Chapter 17

VCS UFSAR Table of Contents

VCS UFSAR Formatting Legend

Color	Description
(black)	Original Westinghouse AP1000 DCD Revision 19 content
(purple)	Departures from AP1000 DCD Revision 19 content
(blue)	Standard FSAR content
(green)	Site-specific FSAR content
(red)	Linked cross-references (chapters, appendices, sections, subsections, tables, figures, and references)

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Chapter 17 Quality Assurance

17.1 Quality Assurance During the Design and Construction Phases

South Carolina Electric & Gas (SCE&G) is responsible for the establishment and execution of quality assurance program requirements during the design and construction phases of V.C. Summer Nuclear Station Units 2 and 3. SCE&G may delegate and has delegated to others, such as NuStart Energy Development, LLC, Westinghouse Electric Company, and Bechtel Power Corporation, the work of establishing and executing the quality assurance program, or any parts thereof, but retains responsibility for the quality assurance program.

Effective during COL application development, through and until COL issuance, the NuStart Energy Development, LLC (NuStart) Quality Assurance Plan (Reference 201), the Westinghouse Electric Company Quality Management System (Section 17.6) and Bechtel Power Corporation Nuclear Quality Assurance Manual (Reference 205) define the QA program requirements for design activities. Safety-related construction activities at SCE&G are not planned before the COL is issued.

NuStart was created by multiple utilities and, as such, is comprised of multiple member utilities to include SCE&G. NuStart's purpose is two-fold of which the first is demonstrating the licensing process defined by 10 CFR Part 52. The second purpose is to work with reactor vendors in completing the engineering work for the standardized plant designs.

Bechtel Power Corporation was contracted by SCE&G to develop the V. C. Summer Nuclear Station Units 2 and 3 COL application. This included the process of collection, review and analysis of specific data for site characterization and was controlled under the Bechtel Power Corporation Quality Assurance Manual (Reference 205).

SCE&G maintains oversight of the COL application development as well as design and construction activities under its existing 10 CFR Part 50, Appendix B, program as described in the NRC approved SCE&G V.C. Summer Nuclear Station Unit 1 "Operational Quality Assurance Plan" (Reference 206). The Unit 1 Quality Assurance Program complies with the requirements of Regulatory Guide 1.28, Revision 0 – "Quality Assurance Program Requirements (Design and Construction)". The "Operational Quality Assurance Plan" (Reference 206) is supplemented, in part, by the SCE&G "New Nuclear Deployment Quality Assurance Plan" (Reference 204). The "New Nuclear Deployment Quality Assurance Plan (Reference 204) serves as an interfacing document between the work activities of the New Nuclear Deployment organization and the "Operational Quality Assurance Plan" (Reference 206). It assures that the proper administrative controls and the quality of activities related to the procurement of services, equipment, oversight of construction/manufacturing, and licensing activities being performed within the New Nuclear Deployment organization conform to the applicable requirements of 10 CFR 50, Appendix B. These plans provide the necessary quality assurance guidance for oversight of site characterization activities and COL application content providers. SCE&G maintains this oversight through the review and approval of the NuStart Quality Assurance Plan (Reference 201) and industry standard COL application sections as well as conducting audits/ surveillances of Bechtel activities, and providing input to the COL application development, including, but not limited to, review of COL application content.

Implementation of the applicable portions of the "Quality Assurance Program Description" (QAPD) discussed in Section 17.5 begins at COL issuance. The program establishes the quality assurance program requirements for the remaining portion of the design and construction phases and for operations; full implementation of the operations related requirements will be no later than as indicated in Table 13.4-201.

17.2 Quality Assurance During the Operations Phase

See Section 17.5.

17.3 Quality Assurance During Design, Procurement, Fabrication, Inspection, and/or Testing of Nuclear Power Plant Items

This section outlines the quality assurance program applicable to the design, procurement, fabrication, inspection, and/or testing of items and services for the AP1000 Project. The design for AP1000 is based upon employing the design of AP600 to the maximum extent possible. As a result, a continuous quality program spanning AP600 design as well as AP1000 design has been used. Westinghouse has and will continue to maintain a quality assurance program meeting the requirements of 10 CFR 50 Appendix B for the AP1000 program that will be applicable to the design, procurement, fabrication, inspection, and/or testing activities.

Effective March 31, 1996, activities affecting the quality of items and services for the AP600 Project during design, procurement, fabrication, inspection, and/or testing were being performed in accordance with the quality plan described in "Westinghouse Electric Corporation – Energy Systems Business Unit, Quality Management System," (Reference 1). The Quality Management System (QMS) has been maintained as the Quality Plan for the AP1000 program and subsequent revisions have been submitted to and accepted by the NRC as meeting the requirements of 10 CFR 50 Appendix B.

Prior to introduction of the QMS as the quality plan applicable to the AP1000 project, activities on the AP600/AP1000 program were performed in accordance with topical report WCAP 8370 (References 2 and 3), Westinghouse Energy Systems Business Unit/Power Generation Business Unit Quality Assurance Plan. WCAP 8370 was subsequently superceded by the Westinghouse QMS to describe the quality assurance plan and Westinghouse commitments to meet the requirements of 10 CFR 50 Appendix B.

The current Westinghouse quality plan for work being performed on the AP1000 is the Westinghouse Electric Company Quality Management System (QMS) (Reference 9). The referenced revision of the QMS was accepted by the NRC as meeting the requirements of 10 CFR 50, Appendix B, on September 13, 2002.

A project-specific quality plan was issued to supplement the quality management system document and the topical reports for design activities affecting the quality of structures, systems, and components for the AP600 project (Reference 4). This plan referenced the NQA-1-1989 edition through NQA-1b-1991 addenda and was applicable to work performed for the AP1000 design prior to March 16, 2007.

Effective March 16, 2007, NQA-1-1994 is the applicable revision of NQA-1 for work performed for the AP1000 project. As such, a project-specific quality plan is no longer required, and the Westinghouse Electric Company Quality Management System (QMS) (Reference 9) is the quality program for work performed for the AP1000 project.

Quality Assurance requirements for systems, structures, and components will be graded based on the safety classification as indicated in Section 3.2. Safety-related systems are classified as Equipment Classes A, B and C, and will meet the requirements of 10 CFR 50, Appendix B. For systems, structures, and components included in the regulatory treatment of nonsafety systems (RTNSS), the quality requirements are identified in Table 17-1. See Section 16.3 for systems that should be considered for designation of systems and components included in the regulatory treatment of nonsafety systems.

While Westinghouse retains the overall responsibility for the AP1000 design, portions of the design are developed by external organizations. Each organization maintains a quality assurance program that meets the NQA-1 criteria that apply to its work scope. In accordance with the QMS, Westinghouse performs an initial evaluation of these programs and monitors their continued effective implementation through audits, surveillance, and evaluation of the performance of external organizations.

17.4 Design Reliability Assurance Program

This subsection presents the AP1000 Design Reliability Assurance Program (D-RAP).

17.4.1 Introduction

The AP1000 D-RAP is implemented as an integral part of the AP1000 design process to provide confidence that reliability is designed into the plant and that the important reliability assumptions made as part of the AP1000 probabilistic risk assessment (PRA) (Reference 5) will remain valid throughout plant life. The PRA quantifies plant response to a spectrum of initiating events to demonstrate the low probability of core damage and resultant risk to the public. PRA input includes specific values for the reliability of the various structures, systems, and components (SSCs) in the plant that are used to respond to postulated initiating events.

The D-RAP, shown in Figure 17.4-1, is implemented during Design Certification. The D-RAP identifies risk-significant SSCs for inclusion into the site Operational Phase Reliability Assurance Activities (OPRAAs) using probabilistic, deterministic, and other methods.

The OPRAAs provides confidence that the operations and maintenance activities performed by the operating plant support should maintain the reliability assumptions made in the plant PRA.

17.4.2 Scope

The D-RAP includes a design evaluation of the AP1000 and identifies the aspects of plant operation, maintenance, and performance monitoring pertinent to risk-significant SSCs. In addition to the PRA, deterministic tools, industry sources, and expert opinion are used to identify and prioritize those risk-significant SSCs.

The quality assurance requirements for non-safety related SSCs within the scope of D-RAP is in accordance with the Quality Assurance Program Description (QAPD), Part III.

17.4.3 Design Considerations

As part of the design process, risk-significant components are evaluated to determine their dominant failure modes and the effects associated with those failure modes. For most components, a substantial operating history is available which defines the significant failure modes and their likely causes.

The identification and prioritization of the various possible failure modes for each component lead to suggestions for failure prevention or mitigation. This information is provided as input to the OPRAAs.

The design reflects the reliability values assumed in the design and PRA as part of procurement specifications. When an alternative design is proposed to improve performance in either area, the revised design is first reviewed to provide confidence that the current assumptions in the other areas are not violated. When a potential conflict exists between safety goals and other goals, safety goals take precedence.

17.4.4 Relationship to Other Administrative Programs

The D-RAP manifests itself in other administrative and operational programs. The technical specifications provide surveillance and testing frequencies for certain risk-significant SSCs, providing confidence that the reliability values assumed for them in the PRA will be maintained during plant operations. Risk-significant systems that provide defense-in-depth or result in significant improvement in the PRA evaluations are included in the scope of the D-RAP.

The OPRAAs are comprised of site administrative, maintenance, operational, and testing programs to enhance operational phase reliability throughout the designed plant life. As documented in Reference 10 and Reference 12, the following reliability assurance programs are credited as OPRAAs:

- Maintenance Rule Program (Reference 10)
- Quality Assurance Program (Section 17.2)
- Inservice Testing Program (Section 3.9)
- Inservice Inspection Program (Section 5.2 and Section 6.6)
- Technical Specifications Surveillance Test Program (Section 16.1)
- AP1000 Investment Protection Short Term Availability Controls Program (Section 16.3)
- Site Maintenance Program

17.4.5 The AP1000 Design Organization

The AP1000 organization of Section 1.4 formulates and implements the AP1000 D-RAP.

The AP1000 management staff is responsible for the AP1000 design and licensing.

The AP1000 staff coordinates the program activities, including those performed within Westinghouse as well as work completed by the architect-engineers and other supporting organizations listed in Section 1.4.

The AP1000 staff is responsible for development of the D-RAP and the design, analyses, and risk and reliability engineering required to support development of the program. Westinghouse is responsible for the safety analyses, the reliability analyses, and the PRA.

The reliability analyses are performed using common databases from Westinghouse and from industry sources such as INPO and EPRI.

The Risk and Reliability organization is responsible for developing the D-RAP and has direct access to the AP1000 staff. Risk and Reliability is responsible for keeping the AP1000 staff cognizant of the D-RAP risk-significant items, program needs, and status. Risk and Reliability participates in the design change control process for the purpose of providing D-RAP-related inputs to the design process. Additionally, a cognizant representative of Risk and Reliability is present at design reviews. Through these interfaces, Risk and Reliability can identify interfaces between the performance of risk-significant SSCs and the reliability assumptions in the PRA. Meetings between Risk and Reliability and the designer are then held to manage interface issues.

17.4.6 Objective

The objective of the D-RAP is to design reliability into the plant and to maintain the AP1000 reliability consistent with the NRC-established PRA safety goals.

The following goals have been established for the D-RAP:

- Provide reasonable assurance that
  - The AP1000 is designed, procured, constructed, maintained and operated in a manner consistent with the assumptions and risk insights in the AP1000 PRA for these risk-significant SSCs
  - The risk-significant SSCs do not degrade to an unacceptable level during plant operations
  - The frequency of transients that challenge the AP1000 risk-significant SSCs are minimized
  - The risk-significant SSCs function reliably when they are challenged
- Provide a mechanism for establishing baseline reliability values for risk-significant SSCs identified by the risk determination methods used to implement the Maintenance Rule (10 CFR 50.65) and consistent with PRA reliability and availability design basis assumptions used for the AP1000 design
- Provide a mechanism for establishing baseline reliability values for SSCs consistent with the defense-in-depth functions to minimize challenges to the safety-related systems
- Generate design and operational information to be used for ongoing plant reliability assurance activities

Development of maintenance assessments and recommendations and the site-specific portion of the program is the responsibility of the Combined License applicant.

17.4.7 D-RAP

The definition portion of the D-RAP includes the initial identification of SSCs to be included in the program, implementation of the aspects applicable to design efforts, and definition of the scope, requirements, and implementation options to be included in the later phases.

17.4.7.1 SSCs Identification and Prioritization

The initial task of the D-RAP is identification of risk-significant SSCs to be included within the scope of the program. As shown in Figure 17.4-1, the AP1000 PRA is used to identify those SSCs, consistent with the criteria of Reference 7 for risk achievement worth (RAW), risk reduction worth (RRW), and Fussel-Vesely worth (FVW). Note that, although Reference 7 was developed for AP600, it is directly applicable to AP1000. The review of light water reactor industry experience and industry notices (such as licensee event reports) supports the process. An expert panel is also employed in the selection process.

PRA-based measurements provide information that contributes to the identification and prioritization of SSCs. A component's RAW is the factor by which the plant's core damage frequency increases if the component reliability is assigned the value 0.0. Components with risk achievement worth values of 2 or greater are considered for inclusion in the D-RAP.

RRW is used in the selection process. A component's risk reduction worth is the amount by which the plant's core damage frequency decreases if the component's reliability is assigned the value 1.0. A threshold measure of 1.005 or greater is used as the cutoff. Components with RRW of 1.005 or greater are considered for inclusion in the D-RAP.

FVW is also used in the screening process. This is a measure of an event's contribution to the overall plant core damage frequency. Components with Fussel-Vesely worth of 0.5 percent or greater are considered for inclusion in the D-RAP.

Deterministic considerations are also instrumental in identifying risk-significant SSCs. The deterministic identification of risk-significant SSCs encompasses the following guidelines and considerations:

- ATWS rule (10 CFR 50.62)
- Loss of all ac power (10 CFR 50.63)
- Post-72-hour actions
- Containment performance
- Adverse interactions with the AP1000 safety-related systems
- Seismic considerations

Nonsafety-related systems identified as risk-significant are considered in the scope of the D-RAP:

- Diverse actuation system
- Non-Class 1E dc and uninterruptible power supply system
- Offsite power, main ac power, and onsite standby power systems
- Normal residual heat removal system
- Component cooling water system
- Service water system

Finally, risk-significant SSCs are selected using industry experience, regulations, and engineering judgment.

17.4.7.1.1 Level 1 PRA and Shutdown Analysis

The Level 1 PRA evaluates accident sequences from initiating events and failures of safety functions to core damage events. The probability of core damage and the identification of dominant contributors to that state are also determined in this analysis.

A low-power and shutdown assessment is conducted to address concerns about risk of operations during shutdown conditions. It encompasses operation when the reactor is in a subcritical state or is in a transition between subcriticality and power operation up to 5 percent of rated power. It consists of a Level 1 PRA and an evaluation of release frequencies and magnitudes.

Included in the D-RAP are events that meet the threshold risk achievement worth, risk reduction worth, or Fussel-Vesely worth values defined in Subsection 17.4.7.1.

17.4.7.1.2 Level 2 Analysis

The Level 2 analysis predicts the plant response to severe accidents and offsite fission product releases. Specifically, the analysis includes the following sections:

- Evaluating severe accident phenomena and fission product source terms
- Modeling the containment event tree
- Analyzing hydrogen burn, mixing, and igniter placement
- Modeling the AP1000 utilizing the MAAP4 code

Equipment used in the prevention of severe accidents and severe post-accident boundary conditions is credited in the Level 1 and Level 2 PRA analyses. An example of this preventive equipment is the reactor coolant system automatic depressurization system (ADS). Successful depressurization leads to core cooling, and in the event that injection fails, results in a low pressure core damage sequence that has fewer uncertainties and can be more easily mitigated than high pressure core damage.

The containment event tree used in the AP1000 Level 2 PRA examines the operation of equipment which mitigates the threat to the containment from severe accident phenomena. The systems credited for the mitigation of large fission product releases are containment isolation, passive containment cooling water (PCS), and operator action to flood the cavity by opening the recirculation valves and energizing the hydrogen igniters.

17.4.7.1.3 External Event Analyses

These analyses consider the events whose cause is external to all the systems associated with normal and emergency operations situations. They include the following:

- Internal flood
- Seismic margins analysis
- External events evaluations (such as high winds and tornados, external floods, and transportation accidents)
- Fire

The internal flood analysis identifies, analyzes, and quantifies the core damage risk contribution as a result of internal flooding during at-power and shutdown conditions. The analysis models potential flood vulnerabilities in conjunction with random failures modeled as part of the internal events PRA.

The seismic margins analysis identifies potential vulnerabilities and demonstrates seismic margin beyond the safe shutdown earthquake. The capacity of those components required to bring the plant to a safe, stable shutdown is evaluated.

17.4.7.1.4 Expert Panel

Meetings were held among Systems Engineering, PRA, and Reliability Engineering to perform the final selection of SSCs that should be included in the D-RAP. As shown in Figure 17.4-1, industry-wide information sources and engineering judgment were employed in considering the addition of SSCs to the D-RAP.

17.4.7.1.5 SSCs to be Included in D-RAP

Table 17.4-1 lists the non-site-specific SSCs included in the D-RAP. In Figure 17.4-1, this list is denoted as "Risk-significant items (non-site-specific)." For each item listed in the "SSC" column, there is a corresponding "Rationale" given. Items whose values exceed the thresholds for RAW or RRW are included and noted as such. Other SSCs are included based upon their significance to Level 2 analysis, external event analyses, or seismic margin analysis. Additional items are included based upon an expert panel review. The "Insights and Assumptions" column provides additional insights into the selection process.

The use of Fussel-Vesely worth resulted in no SSC selections.

17.4.7.2 Not Used

17.4.7.2.1 Not Used

17.4.7.3 Not Used

17.4.7.4 D-RAP Implementation

The following is an example of a system that was reviewed and modified under the D-RAP. The design and analytical results presented here are intended as an example.

The automatic depressurization system, which is part of the reactor coolant system, acts in conjunction with the passive core cooling system to mitigate design basis accidents. The automatic depressurization system valves are discussed in Subsection 5.4.6.

An earlier AP600 automatic depressurization system design contained four depressurization stages, with motor-operated valves in all stages. Preliminary PRA analysis established that fourth stage failure, in certain combination with failures of other stages, was a major contributor to core damage frequency. Thus, it was concluded that the fourth stage valves should be diverse in design from the valves in other stages to reduce common cause failure.

As a result of joint meetings among the AP600 PRA, Design, and staff organizations to discuss core melt frequency improvements, the fourth stage automatic depressurization system was changed from a motor-operated valve to a squib (explosively actuated) valve. The new configuration of the system is shown in the reactor coolant system P&ID (Figure 5.1-5). An example of the analytical results that reflect this change is provided in Table 17.4-2. This design feature is included in the AP1000 design to maintain the core melt frequency improvements included in the AP600 design.

As part of the evaluation of the squib valves, a failure modes and effects analysis (FMEA) was prepared to identify subcomponent failures and critical items that could lead to hazardous or abnormal conditions of the automatic depressurization system and the plant. The identification of failure modes facilitated the development of recommended maintenance and in-service testing activities to maximize valve reliability.

The squib valve is a completely static electromechanical assembly. Prior to activation, there are no moving parts. No powered components are needed to hold a stem seat or globe in place by torque, solenoid coils, or friction. The explosive actuator is a simple, passive device that is triggered by an applied voltage.

Because the automatic depressurization system fourth stage valves perform safety-related functions, they will be subject to in-service testing to verify that they are ready to function in an accident. Subsection 3.9.6 includes in-service testing requirements for these valves.

Example FMEA results for the fourth stage squib valves and the second and third stage motor-operated valves are included in Table 6.3-3. Table 3.9-16 provides testing recommendations for the second and third stage valves.

17.4.8 Glossary of Terms

D-RAP	Design Reliability Assurance Program – performed as part of the AP1000 design effort to assure that the reliability assumptions of the PRA remain valid throughout the plant operating lifetime.
FVW	Fussel-Vesely Worth
MR	Maintenance Rule
OPRAAs	Operational Phase Reliability Assurance Activities
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
Risk-significant	Any SSC determined in the PRA or by risk-significance analysis (e.g., Level 2 PRA and shutdown risk analysis) to be a major contributor to overall plant risk
RRW	Risk Reduction Worth
RTNSS	Regulatory Treatment of Nonsafety Systems
SSC	Structures, Systems, and Components

17.5 Quality Assurance Program Description

The Quality Assurance Program in place during the design, construction, and operations phases is described in the QAPD, which is maintained as a separate document. This QAPD is incorporated by reference (see Table 1.6-201). This QAPD is based on NEI 06-14A, “Quality Assurance Program Description” (Reference 207).

Conformance statements for QA-related Regulatory Guides (including Regulatory Guides 1.28, 1.30, 1.33, 1.38, 1.39, 1.94, and 1.116) are provided in Appendix 1A. While many Regulatory Guide positions can be identified as applicable to the scope of work identified and addressed by the DCD and others can be identified as applicable to the scope of work identified and addressed by the COLA, some QA guidance related positions could be accomplished by either scope of work and thus be addressed in either the DCD or the COLA. These positions are primarily dependent on who performs the work. The DCD conformance statement indicates an exception to apply NQA-1. The COLA identifies an exception to apply NQA-1. Per Section 17.3, WEC work performed up to March 15, 2007 applied a 1991 version of the standard. A 1994 version of the standard is applied for work performed after that date by WEC. If the work is performed under the applicant’s COL program, the 1994 version of NQA-1 identified in the COLA QAPD is applied. Thus, DCD scope (identified in Appendix 1A) and “remaining scope” differentiate the application of the guidance identified in these Regulatory Guides.

The QAPD is the SCE&G VCSNS Units 2 and 3 Quality Assurance Program Description.

Table 13.4-201 provides milestones for operational quality assurance program implementation.

The Quality Assurance Program in place prior to implementation of the QAPD is described in Section 17.1.

17.6 Maintenance Rule Program

This section incorporates by reference NEI 07-02A, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52” (Reference 208). See Table 1.6-201.

Table 13.4-201 provides milestones for maintenance rule program implementation.

The Maintenance Rule (MR) Program provides assurance that structures, systems, and components (SSCs) within the scope of the program remain reliable and capable of fulfilling their intended functions and provides processes for assessing and managing potential increases in risk that might result from proposed maintenance activities. The MR Program meets the requirements of 10 CFR 50.65 (Reference 209).

17.6.1 Maintenance Rule Program Description

The MR program follows the guidance in NUMARC 93-01 (Reference 210), as endorsed and modified by Regulatory Guide 1.160, (Reference 211) and revised Section 11.0 of NUMARC 93-01 (Reference 212), as endorsed and modified by Regulatory Guide 1.182 (Reference 213), without any exceptions that could materially and negatively affect the effectiveness of the program. The principal functions of the program are described in the following subsections.

The MR program includes appropriate control of procedures, documents, computer software and data, as applicable.

17.6.1.1 Maintenance Rule Scoping per 10 CFR 50.65(b)

17.6.1.1.a The SSCs within the scope of the MR program include safety-related SSCs and certain non-safety-related SSCs, as determined using a MR scoping procedure. The scoping procedure addresses:

- Safety-related SSCs.
- Non-safety-related SSCs that mitigate accidents or transients.
- Non-safety-related SSCs that are used in Emergency Operating Procedures, where ‘used’ means directly used to mitigate the accident or transient via explicit reference in the EOP or used in steps of procedures referenced by the EOP. Additionally, SSCs explicitly referenced in back-up or lower-tier methods in the EOPs and provide reasonable assurance of mitigation success, or whose use is implied in an EOP and essential to the completion of an EOP step, are considered within scope of the Maintenance Rule.
- Non-safety-related SSCs whose failure prevents safety-related SSCs from fulfilling their safety-related functions.
- Non-safety-related SSCs whose failure causes scrams or actuates safety systems.

The SSCs within the scope of the MR program are evaluated against performance criteria to determine which SSCs will have goals established and monitoring activities performed in accordance with 10 CFR 50.65(a)(1).

17.6.1.1.b Safety significance classifications and bases of in-scope SSCs, e.g., high safety significance (HSS) or low safety significance (LSS), are determined using processes consistent with Subsection 9.3.1 of NUMARC 93-01. They include determination of risk significance criteria and appropriate consideration of operating experience, generic failure data, component reliability information, probabilistic risk assessment (PRA) insights, and the recommendations of an expert panel. All SSCs identified as risk-significant via the Reliability Assurance Program for the design phase (DRAP – see Section 17.4) are included within the initial MR scope as HSS SSCs. This includes risk-significant SSCs identified as part of the design certification phase or follow-on COL applicant/holder phases of DRAP.

17.6.1.1.c The expert panel is established in accordance with NUMARC 93-01 prior to fuel load authorization and utilizes operating, maintenance and systems expertise, PRA insights, and other applicable information to update and maintain the MR scope and SSC classifications.

17.6.1.2 Monitoring and Corrective Action per 10 CFR 50.65(a)(1)

SSCs within the scope of the MR are initially classified as (a)(2) (ref. Subsection 17.6.1.3), except where it is determined that an SSC should be initially classified as (a)(1), e.g., an SSC that fails during start-up testing.

SSCs that do not meet performance criteria established for (a)(2) monitoring (ref. Subsection 17.6.1.3) are evaluated for (a)(1) classification in accordance with MR program procedures, with recommended corrective actions identified as appropriate. Necessary corrective actions are implemented in accordance with the site Corrective Action Program. The MR expert panel reviews whether SSCs are to be classified as (a)(1). Monitoring goals are established for SSCs classified as (a)(1), as appropriate, commensurate with the SSCs' safety significance, and considering applicable industry operating experience, with the objective of providing reasonable assurance that the SSC is proceeding to acceptable performance levels and that the corrective actions taken were effective.

For SSCs that do not meet established (a)(1) monitoring goals following corrective actions initially identified and implemented, appropriate additional corrective actions are taken.

17.6.1.3 Preventive Maintenance per 10 CFR 50.65(a)(2)

Monitoring as specified in 10 CFR 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance (PM), such that the SSC remains capable of performing its intended function.

The MR program includes procedures for managing SSC performance in accordance with 10 CFR 50.65(a)(2) requirements during plant operation consistent with NUMARC 93-01. To monitor the effectiveness of the maintenance performed on the various SSCs, performance criteria are established at the plant, system, train, or component level commensurate with safety, risk significance and SSC function. SSC performance criteria (e.g., failure rate, unavailability or condition-based) are chosen that are reasonable, measurable, and technically appropriate for the purpose of timely identification of degraded SSC performance or condition. For risk-significant SSCs identified via DRAP, performance criteria are consistent with the reliability and availability assumptions used in the PRA.

When a performance criterion is not met, the SSC is evaluated for (a)(1) classification in accordance with MR program procedures, including review by the Expert Panel. Should the Expert Panel conclude that the SSC should not be classified as (a)(1), or that no (a)(1) monitoring goals need be established, a technical justification establishing the appropriateness of continued management of SSC performance under (a)(2) is documented and maintained.

SSCs that provide little or no contribution to system safety function or can be allowed to run to failure due to an acceptable risk may be categorized in a "run-to-failure" status (i.e., perform corrective maintenance rather than preventive maintenance) consistent with NUMARC 93-01.

Preventive maintenance is subject to risk assessment and management per 10 CFR 50.65(a)(4) (ref. Subsection 17.6.1.5).

17.6.1.4 Periodic Evaluation of Monitoring and Preventive Maintenance per 10 CFR 50.65(a)(3)

The MR program includes procedures for the periodic evaluation of the performance and condition monitoring activities and associated goals and preventive maintenance activities in accordance with 50.65(a)(3). The following considerations are included:

- how procedures govern the scheduling and timely performance of (a)(3) evaluations.
- documenting, reviewing and approving evaluations, providing and implementing results.
- review of 50.65(a)(1) goals and 50.65(a)(2) performance criteria, condition monitoring criteria, SSC performance and condition history and effectiveness of corrective action
- making adjustments to achieve or restore balance between reliability and availability.
- industry operating experience.

17.6.1.5 Risk Assessment and Risk Management per 10 CFR 50.65(a)(4)

The MR program includes procedures for maintenance risk assessment and management in accordance with 10 CFR 50.65(a)(4), employing the methods described in NUMARC 93-01, Section 11 (Reference 212). The risk from maintenance activities is both assessed (i.e., using a risk-informed process to evaluate the overall contribution to risk of the planned maintenance activities) and managed (i.e., providing plant personnel with proper awareness of the risk, and taking actions as appropriate to control the risk).

The MR program and procedures reflect, as appropriate, consideration of issues associated with grid/offsite power reliability as identified in NRC Generic Letter 2006-02, items 5 and 6.

17.6.2 Maintenance Rule Training and Qualification

The MR program is supported by appropriate training and qualification for designated personnel. Training is commensurate with MR responsibilities, including MR program administration, the expert panel process, operations, engineering, maintenance, licensing, and plant management, as appropriate. Maintenance Rule Program training and qualification materials are based on regulatory requirements and guidance, and training records are maintained in accordance with plant procedures.

17.6.3 Maintenance Rule Program Relationship With Reliability Assurance Activities

Reliability during the operations phase is assured through the implementation of operational programs, i.e., the MR program, the Quality Assurance Program, inservice inspection and testing programs, the Technical Specifications surveillance test program, and maintenance programs.

Descriptions of the programs are provided in the following chapters/sections:

The maintenance rule program (Section 17.6)

The quality assurance program (Section 17.5)

Inservice inspection program (Sections 5.2 and 6.6)

Inservice testing program (Section 3.9)

The technical specifications surveillance test program (Chapter 16)

17.6.4 Maintenance Rule Program Relationship With Industry Operating Experience Activities

Industry Operating Experience (IOE) comprises information from a variety of sources that is applicable and available to the nuclear industry with the intent of minimizing, through shared experiences, adverse plant conditions or situations. Sources of IOE include information programs organized by the reactor vendor, safety-related equipment suppliers, the NRC, the Institute of Nuclear Power Operations (INPO) and the Electric Power Research Institute (EPRI).

IOE is reviewed for plant-specific applicability and, where appropriate, is applied in various elements of the MR program and procedures, including scoping, performance/condition criteria development, monitoring, goal-setting, corrective action, training, program assessment, and maintenance and procurement activities. The specific steps for employing IOE in the various MR program areas are contained in program procedures.

Condition monitoring of underground or inaccessible cables is incorporated into the maintenance rule program. The cable condition monitoring program incorporates lessons learned from industry operating experience, addresses regulatory guidance, and utilizes information from detailed design and procurement documents to determine the appropriate inspections, tests and monitoring criteria for underground and inaccessible cables within the scope of the maintenance rule (i.e., 10 CFR 50.65). The program takes into consideration Generic Letter 2007-01.

17.6.5 Maintenance Rule Program Implementation

MR Program documents will be developed and maintained, and the MR program will be implemented by the time that initial fuel loading has been authorized.

17.7 Combined License Information Items

17.7.1 The design phase Quality Assurance program is addressed in Sections 17.1 and 17.5.

17.7.2 The Quality Assurance program for procurement, fabrication, installation, construction and testing of structures, systems and components in the facility, including provisions for seismic Category II structures, systems, and components, is addressed in Section 17.5.

17.7.3 The PRA importance measures, the expert panel process, and other deterministic methods to determine the site-specific list of SSCs under the scope of RAP are addressed in APP-GW-GLR-117 (Reference 11).

17.7.4 The Quality Assurance program for operations is addressed in Section 17.5.

17.7.5 The activities represented in Figure 17.4-1 as "Plant Maintenance Program" include the tasks necessary to maintain the reliability of risk-significant SSCs as addressed in APP-GW-GLR-117 (Reference 11). Reference 8 contains examples of cost-effective maintenance enhancements, such as condition monitoring and shifting time-directed maintenance to condition-direction maintenance.

17.7.6 The Maintenance Rule (10 CFR 50.65) activities that prescribe SSC performance-related goals during plant operation are addressed in APP-GW-GLR-117 (Reference 11).

17.7.7 The D-RAP activities are addressed in APP-GW-GLR-117 (Reference 11), and include:

- Reliability data base — Historical data available on equipment performance. The compilation and reduction of this data provides the plant with source of component reliability information.
- Surveillance and testing — In addition to maintaining the performance of the components necessary for plant operation, surveillance and testing provides a high degree of reliability for the safety-related SSCs.
- Maintenance plan — This plan describes the nature and frequency of maintenance activities to be performed on plant equipment. The plan includes the selected SSCs identified in the D-RAP.

17.7.8 The integration of the objectives of the OPRAAs into the Quality Assurance Program developed to implement 10 CFR 50, Appendix B, including failures of non-safety-related, risk-significant SSCs that result from design and operational errors in accordance with SECY-95-132, Item E, is addressed in Section 17.5.

17.8 References

1. "Energy Systems Business Unit – Quality Management System," Revision 2.

2. WCAP-8370, Revision 12a, "Energy Systems Business Unit - Power Generation Business Unit Quality Assurance Plan."

3. WCAP-8370/7800, Revision 11A/7A, "Energy Systems Business Unit - Nuclear Fuel Business Unit Quality Assurance Plan."

4. WCAP-12600, Revision 4, "AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan," January 1998.

5. APP-GW-GL-022, Revision 8, AP1000 Probabilistic Risk Assessment.

6. Not used.

7. NRC/DCP0669, "Criteria for Establishing Risk Significant Structures, Systems, and Components (SSCs) to be Considered for the AP600 Reliability Assurance Program," January 16, 1997.

8. Lofgren, E. V., Cooper, et al., "A Process for Risk-Focused Maintenance," NUREG/CR-5695, March 1991.

9. Westinghouse Electric Company Quality Management System (QMS), Revision 5, dated October 1, 2002.

10. NEI 07-02, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52.”

11. APP-GW-GLR-117, “Incorporation of the Maintenance Rule,” Westinghouse Electric Company LLC.

12. SECY 95-132, “Policy and Technical Issue With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY 94-084).”

201. NuStart Energy, LLC., “NuStart Energy Project Instruction – Quality Assurance Plan,” PI-009.

202. “NRC Audit Report for the South Carolina Electric and Gas (SCE&G) VC Summer Nuclear Plant Combined License Application Review,” J. W. Chung to A. M. Monroe, November 16, 2007 (ML073100387).

203. Not used.

204. South Carolina Electric & Gas, “New Nuclear Deployment Quality Assurance Plan,” Rev. 2, June 4, 2009.

205. Bechtel Power Corporation, “Nuclear Quality Assurance Manual,” Rev. 4, 11/01/02.

206. South Carolina Electric & Gas, V.C. Summer Nuclear Station Unit 1 “Operational Quality Assurance Plan,” Rev. 27, February 21, 2005.

207. Nuclear Energy Institute, Technical Report NEI 06-14A, “Quality Assurance Program Description,” Revision 7, July 2009.

208. Nuclear Energy Institute, “Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52,” NEI 07-02A, Revision 0, March 2008 (ML080910149).

209. 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants.”

210. Nuclear Management and Resources Council, Inc., "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, Rev. 2, April 1996.

211. Regulatory Guide 1.160, Rev. 2, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

212. Nuclear Management and Resources Council, Inc., “Assessment of Risk Resulting from Performance of Maintenance Activities,” NUMARC 93-01, Section 11, February 22, 2000.

213. Regulatory Guide 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

Table 17-1 (Sheet 1 of 3)
Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

The following outlines the quality assurance program requirements for suppliers of systems, structures, or components to which the requirements for investment protection short-term availability controls apply. 1. Organization The normal line organization may verify compliance with the requirements of this table. A separate or dedicated quality assurance organization is not required.
2. Quality Assurance Program It is expected that the existing body of supplier's procedures or practices will describe the quality controls applied to the subject equipment. A new or separate QA program is not required.
3. Design Control Measures shall be established to ensure that contractually established design requirements are included in the design. Applicable design inputs shall be included or correctly translated into design documents, and deviations therefrom shall be controlled. Normal supervisory review of the designer's work is an adequate control measure.
4. Procurement Document Control Applicable design bases and other requirements necessary to assure component performance, including design requirements, shall be included or referenced in documents for procurement of items and services, and deviations therefrom shall be controlled.
5. Instructions, Procedures, and Drawings Activities affecting quality shall be performed in accordance with documented instructions, procedures, or drawings of a type appropriate to the circumstances. This may include such things as written instructions, plant procedures, cautionary notes on drawings, and special instructions on work orders. Any methodology which provides the appropriate degree of guidance to personnel performing activities important to the component functional performance will satisfy this requirement.
6. Document Control The issuance and change of documents that specify quality requirements or prescribe activities affecting quality shall be controlled to assure that correct documents are employed.
7. Control of Purchased Items and Services Measures are to be established to ensure that all purchased items and services conform to appropriate procurement documents.

Table 17-1 (Sheet 2 of 3)
Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

8. Identification and Control of Purchased Items Measures shall be established where necessary, to identify purchased items and preserve their investment protection important functional performance capability. Examples of circumstances requiring such control include the storage of environmentally sensitive equipment or material, and the storage of equipment or material that has a limited shelf-life.
9. Control of Special Processes Measures shall be established to control special processes, including welding, heat treating, and non-destructive testing. Applicable codes, standards, specifications, criteria, and other special requirements may serve as the basis of these controls.
10. Inspection Inspections shall be performed where necessary to verify conformance of an item or activity to specified requirements, or to verify that activities are being satisfactory accomplished. Inspections need not be performed by personnel who are independent of the line organization. However, inspections, where necessary, shall be performed by knowledgeable personnel.
11. Test Control Measures shall be established, as appropriate, to test equipment prior to installation to demonstrate conformance with design requirements. Tests shall be performed in accordance with test procedures. Test results shall be recorded and evaluated to ensure that test requirements have been met.
12. Control of Measuring and Test Equipment Measures shall be established to control, calibrate, and adjust measuring and test equipment at specific intervals.
13. Handling, Storage, and Shipping Handling, storage, cleaning, packaging, shipping, and preservation of items shall be controlled to prevent damage or loss and to minimize deterioration.
14. Inspection, Test, and Operating Status Measures shall be established to identify items that have satisfactory passed required tests and inspections, and to indicate status of inspection, test, and operability as appropriate.
15. Control of Nonconforming Items Items that do not conform to specified requirements shall be identified and controlled to prevent inadvertent installation or use.

Table 17-1 (Sheet 3 of 3) Quality Assurance Program Requirements for Systems, Structures, And Components Important to Investment Protection

16. Corrective Action Measures shall be established to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are properly identified, reported, and corrected.
17. Records Records shall be prepared and maintained to furnish evidence that the above requirements for design, procurement, document control, inspection, and test activities have been met.
18. Audits Audits which are independent of line management are not required, if line management periodically reviews and documents the adequacy of the suppliers process and takes any necessary corrective action. Line management is responsible for determining whether reviews conducted by line management or audits conducted by and organization independent of line management are appropriate. If performed, audits shall be conducted and documents to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities.

Table 17.4-1 (Sheet 1 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Component Cooling Water (CCS)		
Component Cooling Water Pumps (CCS-MP-01A/B)	EP	These pumps provide cooling of the normal residual heat removal system (RNS) and the spent fuel pool heat exchanger. Cooling the RNS heat exchanger is important to investment protection during shutdown reduced-inventory conditions. CCS valve realignment is not required for reduced-inventory conditions.
System: Containment System (CNS)		
Containment Vessel (CNS-MV-01)	EP, L2	The containment vessel provides a barrier to steam and radioactivity released to the atmosphere following accidents.
Hydrogen Igniters (VLS-EH-1 through -64)	RAW/CCF, L2, Regulations	The hydrogen igniters provide a means to control H_2 concentration in the containment atmosphere, consistent with the hydrogen control requirements of 10 CFR 50.34f.
System: Chemical and Volume Control System (CVS)		
Makeup Pumps (CVS-MP-01A/B)	EP	These pumps provide makeup to the RCS to accommodate leaks and to provide negative reactivity for shutdowns, steam line breaks, and ATWS.
Makeup Pump Suction and Discharge Check Valves (CVS-PL-V113, -V160A/B)	EP	These CVS check valves are normally closed and have to open to allow makeup pump operation.
Letdown Isolation Valves (CVS-PL-V045, -V047)	RAW	The CVS letdown isolation valves automatically close to prevent excessive reactor coolant letdown and provide containment isolation. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Diverse Actuation System (DAS)		
DAS Processor Cabinets and Control Panel (used to provide automatic and manual actuation) (DAS-JD-001, -002, -003, -004, OCS-JC-020)	RAW	The DAS is diverse from the PMS and provides automatic and manual actuation of selected plant features including control rod insertion, turbine trip, passive residual heat removal (PRHR) heat exchanger actuation, core makeup tank actuation, isolation of critical containment lines, and passive containment cooling system (PCS) actuation.
Annex Building UPS Distribution Panels (EDS1-EA-1, EDS1-EA-14, EDS2-EA-1, EDS2-EA-14)	RAW	These panels distribute power to the DAS equipment.

Table 17.4-1 (Sheet 2 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
Rod Drive MG Sets (Field Breakers) (PLS-MG-01A/B)	RAW	These breakers open on a DAS reactor trip signal demand to de-energize the control rod MG sets and allow the rods to drop.
System: Main ac Power System (ECS)		
Reactor Coolant Pump Switchgear (ECS-ES-31, -32, -41, -42, -51, -52, -61, -62)	RAW/CCF	These breakers open automatically to allow core makeup tank operation.
Ancillary Diesel Generators (ECS-MS-01, -02)	EP	For post-72 hour actions, these generators are available to provide power for Class 1E monitoring, MCR lighting and for refilling the PCS water storage tank and spent fuel pool.
6900 Vac Buses (ECS-ES-1, -2)	RAW	These are ac power buses fed by the onsite DGs and offsite power.
System: Main and Startup Feedwater System (FWS)		
Startup Feedwater Pumps (FWS-MP-03A/B)	EP	The startup feedwater system pumps provide feedwater to the steam generator. This capability provides an alternate core cooling mechanism to the PRHR heat exchangers for non-loss-of-coolant-accidents or steam generator tube ruptures.
System: General I&C(4)		
Low Pressure/DP Sensors – IRWST level sensors (PXS-045, -046, -047, -048)	RAW/CCF	The in-containment refueling water storage tank (IRWST) level sensors support PMS functions. They are used in automatic actuation, and they provide indications to the operator. IRWST level supports IRWST recirculation actions.
High Pressure/DP Sensors – RCS Hot Leg Level (RCS-160A/B) – Pressurizer Pressure (RCS-191A/B/C/D) – Pressurizer Level (RCS-195A/B/C/D) – SG Narrow-Range Level (SGS-001, -002, -003, -004, -005, -006, -007, -008) – SG Wide-Range Level (SGS-011, -012, -013, -014, -015, -016, -017, -018)	RAW/CCF/EP	The following sensors are included in this group. These sensors support PMS and PLS functions. They are used in reactor trip and ESF functions, and provide indications to the operator. Main feedwater flow sensors support startup feedwater actuation and startup feedwater flow sensors support PRHR actuation. The hot leg level sensors automatically actuate the IRWST injection and automatic depressurization system (ADS) valves during shutdown conditions.

Table 17.4-1 (Sheet 3 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
– Main Steam Line Pressure (SGS-030, -031, -032, -033, -034, -035, -036, -037) – Main Feedwater Wide-Range Flow (FWS-050B/D/F, -051B/D/F) – Startup Feedwater Flow (SGS-055A/B, -056A/B)		
CMT Level Sensors (PXS-011A/B/C/D, -012A/B/C/D, -013A/B/C/D, -014A/B/C/D)	RAW/CCF	These level sensors provide input for automatic actuation of the ADS. They also provide indications to the operator.
System: Class 1E DC Power and Uninterruptible Power System (IDS)		
250 Vdc 24-hour Buses, Batteries, Inverters, and Chargers (IDSA-DB-1A/B, IDSB-DB-1A/B, IDSC-DB-1A/B, IDSD-DB-1A/B, IDSA-DU-1, IDSB-DU-1, IDSC-DU-1, IDSD-DU-1, IDSA-DC-1, IDSB-DC-1, IDSC-DC-1, IDSD-DC-1, IDSA-DS-1, IDSB-DS-1, IDSC-DS-1, IDSD-DS-1)	RAW/CCF	The batteries provide power for the PMS and safety-related valves. The chargers are the preferred source of power for Class 1E dc loads and are the source of charging for the batteries. The inverters provide uninterruptible ac power to the I&C system. The buses distribute power to the Class 1E dc loads.
250 Vdc and 120 Vac Distribution Panels (IDSA-DD-1, -EA-1/2, IDSB-DD-1, -EA-1/2/3, IDSC-DD-1, -EA-1/2/3, IDSD-DD-1, -EA-1/2)	RAW	These panels distribute power to components in the plant that require 1E power support and for the PMS.
Fused Transfer Switch Boxes (IDSA-DF-1, IDSB-DF-1/-2, IDSC-DF-1/-2, IDSD-DF-1)	RAW	The fused disconnect switches connect the different levels of Class 1E distribution panels.

Table 17.4-1 (Sheet 4 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
250 Vdc Motor Control Centers (IDSA-DK-1, IDSB-DK-1, IDSC-DK-1, IDSD-DK-1)	EP	These buses provide power for the PMS and safety-related valve operation.
System: Passive Containment Cooling System (PCS)		
Recirculation Pumps (PCS-MP-01A/B)	EP	These pumps provide the motive force to refill the PCS water storage tank during post-72 hour support actions.
PCCWST Drain Isolation Valves (PCS-PL-V001A/B/C)	EP, L2	These valves (two AOVs and one MOV) open automatically to drain water from a water storage tank onto the outside surface of the containment shell. This water provides evaporative cooling of the containment shell following accidents.
System: Plant Control System (PLS)		
PLS Actuation Hardware (Control functions listed in Note 5)	RAW/CCF	This common cause failure event is assumed to disable all logic outputs from the PLS associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.
PLS Actuation Software (Control functions listed in Note 5)	RAW/CCF	This common cause failure event is assumed to disable the software in the PLS associated with CVS reactor makeup, RNS reactor injection, spent fuel cooling, component cooling of RNS SFS heat exchangers, service water cooling of CCS heat exchangers, standby diesel generators, and hydrogen igniters.
System: Protection and Safety Monitoring System (PMS)		
PMS Actuation Software	RAW/CCF	The PMS software provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.
PMS Actuation Hardware	RAW/CCF	The PMS hardware provides the automatic reactor trip and ESF actuation functions listed in Tables 7.2-2 and 7.3-1.
Main Control Room (MCR) 1E Displays and System Level Controls (OCS-JC-010, -011)	RAW/CCF	This includes the Class 1E PMS (QDPS) displays and controls. These displays and system level controls provide important plant indications to allow the operator to monitor and control the plant during accidents.
Reactor Trip Switchgear (PMS-JD-RTS A01/02, B01/02, C01/02, D01/02)	RAW/CCF	These breakers open automatically to allow insertion of the control rods.

Table 17.4-1 (Sheet 5 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
System: Passive Core Cooling System (PXS)		
IRWST Vents (PXS-MT-03)	RAW/CCF	The IRWST vents provide a pathway to vent steam from the tank into the containment. The IRWST vents also have a severe accident function to prevent the formation of standing hydrogen flames close to the containment walls. This function is accomplished by designing the vents located further from the containment walls to open with less IRWST internal pressure than the other vents.
IRWST Screens (PXS-MY-Y01A/B/C)	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These screens are located inside the IRWST and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
Containment Recirculation Screens (PXS-MY-Y02A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. The screens are located in the containment and prevent large particles from being injected into the RCS. They are designed so that they will not become obstructed.
CMT Discharge Isolation Valves (PXS-PL-V014A/B, PXS-PL-V015A/B)	RAW/CCF	These air-operated valves automatically open to allow core makeup tank injection.
CMT Discharge Check Valves (PXS-PL-V016A/B, PXS-PL-V017A/B)	RAW/CCF	These check valves are normally open. They close during rapid accumulator injection.
Accumulator Discharge Check Valves (PXS-PL-V028A/B, -V029A/B)	RAW/CCF	These check valves open when the RCS pressure drops below the accumulator pressure to allow accumulator injection.
PRHR Heat Exchanger Control Valves (PXS-PL-V108A/B)	RAW/CCF	The PRHR heat exchangers provide core cooling following non-LOCAs, steam generator tube ruptures, and anticipated transients without scram. The air-operated valves automatically open to initiate PRHR heat exchanger operation.

Table 17.4-1 (Sheet 6 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[1]	Rationale[2]	Insights and Assumptions
Containment Recirculation Squib Valves (PXS-PL-V118A/B, PXS-PL-V120A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow containment recirculation when the IRWST level is reduced to about the same level as the containment level. These squib valves can also allow long-term core cooling to be provided by the RNS pumps. These squib valves can provide a rapid flooding of the containment to support in-vessel retention during a severe accident.
IRWST Injection Check Valves (PXS-PL-V122A/B, -V124A/B)	RAW/CCF	The containment recirculation lines provide long-term core cooling following a LOCA. These check valves open when the IRWST level is reduced to approximately the same level as the containment level.
IRWST Injection Squib Valves (PXS-PL-V123A/B, -V125A/B)	RAW/CCF	The IRWST injection lines provide long-term core cooling following a LOCA. These squib valves open automatically to allow injection when the RCS pressure is reduced to below the IRWST injection head.
IRWST Gutter Bypass Isolation Valves (PXS-PL-V130A/B)	RAW/CCF	These valves direct water collected in the IRWST gutter to the IRWST. This capability extends PRHR heat exchanger operation.
System: Reactor Coolant System (RCS)		
ADS Stage 1/2/3 Valves (MOV) (RCS-PL-V001A/B, -V002A/B, -V003A/B, -V011A/B, -V012A/B, -V013A/B)	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
ADS Stage 4 Valves (Squib) (RCS-PL-V004A/B/C/D)	RAW/CCF	The ADS provides a controlled depressurization of the RCS following LOCAs to allow core cooling from the accumulator, IRWST injection, and containment recirculation. The ADS provides "bleed" capability for feed/bleed cooling of the core. The ADS also provides depressurization of the RCS to prevent a high-pressure core melt sequence.
Pressurizer Safety Valves (RCS-PL-V005A/B)	RRW	These valves provide overpressure protection of the RCS.

**Table 17.4-1 (Sheet 7 of 8)
Risk-Significant SSCs Within the Scope of D-RAP**

System, Structure, or Component (SSC)(1)	Rationale(2)	Insights and Assumptions
Reactor Vessel Insulation Water Inlet and Steam Vent Devices (RCS-MN-01)	EP	These devices provide an engineered flow path to promote in-vessel retention of the core in a severe accident.
Reactor Cavity Doorway Damper	EP	This device provides a flow path to promote in-vessel retention of the core in a severe accident.
Fuel Assemblies (157 assemblies with tag numbers beginning with RXS-FA)	SMA	The nuclear fuel assembly includes the fuel pellets, fuel cladding, and associated support structures. This equipment, which provides a first barrier for release of radioactivity and allows for effective core cooling, had the least margin in the seismic margin analysis.
System: Normal Residual Heat Removal System (RNS)		
Residual Heat Removal Pumps (RNS-MP-01A/B)	RAW/CCF	These pumps provide shutdown cooling of the RCS. They also provide an alternate RCS lower pressure injection capability following actuation of the ADS. The operation of these pumps is important to investment protection during shutdown reduced-inventory conditions. RNS valve realignment is not required for reduced-inventory conditions.
RNS Motor-Operated Valves (RNS-PL-V011, -V022, -V023, -V055)	RRW	These MOVs align a flow path for nonsafety-related makeup to the RCS following ADS operation, initially from the cask loading pit and later from the containment.
RNS Stop Check Valves (RNS-PL-V015A/B), RNS Check Valves (RNS-PL-V017 A/B)	CCF/EP	These stop check valves and check valves are in the discharge of the RNS pumps. They prevent backflow from the RCS.
RNS Check Valves (RNS-PL-V007 A/B, -V013, -V056)	L2 RAW/EP	Check valves V007 A/B and V013 provide a flow path from the RNS pumps to the RCS. Failure of these valves to open will result in the loss of long-term cooling from the RNS. Check valve V056 provides a flow path from the cask loading pit to the RNS pump inlet.
System: Spent Fuel Cooling System (SFS)		
Spent Fuel Cooling Pumps (SFS-MP-01A/B)	EP	These pumps provide flow to the heat exchangers for removal of the design basis heat load.
System: Steam Generator System (SGS)		
Main Steam Safety Valves (SGS-PL-V030A/B, -V031A/B, -V032A/B, -V033A/B, -V034A/B, -V035A/B)	RRW	The steam generator main steam safety valves provide overpressure protection of the steam generator. They also provide core cooling by venting steam from the steam generator.

Table 17.4-1 (Sheet 8 of 8)
Risk-Significant SSCs Within the Scope of D-RAP

System, Structure, or Component (SSC)[(1)]	Rationale[(2)]	Insights and Assumptions
Main Steam and Feedwater Isolation Valves (SGS-PL-V040A/B, -V057A/B)	RAW/EP	The steam generator main steam and feedwater isolation valves provide isolation of the steam generator following secondary line breaks and steam generator tube rupture.
System: Service Water System (SWS)		
Service Water Pumps and Cooling Tower Fans (SWS-MP-01A/B, SWS-MA-01A/B)	EP	These pumps and fans provide cooling of the CCS heat exchanger which is important to investment protection during shutdown reduced-inventory conditions. Service water system valve realignment is not required for reduced-inventory conditions.
System: Nuclear Island Nonradioactive Ventilation System (VBS)		
VBS MCR and I&C Rooms B/C Ancillary Fans (VBS-MA-10A/B, -11, -12)	EP	For post-72 hour actions, these fans are available to provide cooling of the MCR and the two I&C rooms (B/C) that provide post-accident monitoring.
System: Containment Air Filtration System (VFS)		
VFS Containment Purge Isolation Valves (VFS-PL-V003, -V004, -V009, -V010)	RAW	The VFS containment purge isolation valves provide isolation of containment following an accident. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Chilled Water System (VWS)		
Air Cooled Chillers and Pumps (VWS-MS-02, -03, VWS-MP-02, -03)	EP	This VWS subsystem provides chilled cooling water to the CVS makeup pump room. The pumps and chillers are important components of the VWS.
System: Liquid Radwaste System (WLS)		
Sump Containment Isolation Valves (WLS-PL-V055, -V057)	RAW	The sump containment isolation valves provide isolation of containment following an accident. These containment isolation valves are important in limiting offsite releases following core melt accidents.
System: Onsite Standby Power System (ZOS)		
Onsite Diesel Generators (ZOS-MS-05A/B)	RAW/CCF	These diesel generators provide ac power to support operation of nonsafety-related equipment such as the startup feedwater pumps, CVS pumps, RNS pumps, CCS pumps, SWS pumps, and the PLS. Providing ac power to the RNS and the equipment necessary to support its operation is important to investment protection during reduced inventory conditions.
Engine Room Exhaust Fans (VZS-MY-V01A/B, -V02A/B)	RAW/CCF	These fans provide ventilation of the rooms containing the onsite diesel generators.

Notes:

1. Only includes equipment at the **component** level. Other parts of the SSC or support systems are not included unless specifically listed.
2. Definition of Rationale Terms:
   CCF = Common Cause Failure (for the SSCs whose inclusion rationale is RAW/CCF, the RAW is based on common cause failure of two or more of the specified SSCs.
   EP = Expert Panel
   RAW = Risk Achievement Worth
   RRW = Risk Reduction Worth
   SMA = Seismic Margin Analysis
3. Maintenance/surveillance recommendations for equipment are documented in each appropriate section.
4. This category captures instrumentation and control equipment common cause failures across systems.
5. The PLS provides control of the following functions:
   CVS Reactor Makeup
   RNS Reactor Injection from Cask Loading Pit
   Startup Feedwater from CST
   Spent Fuel Cooling
   Component Cooling of RNS and SFS Heat Exchangers
   Service Water Cooling of the CCS Heat Exchangers
   Onsite Diesel Generators
   Hydrogen Igniters

Table 17.4-2
Example of Risk-Significant Ranking of SSCs for the Automatic Depressurization System

Rank[(1)]	Event Code	Description
1	ED3MOD07	EDS3 EA1 distribution panel failure or unavailable due to testing and maintenance
2	AD4MOD07, AD4MOD08, AD4MOD09, AD4MOD10	Hardware failure of 2 of 4 automatic depressurization system Stage 4 squib valves
3	EC1BS001TM, ECBS012TM, EC1BS121TM, EC2BS002TM, EC2BS022TM, EC2BS221TM	Unavailability of bus ECS ES due to unscheduled maintenance
4	AD2MOD01, AD2MOD02, AD2MOD03, AD2MOD04	Hardware failure of 2 of 4 automatic depressurization system Stages 2 and 3 of lines 1 and 2 (includes motor-operated valves)
5	EC0MOD01	Main generator breaker ES01 fails to open
6	ED3MOD01	Fixed component fails: circuit breaker, inverter or static transfer switch
7	Z01MOD01, Z02MOD01	Diesel generator fails to start and run or breaker 102 fails to close
8	Z02DG001TM, Z02DG001TM	Standby diesel generator unavailable due to testing and maintenance

Note:
1. The ranking is in the order of decreasing risk achievement component importance.

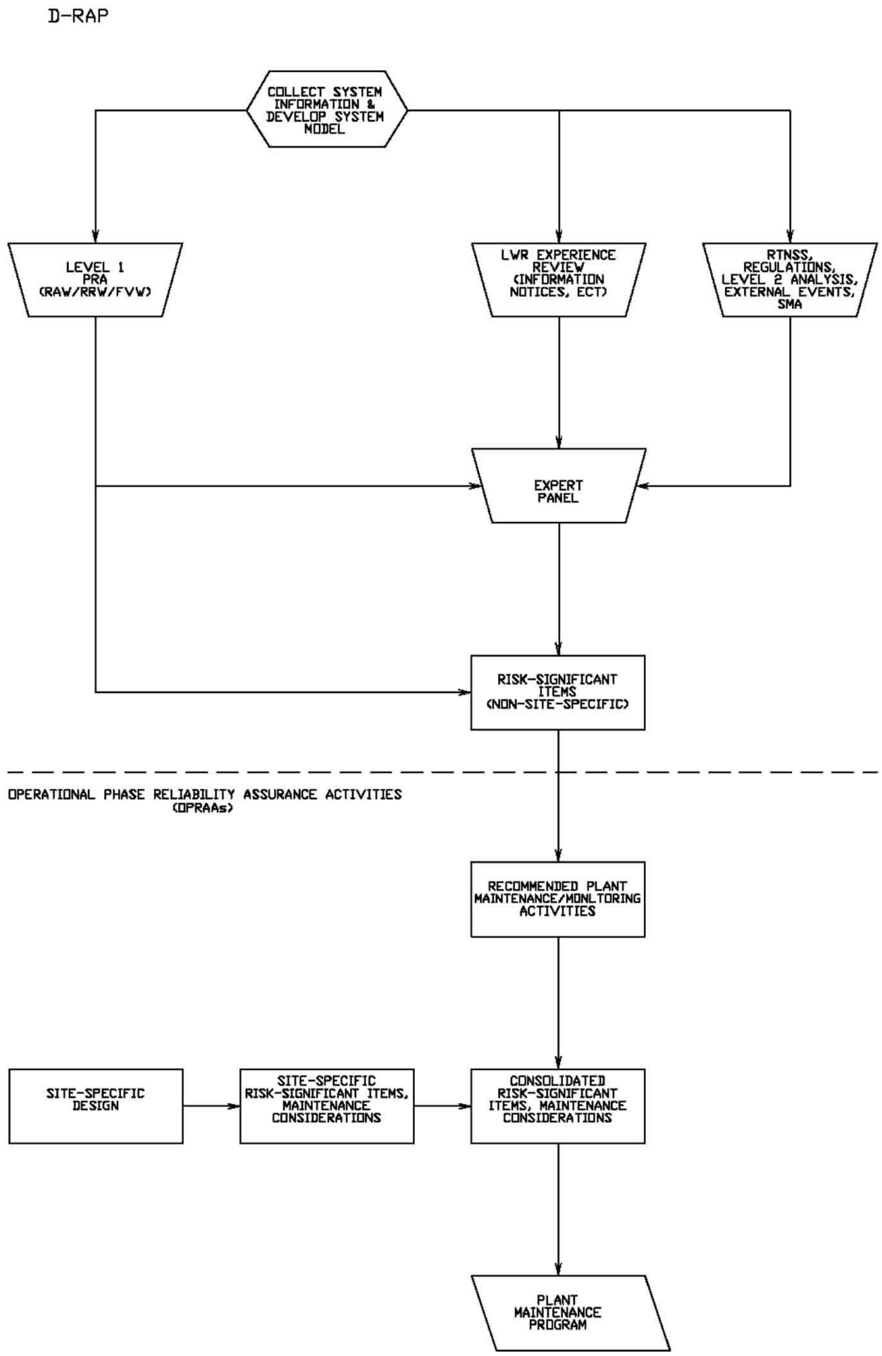

Figure 17.4-1
Design Reliability Assurance Program and Operational Phase Reliability Assurance Activities