proposed change could affect the safety, availability or capacity factor of the ESBWR plant, system reliability is analyzed.

Several design control procedures provide guidance for developing a high quality process for reliability assurance. The documentation procedure establishes the requirements and responsibilities for the preparation, approval, and issue of documents controlled by the engineering design organizations. The quality assurance records procedure provides requirements for quality assurance record retention. The self-assessment, corrective action and audits procedure specify the responsibilities for performing self-assessments; internal audits of the engineering organization; and prompt identification, documentation, and corrective actions on conditions that are adverse to quality.

In addition to the standard engineering design processes and quality controls, specific guidance is used to define and implement an effective RAP. Reference 17.4-1 describes the RAP processes for identifying and prioritizing risk significance, implementing reliability assurance strategies, and monitoring program effectiveness. It is used to incorporate reliability assurance into each aspect of the design, construction, testing, and operation of the ESBWR.

17.4.6 SSC Identification/Prioritization

A list of risk-significant SSCs is developed and controlled as a design specification document. The preliminary list is based on the results of the generic PRA. The list is updated when the plant-specific PRA is developed. At this point, a blended approach is used for identifying and prioritizing risk significant SSCs. This approach combines the various PRA analytical results with operating experience and an expert panel process to develop a comprehensive risk analysis.

The level 1 PRA is used to evaluate accident sequences from initiating events and failures of safety functions that lead to core damage. An assessment is performed for operating and shutdown conditions. The external events analysis considers events whose cause is external to systems associated with normal plant operations, including internal flooding, fire, high winds, and seismic events. The seismic events are analyzed using a seismic margins approach that provides qualitative conclusions on the ability of ESBWR SSCs to cope with seismic events. The other external events are quantified using the level 1 PRA.

Level 1 basic events representing component failures are identified as risk-significant if their importance values for Risk Achievement Worth are greater than or equal to 5.0, or Fussell-Vesely Importance are greater than or equal to 0.01.

Level 2 risk significance is determined by identifying the dominant contributors to severe accidents and offsite release of fission products. This qualitative analysis, which is performed by the expert panel, includes the evaluation of severe accident phenomena and fission product source terms, and containment integrity strategies including pressure suppression, decay heat removal, and hydrogen generation.

SSC functions relied upon under power-operating and shutdown conditions to meet the NRC's safety goal guidelines of a CDF of less than 1.0E-4 per reactor year and Large Release Frequency of less than 1.0E-6 per reactor year are risk-significant. SSC functions needed to meet the containment performance goal, including containment bypass, during severe accidents are also risk-significant.

Operating experience identifies previous failures of components in similar applications, and also reveals situations where inappropriate human actions have led to functional failures of SSCs. The expert panel assesses component operating history and industry operating experience when it can be applied to assessing risk significance.

Safety-related SSCs are already controlled by plant Technical Specifications. If a non-safety SSC is shown through operating experience or PRA to be significant to public health and safety, then it should be controlled by Technical Specifications. In this case, “significant” equates to an SSC that is required to meet the NRC Safety Goals. If it is determined that an SSC is risk significant, but is not required for meeting the NRC Safety Goals, then performance controls should be implemented through the RAP. If the SSC is not significant, then normal controls would be implemented through the site Maintenance Rule and corrective action programs.

17.4.7 Design Considerations

The reliability of risk-significant SSCs, which are identified by the PRA and other sources, are evaluated at the detailed design stage by appropriate design reviews and reliability analyses. The procedure for design change control defines the process for evaluating design changes in engineering controlled documents to ensure that the total effect is considered before a change is approved, and the affected documents are identified and changed accordingly.

A design reliability assessment is a process in which the design engineer builds quality and reliability into the SSC, while ensuring that the basis for SSC design is properly modeled in the PRA. Due to the preliminary nature of the PRA model during the design phase, the model relies on generic information, bounding assumptions, or design requirements as a basis for model development. This design assessment can be performed for changes that occur during the plant design phase, as well as during normal plant operations. It is a systematic method to evaluate the proposed design details with respect to PRA insights. The assessment considers reliability concepts, such as redundancy, diversity, human factors, spatial interactions, external events, etc., to enhance the system design, and considers PRA insights and assumptions. If the assessment reveals that the proposed design could conflict with results and insights calculated in the PRA, or could cause significant unavailability of a safety function, then a design change is pursued.

Proposed design changes are processed by the design change control procedure, which requires PRA review. If a design change affects the PRA model, then it is revised in accordance with the PRA update process described in the PRA procedure.

17.4.8 Defining Failure Modes

The determination of dominant failure modes of risk-significant SSCs includes historical information, analytical models and existing requirements. Many BWR systems and components have compiled a significant historical record, so an evaluation of that record is performed. For those SSCs for which there is not an adequate historical basis to identify critical failure modes, an analytical approach is necessary.

Inputs may include PRA importance analysis, root cause analysis, failure modes and effects analysis, and review of operating experience. In addition, equipment performance information, including vendor manuals, ASME Section XI, technical specifications, Regulatory Treatment of Non-Safety Systems (RTNSS), and other regulatory requirements are reviewed to identify important safety functions.

The design engineer analyzes this information to identify dominant failure modes, such as single failures, latent failures not detected by routine monitoring, common cause failures, or failures that could cascade into more significant safety functional failures.

17.4.9 Operational Reliability Assurance Activities

Once the dominant failure modes are determined for risk-significant SSCs, an assessment is performed to identify O-RAP activities that assure acceptable performance during plant life. Such activities may consist of periodic surveillance inspections or tests, monitoring of SSC performance, and/or periodic preventive maintenance. Some SSCs may require a combination of activities to assure that their performance is consistent with that assumed in the PRA.

Periodic testing of SSCs may include startup of standby systems, surveillance testing of instrument circuits to assure that they respond to appropriate signals, and inspection of SSCs (such as tanks and pipes) to show that they are available to perform as designed. Performance monitoring, including condition monitoring, can consist of measurement of output (such as pump flow rate or heat exchanger temperatures), measurement of magnitude of an important variable (such as vibration or temperature), and testing for abnormal conditions (such as oil degradation or local hot spots).

Periodic preventive maintenance is an activity performed at regular intervals to preclude problems that could occur before the next preventive maintenance (PM) interval. This could be regular oil changes, replacement of seals and gaskets, or refurbishment of equipment subject to wear or age related degradation.

Planned maintenance activities will be integrated with the regular operating plans so that they do not disrupt normal operation. Maintenance that is performed more frequently than refueling outages must be planned so as to not disrupt operation or be likely to cause reactor scram, engineered safety feature actuation or AOOs. Maintenance planned for performance during refueling outages must be conducted in such a way that it has little or no effect on plant safety, outage length or other maintenance work.

The COL applicant shall provide a complete O-RAP to be reviewed by the NRC. See Subsection 17.4.13 for COL applicant information.

17.4.10 Owner/Operator's Reliability Assurance Program

The O-RAP is prepared and implemented by the ESBWR owner/operator, and uses the information provided by GE. This information should help the owner/operator determine activities that should be included in the O-RAP. Examples of elements that might be included in an O-RAP are as follows:

Problem Prioritization: Identification, for each of the risk-significant SSCs, of the importance of that item as a contributor to its system unavailability and assignment of priorities to problems that are detected with such equipment.

Corrective Action Implementation: Carrying out identified corrective action on risk-significant equipment to restore equipment to its intended function in such a way that plant safety is not compromised during work.

Plant Aging: Some of the risk-significant equipment is expected to undergo age related degradation and require equipment replacement or refurbishment.

- **Programmatic Interfaces:** Reliability assurance interfaces related to the work of the several organizations and personnel groups working on risk-significant SSCs.
- **Maintenance Rule Program:** A procedure is developed by the COL holder to implement a Maintenance Rule program with the following scope:
  - Selection of SSCs for inclusion.
  - Establishing and applying safety significant criteria.
  - Setting performance monitoring criteria.
  - Trending the performance of applicable SSCs to demonstrate the effectiveness of maintenance activities.
  - Taking corrective action when SSC performance degrades.
  - Periodically assessing program performance.

- Identifying documentation that is required to support the program.
- Maintenance Rule (a)(4) assessment of real-time risk profile

The plant owner/operator's O-RAP addresses the interfaces with construction, startup testing, operations, maintenance, engineering, safety, licensing, quality assurance and procurement of initial and replacement equipment.

17.4.11 D-RAP Implementation

17.4.11.1 Example: Isolation Condenser System

The ICS is used as an example to demonstrate how the reliability assurance processes are used to identify, analyze, and develop effective reliability assurance strategies. ICS is a safety-related system that removes reactor decay heat following events involving reactor shutdown and containment isolation. It also prevents unnecessary reactor depressurization, and precludes the need for operation of other Engineered Safety Features to bring the reactor to a safe and stable condition. In the event of a LOCA, ICS provides additional liquid inventory by opening the condensate return valves to actuate the system. ICS also assists with initial depressurization of the reactor before ADS in event of loss of feed water, so that the automatic depressurization can take place from a lower pressure.

The ICS consists of four totally independent trains, each containing an isolation condenser that condenses steam on the tube side and transfers heat to the IC/PCC pool, which is vented to the atmosphere. The isolation condensers are connected by piping to the reactor pressure vessel, and are placed at an elevation above the source of steam (i.e., vessel). When the steam is condensed, the condensate is returned to the vessel via a condensate return line. A detailed description of ICS is located in DCD Tier 2 Subsection 5.4.6.

The major differences between the ESBWR ICS and the conventional BWR ICS are:

- Use of four heat exchangers instead of one or two in conventional BWRs.
- Parallel path for condensate return to the vessel instead of single injection path.
- Use of both Nitrogen-Operated and Motor-Operated Valves (MOVs) for condensate return instead of only MOVs.
- Use of large cooling pools instead of shell-side heat exchangers**.**

The design features of the ESBWR ICS contain significant improvements in reliability and availability that are risk-based. The number of heat exchangers is increased for redundancy. The condensate return line to the vessel has two paths for success and each path uses a diverse isolation valve. The large capacity IC pools provide cooling capacity for 72 hours following a reactor scram. Conventional BWRs typically have 20 to 30 minutes of cooling water capacity.

17.4.11.2 Identifying Risk Information

In order to examine the relative importance or dominance of failures of ICS components, a fault tree has been developed with the top gate defined as failure of the ICS to inject water into the RPV upon demand. This tree considers the worst-case scenario with respect to AOOs and accidents, which involves a success criterion that 3 of 4 ICS subsystems must function. This requires a condensate return path and a vent path for non-condensables for each functioning subsystem. This fault tree is quantified to identify the relative importance of ICS components as they contribute to system unreliability.

A risk ranking of the ICS basic events has been performed to identify SSCs with the greatest importance. The ranking is performed using the ICS top event model, described above. In addition a risk ranking is performed using the CDF top event (PRA) model to provide further perspective on the importance of ICS components. The results of the risk rankings are provided in Table 17.4-1.

17.4.11.3 Failure Mode Identification

The importance analysis results indicate that no single SSC has a dominant effect on ICS system unavailability. Therefore, the design and selection of ICS components appears to be reasonable.

The dominant failures, as shown in Table 17.4-1, involve valves. Operating experience indicates that valves, in general, are subject to mechanical problems such as valve stem failure, separation of stem from disk, and failure to stroke. In addition, remote actuated valves can experience actuator failures, electrical failures in the motor winding or motor internals, and problem with torque limit switches and switch settings.

For ICS, the dominant failures involve the condensate return nitrogen-operated isolation valves (B32-F006A, B, C, D). The parallel condensate return valves, B32-F005A, B, C, D are not considered to be dominant failures due to their dependency on AC power and the relative importance of loss of preferred power sequences.

According to the design specifications, the condensate return valve, (F006) is a spring-loaded, pneumatic, piston-operated globe valve, designed to fail open on loss of pneumatic pressure to the valve actuator. This valve is also signaled to open when reactor water level drops to Level 2. A pneumatic accumulator is located close to the valve to provide pneumatic pressure for the purpose of assisting in valve closure when both pilots are energized or in the event of failure of pneumatic supply pressure to the valve operator. Examples of the types of failure that could affect valve reliability are shown in Table 17.4-2. Because the design details are not finalized, this table is not a comprehensive listing. It is intended to indicate the types of failures that are considered for the purpose of providing an example.

17.4.11.4 Identification of Maintenance Requirements

Maintenance activities are developed to assure that the dominant failure modes are reduced, or kept to an acceptably low probability. The types of maintenance and the maintenance frequencies are both important aspects of ensuring that the equipment failure rate will be consistent with that assumed in the PRA model. The designer considers periodic or condition-based testing and maintenance activities to keep the unreliability to an acceptable level.

In this example, the D-RAP analytical process results in a preliminary recommendation for quarterly valve testing of the B32-F006 valves, along with flow testing during each refueling outage. This helps to preserve the unreliability values used in the PRA model. In addition, the B32-F005 valves, which are in the parallel path, are recommended to receive the same testing requirements. This will ensure that these valves do not experience degraded performance that could increase their risk significance.

The recommended maintenance activities and performance monitoring will be governed by the O-RAP and Maintenance Rule Program.

17.4.12 Glossary of Terms

Design Reliability Assurance Program — Performed by the plant designer to assure the plant is designed so that it can be operated and maintained in such a way that the reliability assumptions of the probabilistic risk assessment apply throughout plant life.

Fussell-Vesely Importance — A measure of the component contribution to core damage frequency. Numerically, the percentage contribution of the component to CDF.

Owner/Operator — The utility or other organization that owns and operates the ESBWR following construction.

Operational Reliability Assurance Program — Performed by the plant owner/operator to assure the plant is operated and maintained safely and in such a way that the reliability assumptions of the PRA apply throughout plant life.

Regulatory Treatment of Non-Safety Systems (RTNSS) – A process to determine whether regulatory oversight for certain nonsafety-related systems is needed, and to determine an appropriate level of regulatory oversight commensurate with their risk significance.

Risk-Significant — Those structures, systems and components that are identified as contributing significantly to the core damage frequency.

17.4.13 COL Information

17.4.13.1 Provision for O-RAP

The COL applicant/holder will provide a complete O-RAP to be reviewed by the NRC (Subsection 17.4.9).

17.4.14 References

17.4-1 GE Nuclear Energy, “Reliability Assurance Program Plan”, NEDO-33289.

Table 17.4-1

ICS Example Importance Analysis

Component	Description	Fussell-Vesely Importance	Risk Achievement Worth
B32-F006B	Air Operated Valve F006B Fails To Open	8.27E-04	1.41
B32-F006C	Air Operated Valve F006C Fails To Open	8.27E-04	1.41
B32-F006D	Air Operated Valve F006D Fails To Open	8.27E-04	1.41
B32-F006A	Air Operated Valve F006A Fails To Open	1.04E-04	1.05

Table 17.4-2

ICS Example Failure Modes and Reliability Strategy

Component	Failure Mode	Cause	Reliability Strategy
B32-F006A, B, C, D	Failure to Open due to mechanical problems	Binding, fatigue failure, foreign material	Inspect Valve Internals, System Flow Test, Valve Stroke Test
	Failure to Open due to electrical problems with valve operator	Windings, Wiring, Relays, Contacts	Logic System Functional Test, Valve Stroke Test