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HARRIS UNIT 1

ADMINISTRATIVE PROCEDURE

NON-SAFETY RELATED

PLP-106

TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM

REVISION 069

REVISION SUMMARY

PRR 2152196

DESCRIPTION

This revision (PRR 2152196) updates PLP-106, TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM (Revision 069) to include changes for HNP Cycle 22 Core Reload. The major portion of this change is to move the Core Operating Limits Report (COLR) from PLP-106 to HNEI-0400-0012, Harris Unit 1 Cycle 22 Core Operating Limits Report (COLR).

PRR 2144284 has also been incorporated: RECOMMENDED CHANGE: Wording in PLP-106, Scope (section 2.3.d) states: This procedure is incorporated by reference into the FSAR and is therefore made a part of the FSAR, which is part of the Harris Current Licensing Basis (CLB). This procedure is subject to the update and reporting requirements of 10 CFR 50.71(e) and change controls of 10 CFR 50.59 and as such cannot be exempted from 50.59 screening. Recommend change is to replace "50.59 screening" with "50.59 review". REASON FOR CHANGE: This language is outdated by the current 50.59 process. Per AD-LS-ALL-0007, a 50.59 review is a more accurate description of the expectation to review procedure changes under 50.59 (the previous legacy progress procedure required screens for everything that was not exempted, whereas the current procedure requires an additional step, known as the AD process.

Description of Changes:

Cover Page: Updated title to "TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM" as the COLR is no longer represented in PLP 106. PRR 2188837

Page 5, Purpose 1.0.3: Modified to state "Attachment 9 instructs the user how to locate the Harris Unit 1 Core operating Limits Report (COLR)."

Page 6, Scope 2.0.2: Modified to state "The Harris Unit 1 Core Operating Limits Report (COLR) is located in a cycle specific Engineering Instruction (see Attachment 9 for details)."

Page 6, Scope 2.0.3.c: Changed verbiage from "50.59 screening" to "50.59 review." PRR 2144284

Page 8, Responsibilities 4.0.1-4.0.3: Deleted as the COLR is no longer represented in PLP-106.

Page 10, Section 5.1.9: Modified to state "See Attachment 9, Harris Unit 1 Core Operating Limits Report, for instruction in locating the cycle specific Engineering Instruction containing the COLR."

Page 11, Step 5.2.1.d: Deleted as the COLR is no longer represented in PLP 106.

Page 72, Att 9, Updated attachment 9 to help the user find the appropriate cycle-specific Engineering Instruction in Fusion that represents the HNP Unit 1 COLR.

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1.0 PURPOSE

1. This procedure identifies and controls the various equipment lists required by Shearon Harris Technical Specifications. These lists are comprised of plant equipment used to fulfill the respective LCO/Surveillance Requirements, as follows:

    - Reactor Trip System Instrumentation Response Times
    - Engineered Safety Features Actuation System Instrumentation Response Times
    - Reactor Vessel Material Surveillance Program
    - Snubber Surveillance Program
    - Containment Isolation Valves
    - Containment Penetration Conductor Overcurrent Protection Devices
    - Motor-Operated Valve Thermal Overload Protection Bypass
    - Instrument Uncertainties associated with Reactor Vessel Pressure - Temperature Limits and Low Temperature Overpressure Protection System
    - Reactor Trip System Instrumentation Setpoint As-Found/As-Left Tolerances

2. Additionally, the following equipment list is used to provide actions for components not covered by Technical Specifications, yet these components provide backup (not redundant) protective actions:

    - Safeguards Systems Isolation Valves in Closed Systems

3. Attachment 9, Harris Unit 1 Core Operating Limits Report, instructs the user how to locate the Harris Unit 1 Core Operating Limits Report (COLR).

4. This procedure is required to implement License Renewal (LR) commitments and requirements and supports the following LR Aging Management programs (AMPs) [7.3.3]:

    - Reactor Vessel Surveillance Program {7.1.2}

5. This procedure also contains the requirements of the Diesel Fuel Oil Testing Program.

2.0 SCOPE

1. The equipment lists, figures, and surveillance programs attached to this procedure were generated to fulfill the requirements of various Technical Specifications for the Shearon Harris Nuclear Power Plant and to ensure prudent actions are taken for other equipment for which no Technical Specification provides explicit actions.

   Historically, those items comprising the Technical Specification Equipment List Program were maintained directly in the Technical Specifications. By maintaining these items independent of the Technical Specifications but still fulfilling the regulatory requirements, they are maintained more current and accurate with respect to the actual plant configuration without the delay typically encountered with Operating License/Technical Specification amendments.

2. The Harris Unit 1 Core Operating Limits Report (COLR) is located in a cycle-specific Engineering Instruction (see Attachment 9, Harris Unit 1 Core Operating Limits Report).

3. The controls maintained on this procedure will exceed those normally required by AD-DC-ALL-0201, Development and Maintenance of Controlled Procedure Manual Procedures.

   a. Changes to this procedure will also require concurrent changes to the FSAR per AD-LS-ALL-0005, UFSAR Updates, when changes are made to the Technical Specification Equipment Lists.

   b. Changes to the Core Operating Limits Report are submitted to NRC per Technical Specification 6.9.1.6.4.

   c. This procedure is incorporated by reference into the FSAR and is therefore made a part of the FSAR, which is part of the Harris Current Licensing Basis (CLB). This procedure is subject to the update and reporting requirements of 10 CFR 50.71(e) and change controls of 10 CFR 50.59 and as such cannot be exempted from 50.59 review.

   By maintaining these additional controls on this procedure, its contents may be modified with assurance that they are accurate and that the full implications of the changes have been considered even more carefully than using routine methodology.

2.0 SCOPE (continued)

4. Operability of Structures, Systems, or Components (SSC) described in Technical Specifications

   The scope of the SSCs considered in the operability determination process includes the following:

   a. SSCs required to be OPERABLE by Tech Specs. These SSCs may perform required support functions for other SSCs required to be OPERABLE by Tech Specs (example - Emergency Diesel and Service Water)

   b. SSCs that are not explicitly required to be OPERABLE by Tech Specs, but that perform required support functions (as specified by the TS definition of operability) for SSCs required to be OPERABLE by Tech Specs.

3.0 DEFINITIONS

1. **COLR**: Core Operating Limits Report
2. **FSAR**: Final Safety Analysis Report
3. **LCO**: Limiting Condition for Operation
4. **LR**: License Renewal
5. **SSC**: Structures, Systems, or Components
6. **TS**: Technical Specification

4.0 RESPONSIBILITIES

1. Licensing/Regulatory Programs is responsible for administering the Technical Specifications List Program.

2. Operations is responsible for monitoring plant conditions to ensure the Technical Specifications List Program specified in this procedure is met.

5.0 INSTRUCTIONS

5.1 Criteria for Establishing the Equipment Lists/Surveillance Programs/Limits

5.1.1 Reactor Trip System Instrumentation Response Times (TS Table 3.3-2)

The response time of each Reactor Trip function shown in Technical Specification Table 3.3-1 shall be as shown in Attachment 1.

5.1.2 Engineered Safety Feature Actuation System Instrumentation Response Times (TS Table 3.3-5)

The response time of each Engineered Safety Feature Actuation System Function shown in Technical Specification Table 3.3-3 shall be as shown in Attachment 2.

5.1.3 Reactor Vessel Material Surveillance Program (TS Table 4.4-5)

The reactor vessel material irradiation surveillance specimens shall be withdrawn and examined to determine changes in material properties, as required by 10CFR50, Appendix H, in accordance with the schedule in Attachment 3.

The results of these examinations shall be used to update Technical Specification Figures 3.4-2 and 3.4-3.

NOTE

Based on HNP fluence projections, the HNP reactor vessel fracture toughness properties have been evaluated using established methods and techniques in accordance with the requirements of Regulatory Guide 1.99, Revision 2, and the Code of Federal Regulations, Title 10, Part 50.61. {7.1.1}

If future plant operations exceed the limitations in Section 1.3 of Regulatory Guide 1.99, Revision 2, {7.1.2}, or the applicable bounds, e.g., cold leg operating temperature and neutron fluence, as applied to the surveillance capsules, the impact of these plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified.

5.1.4 Snubbers (TS 4.7.8 and TS Figure 4.7-1)

The surveillance requirements for snubbers, as defined by Technical Specification 4.7.8, shall be performed in accordance with the augmented in-service inspection program as described in Attachment 4.

5.1.5 Containment Isolation Valves (TS Table 3.6-1)

The containment isolation valves required to be OPERABLE by Technical Specification 3.6.3 shall be demonstrated OPERABLE with isolation times shown in Attachment 5.

5.1.6 Containment Penetration Conductor Overcurrent Protective Devices (TS Table 3.8-1)

The containment penetration conductor overcurrent protective devices required by Technical Specification shall be as shown in Attachment 6.

5.1.7 Motor-Operated Valves Thermal Overload Protection Bypass (TS Table 3.8-2)

The thermal overload protection for motor-operated valves required to be bypassed by Technical Specification 3.8.4.2 shall be as shown in Attachment 7.

5.1.8 Safeguards Systems Isolation Valves in Closed Systems

The isolation ability of nonessential lines in closed systems is designed to meet single failure criteria, although only the outside containment isolation valve is covered by Technical Specification. The functionality requirement for non-containment isolation valves shall be as shown in Attachment 8. This functionality requirement is a desired backup (not a redundancy) feature. Unavailability of these valves to perform isolation functions is an off-normal condition but does not warrant required Technical Specifications actions.

5.1.9 Core Operating Limits Report

See Attachment 9, Harris Unit 1 Core Operating Limits Report, for instruction in locating the cycle-specific Engineering Instruction containing the COLR.

5.1.10 Instrument Uncertainties Associated with Reactor Vessel Pressure & Temperature Limits & Low Temperature Overpressure Protection

The values that will provide a conservative margin to compensate for instrument uncertainties related to reactor vessel pressure and temperature limits and low temperature overpressure protection are provided in Attachment 10.

5.1.11 Reactor Trip System Instrumentation Setpoint As-Found/As-Left Tolerances

The As-Found/As-Left tolerances of Reactor Trip Nuclear Instrumentation Setpoints shall be as shown in Attachment 11.

5.1.12 Diesel Fuel Oil Testing Program

The testing requirements of both new and stored fuel oil, along with the sediment cleaning of the fuel oil storage tank, have been relocated from Technical Specifications and are provided in Attachment 12.

5.2 Additions, Changes, and Deletions to the Attachments Required By Technical Specifications

1. If modifications are required to the respective lists, the following process shall be performed. This process exceeds the normal procedure review process as indicated in AD-DC-ALL-0201, Development and Maintenance of Controlled Procedure Manual Procedures.

    a. Prepare a revision to the procedure in accordance with AD-DC-ALL-0201 for review by plant personnel.

    b. Prior to approval by the Plant Manager - Harris, Licensing/Regulatory Programs shall ensure procedure revisions or new procedures are prepared to incorporate changes made to the attachments or shall establish action items per AD-PI-ALL-0100 for changes not requiring immediate implementation.

    c. The procedure revision shall be submitted to the General Manager - Harris Plant for final approval along with an FSAR revision (if required for changes to the Technical Specification Equipment List Program) which incorporates the changes made to the attachments in accordance with AD-LS-ALL-0005.

6.0 RECORDS

1. None

7.0 REFERENCES

7.1 Commitments

1. BAW-2355, Supplement 3, June 2006, "Supplement to the Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant - Reactor Vessel Material Surveillance Program," AREVA NP Inc.

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

7.2 Procedures

1. AD-DC-ALL-0201, Development and Maintenance of Controlled Procedure Manual Procedures

2. AD-DC-ALL-0202, Procedure Numbering System

3. AD-EG-ALL-1132, Preparation and Control of Design Change Engineering Changes

4. AD-EG-ALL-1137, Engineering Change Product Selection

5. AD-EG-ALL-1202, Preventive Maintenance and Surveillance Testing Administration

6. AD-EG-HNP-1618, Snubber Program Plan

7. AD-LS-ALL-0005, UFSAR Updates

8. AD-LS-ALL-0008, 10 CFR 50.59 Review Process

9. AD-PI-ALL-0100, Corrective Action Program

10. AP-013, Plant Nuclear Safety Committee

7.3 Miscellaneous Documents

1. SHNPP FSAR

2. SHNPP Technical Specifications

3. CR 188047, HNP LR Commitments 'Enhancement to Existing Program'

4. BAW-2355, Supplement 1, November 1999, "Supplement to the Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant - Reactor Vessel Material Surveillance Program," Framatome Technologies, Inc.

7.3 Miscellaneous Documents (continued)

5. BAW-2355, Supplement 2, June 2006, "Supplement to the Analysis of Capsule X Carolina Power & Light Company Shearon Harris Nuclear Power Plant - Reactor Vessel Material Surveillance Program," AREVA NP Inc.

6. EC 252649, Reduced Minimum Thimble Requirements for Full Core Flux Mapping

7. EC 254021, Disablement of Remote Actuation Capability of Unused Ammonia/Hydrazine Injection Valves, AR 95397

8. EC 260001, H1C14 Reload Core Design and Safety Analysis

9. EC 260241, Resolution of Unanalyzed Single Failure for E-6

10. EC 264030, H1C15 Reload Core Design and Safety Analysis

11. EC 267297, Change HNP COLR to Modify Rod Insertion Limit for Shutdown Rods

12. EC 272465, Harris Cycle 15 Burnup Extension

13. EC 269003, HNP Cycle 16 Core Design and Safety Analysis

14. EC 276187, H1C16 Reduced Minimum Thimble Requirements for Flux Mapping

15. EC 273394, HNP Cycle 17 Core Design and Safety Analysis

16. EC 283899, HNP Cycle 17 COLR Update - EOL MTC Surveillance Limit

17. EC 274914, Implement License Change for the MUR Uprate

18. EC 275840, HNP Cycle 18 Core Design and Safety Analysis

19. EC 288072, HNP Cycle 19 Core Design and Safety Analyses

20. EC 294210, Resolve AR 613482 (Fuse 1A-16/2746 Label Issues)

21. EC 403338, PHPP-1C Heater Restoration (EC300229, Temporarily Restored Heaters to Group C)

22. EC 409453, PRZR Level LT-460; Replace Barton With Rosemount

23. ESR 9400001, Steam Generator Replacement and Power Uprate

24. ESR 9400004, Instantaneous Trip Setting Tolerances for MCCBs

25. ESR 9400072, 1SW 39 & 40 Stroke Versus Flow Questions

7.3 Miscellaneous Documents (continued)

26. ESR 9400546, 1SW-39 & 40 Stroke Versus Flow Questions

27. ESR 9500941, Feedwater Isolation Time Acceptance Criteria

28. ESR 9600093, Cycle 8 Core Design and Analysis

29. ESR 9600098, 42/0 Seal-In Contact Modification for Control Room Isolation (EC 206196)

30. ESR 9700351, Change Status of SI-1&2 "BIT Inlet Isol Valves" to Normally (EC 245386)

31. ESR 9700807, SGR Large Bore Piping Modifications (EC 244109)

32. ESR 9700808, SGR Small Bore Piping Modifications (EC 206403)

33. ESR 9800537, Tempering Lines Modification (EC 248470)

34. ESR 9900093, Reactor Vessel Surveillance Capsule 'X' Analysis

35. ESR 0000221, Cycle 11 Core Design and Analysis (EC 245816)

36. ESR 0000325, RCS Head Vent Path Redesign (EC 244336)

37. ESR 0000262, PUR/SGR Instrumentation Changes Converted to EC 0000048392 Rev 009, Including ADL (EC 244641)

38. FCQL-465, 4/13/87, Parker to Loflin, VCT to RWST Alignment for Steam Line Break

39. HNEI-0400-0004, Harris Cycle 21 Core Operating Limits Report (COLR)

40. HNP Calculation HNP-P/LR-0613, License Renewal Aging Management Program Description of the Reactor Vessel Surveillance Program

41. NRC Inspection Manual Part 9900: Technical Guidance

42. NRC Letter to Mr. W. R. Robinson, "Issuance of Amendment No. 65 of Facility Operating License No. NPF-63 Regarding Relocation of Incore Instrument Requirements - Shearon Harris Nuclear Power Plant, Unit 1 (TAC No. M95064", July 24, 1996)

43. NRC letter to W.R. Jefferson, "Safety Evaluation for Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule", dated October 21, 2011.

7.3 Miscellaneous Documents (continued)

44. HNP Letter HNP-06-136, Letter from HNP to US Nuclear Regulatory Commission, Application for Renewal of Operating License, dated November 14, 2006

45. HNP Letter HNP-07-112, Letter from HNP to US Nuclear Regulatory Commission, "License Renewal Application, Amendment 1: Changes resulting from Request for Additional information, Site Audit Questions, and Applicant Identified Changes", dated August 20, 2007.

46. US NRC Letter, December 17, 2008, "Issuance of Renewed Facility Operating license No. NFP-63 for the Shearon Harris Nuclear Power Plant, Unit 1", License Condition 2.K.

47. Evaluation of RTS/ESFAS Tech Spec Related Setpoints, Allowable Values and Uncertainties

48. PCR 4839, Component Cooling Water Mechanical Snubber Failure

49. Shearon Harris Nuclear Plant – Safety Evaluation for Revision to Reactor Vessel Surveillance Capsule Withdrawal Schedule (TAC No, ME6998), dated October 21, 2011, ADAMS Accession No. ML11293A076

50. NRC Letter to T.M. Hamilton, “Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment Re: Changes to Technical Specifications Section 3.8, Electrical Power Systems and Section 6.8, Procedures and Programs (CAC No. MF7775), dated May 24, 2017.

51. NRC Letter to T.M. Hamilton, “Shearon Harris Nuclear Power Plant, Unit 1 - Issuance of Amendment : (1) Adopting TSTF-339, "Relocate TS parameters to the COLR"; (2) Adopting TSTF-5, "Delete Safety Limit Violation Notification requirements"; and (3) Removing a plant procedure referenced in Technical Specifications as it pertains to the Core Operating Limits Report (CAC No. MF8894; EPID L-2016-LLA-0023)

<< Reactor Trip System Instrumentation Response Times >>

FUNCTIONAL UNIT		**RESPONSE TIME**
1.	Manual Reactor Trip	NA
2.	Power Range, Neutron Flux	≤ 0.5 second*
3.	Power Range, Neutron Flux, High Positive Rate	NA
4.	Power Range, Neutron Flux, High Negative Rate	≤ 0.5 second*
5.	Intermediate Range, Neutron Flux	NA
6.	Source Range, Neutron Flux	NA
7.	Overtemperature ΔT	≤ 9.50 seconds * (through ΔT)**(1)**; ≤ 5.50 seconds * (through other inputs)**(1)**
8.	Overpower ΔT	≤ 9.50 seconds * (through ΔT)**(1)**; ≤ 5.50 seconds * (through other inputs)**(1)**
9.	Pressurizer Pressure--Low	≤ 2 seconds
10.	Pressurizer Pressure--High	≤ 2 seconds
11.	Pressurizer Water Level--High	NA
12.	Reactor Coolant Flow--Low	
	a. Single Loop (Above P-8) b. Two Loops (Above P-7 and below P-8)	≤ 1 second ≤ 1 second
13.	Steam Generator Water Level--Low-Low	≤ 3.5 seconds

<< Reactor Trip System Instrumentation Response Times >>

FUNCTIONAL UNIT		**RESPONSE TIME**
14.	Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	NA
15.	Undervoltage - Reactor Coolant Pumps (Above P-7)	≤ 1.5 seconds
16.	Underfrequency - Reactor Coolant Pumps (Above P-7)	≤ 0.6 second
17.	Turbine Trip	
	a. Low Fluid Oil Pressure	NA
	b. Turbine Throttle Valve Closure	NA
18.	Safety Injection Input from ESF	NA
19.	Reactor Trip System Interlocks	NA
20.	Reactor Trip Breakers	NA
21.	Automatic Trip and Interlock Logic	NA
22.	Reactor Trip Bypass Breakers	NA

TABLE NOTATIONS

*	Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.
(1)	The 9.50 seconds includes a 4.25-second limit for each RTD/thermowell thermal time constant (measured by a loop current step response [LCSR] test), a 1.25-second limit on the trip circuit electronics delay, and an additional 4.0-second electronics delay in only the measured ΔT circuitry (due to a lag filter time constant). Note that the safety analysis uses a 4.75-second thermal time constant due to allowances for the accuracy of the LCSR test, and a 4.40-second lag filter time constant (on measured ΔT) due to allowance for a 10% calibration tolerance.

<< Engineered Safety Features Response Times >>

INITIATION SIGNAL AND FUNCTION		**RESPONSE TIME IN SECONDS**
1.	Manual Initiation	
	a. Safety Injection (ECCS) (12)	NA
	b. Containment Spray	NA
	c. Containment Phase "A" Isolation	NA
	d. Containment Ventilation Isolation	NA
	e. Steam Line Isolation	NA
	f. Reactor Trip	NA
2.	Containment Pressure - High-1	
	a. Safety Injection (ECCS) (10)(12)	≤ 27 (1) / 12 (5) [High Press. Injec.] ≤ 29 (14) / 27 (15) [Low Press. Injec.]
	1) Reactor Trip	≤ 2
	2) Feedwater Isolation	≤ 10 (3)
	3) Containment Phase "A" Isolation	≤ 62 (2) / 72 (1)
	4) Containment Ventilation Isolation	≤ 4.75 (6)
	5) Motor-Driven Auxiliary Feedwater Pumps	≤ 60
	6) Emergency Service Water Pumps	≤ 32 (1) / 22 (8)
	7) Containment Fan Coolers	≤ 80 (1)(13) / 70 (8)(13)
	8) Control Room Isolation	NA

<< Engineered Safety Features Response Times >>

INITIATION SIGNAL AND FUNCTION		**RESPONSE TIME IN SECONDS**
3.	Pressurizer Pressure - Low	
	a. Safety Injection (ECCS) (10)(12)	≤ 27 (1) / 12 (5) [High Press. Injec.] ≤ 29 (14) / 27 (15) [Low Press. Injec.]
	1) Reactor Trip	≤ 2
	2) Feedwater Isolation	≤ 10 (3)
	3) Containment Phase "A" Isolation	≤ 62 (2) / 72 (1)
	4) Containment Ventilation Isolation	≤ 4.75 (6)
	5) Motor-Driven Auxiliary Feedwater Pumps	≤ 60
	6) Emergency Service Water Pumps	≤ 32 (1) / 22 (8)
	7) Containment Fan Coolers	≤ 80 (1)(13) / 70 (8)(13)
	8) Control Room Isolation	NA
4.	Main Steam Line Pressure - Low	
	a. Safety Injection (ECCS) (10)(12)	≤ 12 (5) / 22 (4)
	1) Reactor Trip	≤ 2
	2) Feedwater Isolation	≤ 10 (3)
	3) Containment Phase "A" Isolation	≤ 62 (2) / 72 (1)
	4) Containment Ventilation Isolation	≤ 4.75 (6)
	5) Motor-Driven Auxiliary Feedwater Pumps	≤ 60
	6) Emergency Service Water Pumps	≤ 32 (1) / 22 (8)
	7) Containment Fan Coolers	≤ 80 (1)(13) / 70 (8)(13)
	8) Control Room Isolation	NA
	b. Steam Line Isolation	≤ 7 (9)

<< Engineered Safety Features Response Times >>

INITIATION SIGNAL AND FUNCTION		**RESPONSE TIME IN SECONDS**
5.	Containment Pressure - High-3	
	a. Containment Spray	≤ 12.3 (8) / 24.3 (11)
	b. Containment Phase "B" Isolation	≤ 22.5 (1) / 12 (2)
6.	Containment Pressure - High-2	
	Steam Line Isolation	≤ 7 (9)
7.	Negative Steam Line Pressure Rate - High	
	Steam Line Isolation	≤ 7 (9)
8.	Steam Generator Water Level - High-High	
	a. Turbine Trip	≤ 2.5
	b. Feedwater Isolation	≤ 10 (3)
9.	Steam Generator Water Level - Low-Low	
	a. Motor-Driven Auxiliary Feedwater Pumps	≤ 61.5
	b. Turbine-Driven Auxiliary Feedwater Pumps	≤ 61.5
10.	Loss-of-Off-Site Power	
	Motor- and Turbine-Driven Auxiliary Feedwater Pumps	≤ 60
11.	Trip of All Main Feedwater Pumps	
	Motor-Driven Auxiliary Feedwater Pumps	NA
NOTE: AFW Isolation response time is only applicable when there is AFW flow.		
12	Steam Line Differential Pressure--High Coincident with Main Steam Line Isolation Signal	
	Isolate Auxiliary Feedwater to the Affected Steam Generator	≤ 41

<< Engineered Safety Features Response Times >>

INITIATION SIGNAL AND FUNCTION		**RESPONSE TIME IN SECONDS**
13.	RWST Level--Low-Low	
	a. Safety Injection Switchover to Containment Sump Coincident with Safety Injection	≤ 20
	b. Safety Injection Switchover to Containment Sump Coincident with Containment Spray	≤ 103
14.	Containment Radioactivity-High	
	a. Normal Containment Purge Isolation	≤ 3.5 (7)
	b. Pre-Entry Containment Purge Isolation	≤ 15 (7)

TABLE NOTATIONS	
(1)	Diesel generator starting and sequence loading delays included. RHR pump not included.
(2)	Diesel generator starting and sequence loading delay not included. Off site power available.
(3)	Applicable to Main Feedwater Isolation Valves only. Other valves that get a Feedwater Isolation signal are excluded.
(4)	Diesel generator starting and sequence loading delay included. RHR pumps not included.
(5)	Diesel generator starting and sequence loading delays not included. RHR pumps not included.
(6)	Isolation of Normal Containment Purge. This value is not applicable to Pre-entry Containment Purge which is permitted to be operating only in MODES 5 or 6 as per Technical Specification 3.6.1.7.
(7)	Response time testing of radiation monitors is not required.
(8)	Diesel generator starting delay not included, but sequencer loading delays are included.

<< Engineered Safety Features Response Times >>

TABLE NOTATIONS	
(9)	Applicable to main steam isolation valves only. Other valves that get a Main Steam Isolation signal (for example, bypass isolation, main steam drains to the condenser) are excluded.
(10)	RWST to CSIP opening times and VCT to CSIP closing times not included. Response time for these valves shall be ≤ 17[(2)] / 27[(1)] seconds for the RWST to CSIP valves and ≤ 27[(2)] / 37[(1)] seconds for the VCT to CSIP valves. [7.3.24]
	a) The Table Notation NOTEs (1) and (2) are related to on-site and off-site power considerations.
	b) The 27 second value is the collective stroke times for both valves to stroke,
	c) The 17 second value is specific to the RWST to CSIP stroke open time,
	d) The VCT to CSIP stroke close time is 10 seconds,
	e) The RWST to CSIP valve must complete its stroke open function before the system logic will allow the permissive for the VCT to CSIP valve to begin closure - hence 27 seconds minus 17 seconds equals 10 seconds maximum for this valve.
(11)	Containment spray response times includes diesel generator starting and sequencer loading delays and HI-1 signal. The containment spray HI-3 signal is excluded since HI-3 is actuated prior to the closure of the diesel generator output breaker.
(12)	Alternate Miniflow Valve (1CS-746 and 1CS-752) open/close times on Safety Injection signal coincident with RCS high pressure and low pressure are not included. These valves shall open in less than 15 seconds and close in less than 15 seconds.
(13)	These values are based on stroke times of valves 1SW-39 and 1SW-40. Since the Containment Fan Coolers are direct driven, their ESF response time is the time from the ESF actuation setpoint until the fan cooler circuit breaker closure actuation. This time shall not exceed the valve stroke time. (Reference letter number MS-864348(E) and ESR 94-00546.) The Containment Analysis assumes 110 seconds due to two-phase flow with LOCA/LOOP.
(14)	Diesel generator starting and sequence loading delays included. This entry is specific to RHR response.
(15)	Diesel generator starting and sequence loading not included. This entry is specific to RHR response.

ATTACHMENT 3
Page 1 of 2

<< Reactor Vessel Material Surveillance Program - Withdrawal Schedule* >>**

CAPSULE NUMBER	VESSEL LOCATION	LEAD* FACTOR	WITHDRAWAL TIME (EFPY)**
U	343°	2.9	1st Refueling
V	107°	3.3	3 EFPY
X	287°	2.68	9 EFPY
W	110°	2.38[(a)] 2.68[(b)]	18 EFPY ****
Y	290°	2.38[(a)] 2.68[(b)]	Either Y or Z but not both shall be withdrawn during Refueling Outage 21, at 27.2 EFPY. *****
Z	340°	2.38[(a)] 2.68[(b)]	

TABLE NOTATIONS	
(a)	Factor by which the capsule leads the vessel's maximum inner wall fluence for cycles 1 through 10.
(b)	Factor by which the capsule leads the vessel's maximum inner wall fluence for cycles 11 through 55 EFPY. Lead factor updated based on operation at an uprated core power level of 2948 MWt and due to the equilibrium loading pattern near the periphery for uprated power conditions.
*	The factor by which the capsule fluence leads the vessel maximum inner wall fluence. (Values are estimates based on calculations)
**	Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedules.
***	Changes to the reactor materials surveillance schedule must receive NRC approval prior to implementation. (Reference: Section III.B.3 of 10 CFR 50 Appendix H)

<< Reactor Vessel Material Surveillance Program - Withdrawal Schedule* >>**

TABLE NOTATIONS	
****	A withdrawal time range of 13.66 to 21.32 EFPY is permitted. Withdrawal of Capsule "W" occurred during Refueling Outage 16 (RFO-16), which is the outage immediately following the operating cycle in which the capsule fluence is equivalent to the 60-year maximum vessel fluence. {7.1.1}, {7.1.2}
*****	The licensee proposes to remove either Capsule Y or Z during their 21st refueling outage, circa 2018, when the capsule will have been exposed to a total neutron fluence of 9.39×10^{19} n/cm^2 (E > 1.0 MeV). Removing the capsule at this target fluence will provide valuable information in the higher fluence ranges, for which there is currently little experience or data. The staff reiterates the recommendation found in the GALL Report that either Capsule Y or Z be withdrawn and tested while the other is left in place and maintained in readiness should it become necessary at a future date. [7.3.49]

<< Snubbers >>

1.0 PURPOSE

To demonstrate that each snubber is OPERABLE by the performance of the following augmented inspection program.

Each snubber shall be demonstrated OPERABLE by performance of the following augmented Inservice Inspection Program.

2.0 REFERENCES

1. Technical Specification 3/4.7.8
2. ASME OM Code 2004 edition through 2006 addenda (includes Code Case OMN-13)
3. AD-EG-HNP-1618

3.0 DEFINITIONS

Definitions applicable to the snubber program and snubbers are in AD-EG-HNP-1618.

4.0 INSTRUCTIONS

4.1 Snubber Removal or Out of Service

Technical Specification 3.7.8 and the surveillance requirements apply to all snubbers. The only snubbers excluded from the requirements are those installed on non-safety related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system. A listing of individual snubbers is maintained in accordance with AD-EG-HNP-1618.

<< Snubbers >>

NOTE

Hangers are not subject to T.S. 3.7.8; the supported system is immediately declared INOPERABLE when hangers are removed or found damaged, unless an engineering evaluation has been completed. An exception to this requirement is if the hanger's only function is to secure a snubber. In this case, the snubber is INOPERABLE and only T.S. 3.7.8 applies.

4.1.1 System Operability

Technical Specification 3.7.8 allows 72 hours for one or more snubbers to be out of service. The attached system must be declared INOPERABLE by the end of this 72 hour period unless the snubber(s) are repaired or replaced, and a satisfactory engineering evaluation on the system is completed, except as specified in Section 4.1.2. Snubber removal on redundant trains of a system is not allowed on systems required to be OPERABLE in the existing mode. For example, a snubber cannot be removed from "A" RHR concurrently with an INOPERABLE snubber on "B" RHR.

For those systems covered by a specific technical specification, the action required shall be as specified. If the attached piping is common to both safety trains of a system, the snubber shall be isolated from at least one train, or both shall be declared INOPERABLE. However, if the piping attached to the INOPERABLE snubber is a non-essential portion of the system, it may be isolated from the rest of the system, removed from service and depressurized, and the system may be declared OPERABLE, and operation may continue. Such actions must be accomplished within the times specified by the system technical specification.

Snubbers located on piping with a containment isolation valve where the snubber is adjacent to the valve require application of Specification 3.6.3 for that valve, after expiration of the 72 hour limit per Technical Specification 3.7.8.

During power operations, the required number of hydraulic snubbers per steam generator is three (RAF-3106).

For those systems not covered by a specific Technical Specification, the piping attached to the INOPERABLE snubber shall be isolated, removed from service and depressurized within the next 72 hours; that is, an additional 72 hours after expiration of the 72 hours in Technical Specification 3.7.8.

Operation of Technical Specification systems in modes where their LCOs are not applicable is not restricted due to snubber removal (for example, use of AFW in MODE 5). However, if water hammer does occur, it should be recorded and referred to Harris Nuclear Plant Engineering for inspection and evaluation.

<< Snubbers >>

4.1.2 Determination of Inoperability & Engineering Evaluations

The responsibility for determination of operability or inoperability rests with the Operating Shift. Engineering will be responsible for providing the necessary technical input to assist with making or validating any operability determinations.

The 72 hour clock in Technical Specification 3.7.8 can be started by three events: removal for access or testing, visual inspection, or bench testing. These events are discussed by the following:

a. Removal of a snubber for testing, or access to do work, requires starting of the 72 hour clock but does not require an engineering evaluation under the action statement of the specification. The clock is stopped when the snubber is reinstalled and satisfactorily examined.

b. If a snubber is declared INOPERABLE based on visual inspection, the 72 hour clock may be stopped with either:

   (1) a successful functional test and reinstallation of the snubber, or

   (2) installation of a repaired or replacement snubber and verification that the "as left" condition of the pipe is acceptable and meets the design criteria of the system (PCR 4839); this is documented on the Work Order (WO). If the snubber fails the functional test, the clock continues to run, unless option (2) is completed.

c. If a snubber is declared INOPERABLE based on a functional test, or is verified INOPERABLE by a functional test following visual determination of inoperability, the 72 hour clock is started or continues to run, and continues until a repaired or replacement snubber is installed and satisfactorily examined, and an engineering evaluation is complete either per option (2) above or by separate Engineering Change Request (ECR). The evaluation must also address the cause of the failure. Determination of reportability is required when a generic program deficiency or specific design deficiency is identified as the cause.

4.2 Visual Examinations, Functional Testing and Service Life Monitoring

Visual examinations, functional testing and service life monitoring shall be performed in accordance with the ASME OM Code Subsection ISTD 2004 edition through 2006 addenda and Code Case OMN-13 as described in AD-EG-HNP-1618.

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO.CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1. PHASE A ISOLATION					
1	1MS-29 (MS-V126)	MS LINE C TO SAMPLING SYSTEM	60	1, 6, 13	NONE
2	1MS-27 (MS-V124)	MS LINE B TO SAMPLING SYSTEM	60	1, 6, 13	NONE
3	1MS-25 (MS-V122)	MS LINE A TO SAMPLING SYSTEM	60	1, 6, 13	NONE
7	1CS-7 (CS-V511)	CVCS NORMAL LTDN ISOL	10	7, 13	1CS-11
7	1CS-8 (CS-V512)	CVCS NORMAL LTDN ISOL	10	7, 13	1CS-11
7	1CS-9 (CS-V513)	CVCS NORMAL LTDN ISOL	10	7, 13	1CS-11
7	1CS-11 (CS-V518)	CVCS NORMAL LTDN ISOL	10	7, 13	1CS-7,8, 9 and 10
12	1CS-470 (CS-V516)	RCP SEAL WTR RETURN & EXCESS LTDN	10	7, 13	1CS-472
12	1CS-472 (CS-V517)	RCP SEAL WTR RETURN & EXCESS LTDN	10	7, 13	1CS-470 and 471
33	1SP-209 (SP-V408)	GAS RETURN FROM PASS SKID #2	5	7, 13	1SP-208
33	1SP-208 (SP-V409)	GAS RETURN FROM PASS SKID #2	5	7, 13	1SP-209
37	1CC-176 (CC-V172)	CCW TO RCDT & EXCESS LTDN HEAT EXCHS	10	1, 6, 13	NONE
38	1CC-202 (CC-V182)	CCW FROM RCDT & EXCESS LTDN HEAT EXCHS	10	1, 6, 13	NONE
40	1RC-161 (RC-D525)	REACTOR MAKEUP WTR TO PRT	60	7, 13	1RC-164
42	1ED-121 (WL-L600)	RCDT PUMPS DISCH	10	7, 13	1ED-125
42	1ED-125 (WL-D650)	RCDT PUMPS DISCH	10	7, 13	1ED-121 and 119
73A	1SP-12 (SP-V300)	HYDROGEN ANALYZER A	60	7, 13	1SP-915
73A	1SP-915 (SP-V348)	HYDROGEN ANALYZER A	60	7, 13	1SP-12

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO.CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1.	PHASE A ISOLATION (continued)				
73B	1SP-941 (SP-V301)	HYDROGEN ANALYZER A	60	7, 13	1SP-917
73B	1SP-917 (SP-V349)	HYDROGEN ANALYZER A	60	7, 13	1SP-941
74	1ED-94 (MD-V36)	CNMT SUMP PUMP DISCH	60	7, 13	1ED-95
74	1ED-95 (MD-V77)	CNMT SUMP PUMP DISCH	60	7, 13	1ED-94
76A	1SI-179 (SI-V554)	ACCUMULATOR FILL FROM RWST	10	7, 13	1SI-182
76B	1SI-263 (SI-V555)	ACCUMULATOR DRAIN TO RWST	10	7, 13	1SI-264
76B	1SI-264 (SI-V550)	ACCUMULATOR DRAIN TO RWST	10	7, 13	1SI-263
77A	1SI-287 (SI-V530)	NITROGEN SUPPLY	10	7, 13	1SI-290
77B	1RC-141 (RC-D528)	PRT NITROGEN CONNECTION	10	7, 13	1RC-144
77B	1RC-144 (RC-D529)	PRT NITROGEN CONNECTION	10	7, 13	1RC-141
77C	1ED-164 (WG-D590)	RCDT HYDROGEN CONNECTION	10	7, 13	1ED-161
77C	1ED-161 (WG-D291)	RCDT HYDROGEN CONNECTION	10	7, 13	1ED-164
78A	1SP-948 (SP-V111)	RCS SAMPLE	60	7, 13	1SP-949
78A	1SP-949 (SP-V23)	RCS SAMPLE	60	7, 13	1SP-948
78B	1SP-40 (SP-V11)	PRESSURIZER LIQ SAMPLE	60	7, 13	1SP-41
78B	1SP-41 (SP-V12)	PRESSURIZER LIQ SAMPLE	60	7, 13	1SP-40
78C	1SP-59 (SP-V1)	PRESSURIZER STEAM SAMPLE	60	7, 13	1SP-60
78C	1SP-60 (SP-V2)	PRESSURIZER STEAM SAMPLE	60	7, 13	1SP-59
78D	1SP-78 (SP-V113)	ACCUMULATOR A SAMPLE	60	7, 13	1SP-85

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO.CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
1.	PHASE A ISOLATION (continued)				
78D	1SP-81 (SP-V114)	ACCUMULATOR B SAMPLE	60	7, 13	1SP-85
78D	1SP-84 (SP-V115)	ACCUMULATOR C SAMPLE	60	7, 13	1SP-85
78D	1SP-85 (SP-V116)	ACCUMULATORS SAMPLE	60	7, 13	1SP-78, 81, and 84
80	1IA-819 (IA-V192)	INSTRUMENT AIR SUPPLY	60	7, 13	1IA-220
83A	1SP-916 (SP-V448)	RADIATION MONITOR REM-3502A	60	7, 13	1SP-16
83A	1SP-16 (SP-V449)	RADIATION MONITOR REM-3502A	60	7, 13	1SP-916
83B	1SP-918 (SP-V450)	RADIATION MONITOR REM-3502A	60	7, 13	1SP-939
83B	1SP-939 (SP-V451)	RADIATION MONITOR REM-3502A	60	7, 13	1SP-918
86A	1SP-42 (SP-V308)	HYDROGEN ANALYZER B	60	7, 13	1SP-919
86A	1SP-919 (SP-V314)	HYDROGEN ANALYZER B	60	7, 13	1SP-42
86B	1SP-62 (SP-V309)	HYDROGEN ANALYZER B	60	7, 13	1SP-943
86B	1SP-943 (SP-V315)	HYDROGEN ANALYZER B	60	7, 13	1SP-62
88	1SP-201 (SP-V406)	LIQUID RETURN FROM PASS SKID #1	5	7, 13	1SP-200
88	1SP-200 (SP-V407)	LIQUID RETURN FROM PASS SKID #1	5	7, 13	1SP-201
91	1SW-240 (SW-B89)	SERVICE WTR FROM NNS FAN COILS	60	7, 13	1SW-242
91	1SW-242 (SW-B90)	SERVICE WTR FROM NNS FAN COILS	60	7, 13	1SW-240
92	1SW-231 (SW-B88)	SERVICE WTR TO NNS FAN COILS	60	7, 13	1SW-233
105	1FP-347 (FP-B1)	FIRE WATER SPRINKLER SUPPLY	60	7, 13	1FP-349

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO.CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
2. PHASE B ISOLATION					
NOTE: Valve 1CC-207 in penetration 35 is not classified as a containment isolation valve.					
35	1CC-208 (CC-V170)	CCW TO RCPS	10	7, 13	1CC-211
36	1CC-297 (CC-V184)	CCW FROM RCP BEARING OIL HXS	10	7, 13	1CC-299
36	1CC-299 (CC-V183)	CCW FROM RCP BEARING OIL HXS	10	7, 13	1CC-297 and 298
39	1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIER HXS	10	7, 13	1CC-251
39	1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIER HXS	10	7, 13	1CC-249 and 250
3. SAFETY INJECTION ACTUATION					
NOTE: Valve 1CS-235 in penetration 8 is not classified as a containment isolation valve; refer to Specifications 3.1.2.1, 3.1.2.2, 3.3.3.5.b, 3.5.2, and 3.5.3.					
8	1CS-238 (CS-V610)	CVCS NORMAL CHARGING	10	7, 13	1CS-477
NOTE: Refer to Attachment 8 for additional requirements for SG Blowdown and Sample valves.					
51	1BD-11 (BD-V11)	SG A BLOWDOWN	60	1, 6, 13	NONE
52	1BD-30 (BD-V15)	SG B BLOWDOWN	60	1, 6, 13	NONE
53	1BD-49 (BD-V19)	SB C BLOWDOWN	60	1, 6, 13	NONE
54	1SP-217 (SP-V120)	SG A SAMPLE	60	1, 6, 13	NONE
55	1SP-222 (SP-V121)	SG B SAMPLE	60	1, 6, 13	NONE
56	1SP-227 (SP-V122)	SG C SAMPLE	60	1, 6, 13	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
4. CONTAINMENT VENTILATION ISOLATION					
57	1CP-9 (CP-B1)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5, 7, 13	1CP-6 and 7
57	1CP-10 (CP-B3)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	5, 7, 13	1CP-6 and 7
57	1CP-7 (CP-B4)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (42")	15	5, 7, 13	1CP-9 and 10
57	1CP-6 (CP-B2)	CONTAINMENT ATMOSPHERE PURGE MAKEUP (8")	3.5	5, 7, 13	1CP-9 and 10
58	1CP-4 (CP-B7)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	5, 7, 13	1CP-1 and 3
58	1CP-5 (CP-B5)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5, 7, 13	1CP-1 and 3
58	1CP-1 (CP-B8)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (42")	15	5, 7, 13	1CP-4 and 5
58	1CP-3 (CP-B6)	CONTAINMENT ATMOSPHERE PURGE EXHAUST (8")	3.5	5, 7, 13	1CP-4 and 5
59	1CB-2 (CB-B1)	CONTAINMENT VACUUM RELIEF	5	7, 13	1CB-3
98	1CB-6 (CB-B2)	CONTAINMENT VACUUM RELIEF	5	7, 13	1CB-7

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
5.	MAIN STEAM LINE ISOLATION				
1	1MS-84 (MS-V3)	MSIV (SG C)	5	1, 4, 13	NONE
1	1MS-85 (MS-F3)	MSIV BYPASS (SG C)	10	1, 6, 13	NONE
1	1MS-301 (MS-V61)	MS DRAIN TO CONDENSER	60	1, 6, 13	NONE
2	1MS-82 (MS-V2)	MSIV (SG B)	5	1, 4, 13	NONE
2	1MS-83 (MS-F2)	MSIV BYPASS (SG B)	10	1, 6, 13	NONE
2	1MS-266 (MS-V60)	MS DRAIN TO CONDENSER	60	1, 6, 13	NONE
3	1MS-80 (MS-V1)	MSIV (SG A)	5	1, 4, 13	NONE
3	1MS-81 (MS-F1)	MSIV BYPASS (SG A)	10	1, 6, 13	NONE
3	1MS-231 (MS-V59)	MS DRAIN TO CONDENSER	60	1, 6, 13	NONE
6.	MAIN FEEDWATER LINE ISOLATION				

NOTE: Refer to Attachment 8 for additional requirements for Main Feedwater valves.

4	1FW-159 (FW-V26)	FEEDWATER LOOP A	8	1, 6, 12, 13	NONE
5	1FW-277 (FW-V27)	FEEDWATER LOOP B	8	1, 6, 12, 13	NONE
6	1FW-217 (FW-V28)	FEEDWATER LOOP C	8	1, 6, 12, 13	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
7.	AUXILIARY FEEDWATER ISOLATION				
NOTE: AFW flow control valves are not classified as containment isolation valves; refer to Specification 3.7.1.2.					
108	1AF-55 (AF-V10)	AUXILIARY FEEDWATER TO SG A	24	1, 6, 13	NONE
108	1AF-137 (AF-V116)	AUXILIARY FEEDWATER TO SG A	24	1, 6, 13	NONE
109	1AF-93 (AF-V19)	AUXILIARY FEEDWATER TO SG B	24	1, 6, 13	NONE
109	1AF-143 (AF-V117)	AUXILIARY FEEDWATER TO SG B	24	1, 6, 13	NONE
110	1AF-74 (AF-V23)	AUXILIARY FEEDWATER TO SG C	24	1, 6, 13	NONE
110	1AF-149 (AF-V118)	AUXILIARY FEEDWATER TO SG C	24	1, 6, 13	NONE
8.	REMOTE MANUAL VALVES				
1	1MS-72 (MS-V9)	MAIN STEAM C TO AUXILIARY FW TURBINE	N/A	1, 2, 6	NONE
1	1MS-62 (MS-P20)	SG PORV C	N/A	1, 2, 6	1MS-63
2	1MS-70 (MS-V8)	MAIN STEAM B TO AUXILIARY FW TURBINE	N/A	1, 2, 6	NONE
2	1MS-60 (MS-P19)	SG PORV B	N/A	1, 2, 6	1MS-61
3	1MS-58 (MS-P18)	SG PORV A	N/A	1, 2, 6	1MS-59
9	1CS-341 (CS-V522)	CVCS SEAL WATER TO RCP A	N/A	2, 7	1CS-344
10	1CS-382 (CS-V523)	CVCS SEAL WATER TO RCP B	N/A	2, 7	1CS-385
11	1CS-423 (CS-V524)	CVCS SEAL WATER TO RCP C	N/A	2, 7	1CS-426
13	1SI-340 (SI-V579)	SI-LOW HEAD TO COLD LEGS	N/A	1, 2, 7	1SI-346
14	1SI-341 (SI-V578)	SI-LOW HEAD TO COLD LEGS	N/A	1, 2, 7	1SI-347

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
8.	REMOTE MANUAL VALVES (continued)				
15	1RH-2 (RH-V503)	RHR PUMP A SUCTION FROM HOT LEG	N/A	1, 2, 8	NONE
16	1RH-40 (RH-V501)	RHR PUMP B SUCTION FROM HOT LEG	N/A	1, 2, 8	NONE
17	1SI-3 (SI-V505)	SI-HIGH HEAD TO COLD LEGS	N/A	1, 2, 7	1SI-8, 9, and 10
17	1SI-4 (SI-V506)	SI-HIGH HEAD TO COLD LEGS	N/A	1, 2, 7	1SI-8, 9, and 10
18	1SI-359 (SI-V587)	SI-LOW HEAD TO HOT LEGS	N/A	1, 2, 7	1SI-134 and 135
20	1SI-107 (SI-V500)	SI-HIGH HEAD TO HOT LEGS	N/A	1, 2, 7	1SI-127, 128, and 129
21	1SI-86 (SI-V501)	SI-HIGH HEAD TO HOT LEGS	N/A	1, 2, 7	1SI-104, 105, and 106
22	1SI-52 (SI-V502)	SI-HIGH HEAD TO COLD LEGS	N/A	1, 2, 7	1SI-72, 73, and 74
23	1CT-50 (CT-V21)	CONTAINMENT SPRAY A	N/A	2, 7	1CT-53
24	1CT-88 (CT-V43)	CONTAINMENT SPRAY B	N/A	2, 7	1CT-91
25	1SW-92 (SW-B46)	SERVICE WATER TO FAN COOLER AH-3	N/A	1, 2, 6	NONE
26	1SW-91 (SW-B45)	SERVICE WATER TO FAN COOLER AH-2	N/A	1, 2, 6	NONE
27	1SW-225 (SW-B52)	SERVICE WATER TO FAN COOLER AH-1	N/A	1, 2, 6	NONE
28	1SW-227 (SW-B51)	SERVICE WATER TO FAN COOLER AH-4	N/A	1, 2, 6	NONE
29	1SW-97 (SW-B47)	SERVICE WATER FROM FAN COOLER AH-3	N/A	1, 2, 6	NONE
30	1SW-109 (SW-B49)	SERVICE WATER FROM FAN COOLER AH-2	N/A	1, 2, 6	NONE
31	1SW-98 (SW-B48)	SERVICE WATER FROM FAN COOLER AH-1	N/A	1, 2, 6	NONE
32	1SW-110 (SW-B50)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1, 2, 6	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
8.	REMOTE MANUAL VALVES (continued)				
47	1SI-300 (SI-V571)	CONTAINMENT SUMP TO RHR PUMP A	N/A	1, 2, 8	NONE
48	1SI-301 (SI-V570)	CONTAINMENT SUMP TO RHR PUMP B	N/A	1, 2, 8	NONE
49	1CT-105 (CT-V6)	CONTAINMENT SUMP TO CT PUMP A	N/A	1, 2, 8	NONE
50	1CT-102 (CT-V7)	CONTAINMENT SUMP TO CT PUMP B	N/A	1, 2, 8	NONE
63	1CM-2 (CM-B5)	HYDROGEN PURGE EXHAUST	N/A	2, 7	1CM-4
9.	MANUAL VALVES				
NOTE: Manual valves must be locked closed to be considered OPERABLE, except as provided in Table Notation 3.					
4	1FW-165 (FW-V89)	FEEDWATER LOOP A HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
4	1FW-163 (FW-V90)	FEEDWATER LOOP A AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
5	1FW-279 (FW-V91)	FEEDWATER LOOP B AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
5	1FW-281 (FW-V92)	FEEDWATER LOOP B HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
6	1FW-223 (FW-V93)	FEEDWATER LOOP C HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
6	1FW-221 (FW-V94)	FEEDWATER LOOP C AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6, 13	NONE
34	1LT-6 (LT-V2)	ILRT ROTOMETER (LOCKED CLOSED)	N/A	N/A	NONE
41	1SA-80 (SA-V14)	SERVICE AIR SUPPLY (LOCKED CLOSED)	N/A	3	1SA-82

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
8. MANUAL VALVES (continued)					
42	1ED-119 (WL-D651)	RCDT PUMP DISCH BYPASS (LOCKED CLOSED)	N/A	3	1ED-125
44	1SF-145 (SF-D164)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	N/A	1SF-144
44	1SF-144 (SF-D165)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	N/A	1SF-145
45	1SF-118 (SF-D25)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	N/A	1SF-119
45	1SF-119 (SF-D26)	REFUELING CAVITY CLEANUP (LOCKED CLOSED)	N/A	N/A	1SF-118
51	1BD-270 (BD-V183)	SG 1A 1BD-11 BYPASS CIV (LOCKED CLOSED)	N/A	1, 3, 6	NONE
52	1BD-272 (BD-V184)	SG 1B 1BD-30 BYPASS CIV (LOCKED CLOSED)	N/A	1, 3, 6	NONE
53	1BD-274 (BD-V185)	SG 1C 1BD-49 BYPASS CIV (LOCKED CLOSED)	N/A	1, 3, 6	NONE
61	1CM-5 (CM-B6)	H2 PURGE MAKEUP (LOCKED CLOSED)	N/A	3	1CM-7
62	1LT-10 (LT-V4)	ILRT (LOCKED CLOSED)	N/A	N/A	NONE
63	1CM-4 (CM-B4)	H2 PURGE EXHAUST (LOCKED CLOSED)	N/A	3	1CM-2
79	1FP-355 (FP-V44)	FIRE WATER STANDPIPE SUPPLY (LOCKED CLOSED)	N/A	3	1FP-357
90	1DW-63 (DW-V120)	DEMIN WATER SUPPLY (LOCKED CLOSED)	N/A	3	1DW-65
96	1LT-4 (LT-V1)	ILRT (LOCKED CLOSED)	N/A	N/A	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
9. MANUAL VALVES (continued)					
108	1AF-174 (AF-V189)	WET LAY-UP TO SG A AFW HEADER (LOCKED CLOSED)	N/A	1, 6	NONE
108	1AF-155 (AF-V162)	AUX FW LOOP A HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE
108	1AF-153 (AF-V163)	AUX FW LOOP A AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE
109	1AF-173 (AF-V190)	WET LAY-UP TO SG B AFW HEADER (LOCKED CLOSED)	N/A	1, 6	NONE
109	1AF-159 (AF-V164)	AUX FW LOOP B HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE
109	1AF-157 (AF-V165)	AUX FW LOOP B AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE
110	1AF-175 (AF-V191)	WET LAY-UP TO SG C AFW HEADER (LOCKED CLOSED)	N/A	1, 6	NONE
110	1AF-163 (AF-V166)	AUX FW LOOP C HYDRAZINE CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE
110	1AF-161 (AF-V167)	AUX FW LOOP C AMMONIA CONNECTION (LOCKED CLOSED)	N/A	1, 6	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
10. CHECK VALVES					
8	1CS-477 (CS-V515)	CVCS NORMAL CHARGING	N/A	7	1CS-238
9	1CS-344 (CV-V25)	CVCS SEAL WATER TO RCP A	N/A	7	1CS-341
10	1CS-385 (CS-V26)	CVCS SEAL WATER TO RCP B	N/A	7	1CS-382
11	1CS-426 (CS-V27)	CVCS SEAL WATER TO RCP C	N/A	7	1CS-423
12	1CS-471 (CS-V67)	CVCS SEAL WATER RETURN & EXCESS LETDOWN	N/A	7	1CS-472
13	1SI-346 (SI-V581)	SI-LOW HEAD TO COLD LEGS	N/A	1, 7	1SI-340
14	1SI-347 (SI-V580)	SI-LOW HEAD TO COLD LEGS	N/A	1, 7	1SI-341
17	1SI-8 (SI-V17)	SI-HIGH HEAD TO LOOP A COLD LEG	N/A	1, 7	1SI-3, 4, and 43
17	1SI-9 (SI-V23)	SI-HIGH HEAD TO LOOP B COLD LEG	N/A	1, 7	1SI-3, 4, and 43
17	1SI-10 (SI-V29)	SI-HIGH HEAD TO LOOP C COLD LEG	N/A	1, 7	1SI-3, 4, and 43
18	1SI-134 (SI-V510)	SI-LOW HEAD TO LOOP A HOT LEG	N/A	1, 7	1SI-359
18	1SI-135 (SI-V511)	SI-LOW HEAD TO LOOP B HOT LEG	N/A	1, 7	1SI-359
20	1SI-127 (SI-V84)	SI-HIGH HEAD TO LOOP A HOT LEG	N/A	1, 7	1SI-107
20	1SI-128 (SI-V90)	SI-HIGH HEAD TO LOOP B HOT LEG	N/A	1, 7	1SI-107
20	1SI-129 (SI-V96)	SI-HIGH HEAD TO LOOP C HOT LEG	N/A	1, 7	1SI-107
21	1SI-104 (SI-V39)	SI-HIGH HEAD TO LOOP A HOT LEG	N/A	1, 7	1SI-86
21	1SI-105 (SI-V45)	SI-HIGH HEAD TO LOOP B HOT LEG	N/A	1, 7	1SI-86
21	1SI-106 (SI-V51)	SI-HIGH HEAD TO LOOP C HOT LEG	N/A	1, 7	1SI-86

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
10. CHECK VALVES (continued)					
22	1SI-72 (SI-V63)	SI-HIGH HEAD TO LOOP A COLD LEG	N/A	1, 7	1SI-52
22	1SI-73 (SI-V69)	SI-HIGH HEAD TO LOOP B COLD LEG	N/A	1, 7	1SI-52
22	1SI-74 (SI-V75)	SI-HIGH HEAD TO LOOP C COLD LEG	N/A	1, 7	1SI-52
23	1CT-53 (CT-V27)	CONTAINMENT SPRAY TRAIN A	N/A	7	1CT-50
24	1CT-91 (CT-V51)	CONTAINMENT SPRAY TRAIN B	N/A	7	1CT-88
35	1CC-211 (CC-V171)	CCW TO RCPS	N/A	7	1CC-208
36	1CC-298 (CC-V51)	CCW FROM RCP BEARING OIL HSX	N/A	7	1CC-299
39	1CC-250 (CC-V50)	CCW FROM RCP THERMAL BARRIER HXS	N/A	7	1CC-251
40	1RC-164 (RC-V525)	DEMIN WATER TO PRT	N/A	7	1RC-161
41	1SA-82 (SA-V15)	SERVICE AIR SUPPLY	N/A	7	1SA-80
59	1CB-3 (CB-V1)	CONTAINMENT VACUUM RELIEF	N/A	7	1CB-2
61	1CM-7 (CM-V1)	HYDROGEN PURGE MAKEUP	N/A	7	1CM-5
76A	1SI-182 (SI-V150)	ACCUMULATOR FILL FROM RWST	N/A	7	1SI-179
77A	1SI-290 (SI-V188)	NITROGEN SUPPLY	N/A	7	1SI-287
79	1FP-357 (FP-V48)	FIRE WATER STANDPIPE SUPPLY	N/A	7	1FP-355
80	1IA-220 (1A-V33)	INSTRUMENT AIR SUPPLY	N/A	7	1IA-819
90	1DW-65 (DW-V121)	DEMIN WATER SUPPLY	N/A	7	1DW-63
92	1SW-233 (SW-V142)	SERVICE WATER TO NNS FAN COILS	N/A	7	1SW-231

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
10. CHECK VALVES (continued)					
94A	PDT-01CB-7680BISB-CV (B)	EXCS FLOW CHK VLV FOR CNMT VACUUM RELIEF SENSING	N/A	1, 10	NONE
94B	PDT-01CB-7680BSB-CV (B)	EXCS FLOW CHK VLV FOR CNMT VACUUM RELIEF SENSING	N/A	1, 10	NONE
94C	PDT-01CP-7611S-CV (B)	EXCS FLOW CHK VLV FOR CNMT VACUUM RELIEF SENSING	N/A	1, 10	NONE
95A	PDT-01CB-7680ASA-CV (B)	EXCS FLOW CHK VLV FOR CNMT VACUUM RELIEF SENSING	N/A	1, 10	NONE
95B	PDT-01CB-7680AISA-CV (B)	EXCS FLOW CHK VLV FOR CNMT VACUUM RELIEF SENSING	N/A	1, 10	NONE

<< Containment Isolation Valves >>

PENETRATION NO.	VALVE NO. CP&L (EBASCO)	FUNCTION	MAXIMUM ISOLATION TIME (SEC)	APPLICABLE NOTES	REDUNDANT VALVE(S)
11.	RELIEF VALVES				
NOTE: SG Safety Relief Valves are governed by Specification 3.7.1.1.					
7	1CS-10 (CS-R500)	CVCS NORMAL LETDOWN	N/A	7, 9	1CS-11
15	1RH-7 (RH-R501)	RHR SUCTION FROM HOT LEG	N/A	1, 8, 9	NONE
16	1RH-45 (RH-R500)	RHR SUCTION FROM HOT LEG	N/A	1, 8, 9	NONE
29	1SW-95 (SW-R1)	SERVICE WATER FROM FAN COOLER AH-3	N/A	1, 6, 9	NONE
30	1SW-107 (SW-R3)	SERVICE WATER FROM FAN COOLER AH-2	N/A	1, 6, 9	NONE
31	1SW-96 (SW-R2)	SERVICE WATER FROM FAN COOLER AH-1	N/A	1, 6, 9	NONE
32	1SW-108 (SW-R4)	SERVICE WATER FROM FAN COOLER AH-4	N/A	1, 6, 9	NONE
38	1CC-194 (CC-R6)	CCW FROM EXCESS LETDOWN HEAT EXCHANGER	N/A	1, 7, 9	1CC-202 and 176
38	1CC-186 (CC-R5)	CCW FROM RCDT HEAT EXCHANGER	N/A	1, 7, 9	1CC-202 and 176
91	1SW-1494 (SW-R18)	NSW CONT FAN UNITS RETURN HEADER RELIEF VLV (AH-37, AH-38, AND AH-39)	N/A	7, 9	1SW-242

<< Containment Isolation Valves >>

TABLE NOTATIONS	
(1)	This valve is not subject to Type C leakage tests.
(2)	Remote manual valves may be opened as required for plant operations.
(3)	Manual valves in this penetration may be unlocked and opened as allowed under the action requirements, or on an intermittent basis under administrative controls established by the Plant Nuclear Safety Committee for each particular situation. The administrative controls shall include as a minimum stationing a dedicated operator at the valve controls, who is in continuous communication with the control room.
(4)	The Main Steam Isolation Valves (MSIVs) are included for table completeness. The requirements of Technical Specification 3.6.3 do not apply because the OPERABILITY requirements for the MSIVs are governed by Technical Specification 3.7.1.5.
(5)	This valve may be opened only as permitted by Technical Specification 3.6.1.7.
(6)	For this valve, the closed system in which it is located is considered to be an OPERABLE isolation valve for purposes of compliance with the initial ACTION statement of Specification 3.6.3 which states "maintain at least one isolation valve OPERABLE. . ." Further action under Specification 3.6.3 is still required to isolate the affected penetration or to shut down in accordance with Actions a, b, c, or d. Reopening of a valve declared INOPERABLE to comply with Tech Spec ACTIONS is allowed to permit surveillance testing to demonstrate its operability or the operability of other equipment per Specification 3.0.5. The following guidance is provided for complying with the follow-up action requirement, specified in Actions a, b, or c, to isolate the penetration: • Either the INOPERABLE valve must be closed (and de-activated, if applicable for power-operated valves), OR • A valve having the same safety class and seismic design class in series with the INOPERABLE valve must be closed (and de-activated, if applicable for power-operated valves). If the piping branches, each branch must be isolated. • Check valves may not be used to isolate a penetration beyond the four-hour period.

<< Containment Isolation Valves >>

TABLE NOTATIONS	
(7)	For this valve, the valves listed in the REDUNDANT VALVE column are the valves which are used to meet the initial ACTION statement of Specification 3.6.3 which states "maintain at least one isolation valve OPERABLE. . ." Further action under Specification 3.6.3 is still required to isolate the affected penetration or to shut down in accordance with Actions a, b, c, or d. Reopening of a valve declared INOPERABLE to comply with Tech Spec ACTIONS is allowed to permit surveillance testing to demonstrate its operability or the operability of other equipment per Specification 3.0.5. The following guidance is provided for complying with the follow-up action requirement, specified in Actions a, b, or c, to isolate the penetration: • Either the INOPERABLE valve, or all containment isolation valves listed under "REDUNDANT VALVE(S)" column, must be closed (and de-activated, if applicable, for power-operated valves), OR • A valve having the same safety class and seismic design class in series with the INOPERABLE valve must be closed (and de-activated, if applicable for power-operated valves). If the piping branches, each branch must be isolated. • Check valves may not be used to isolate a penetration beyond the four-hour period.
(8)	For this valve, the closed, water-sealed system outside containment is considered to be an OPERABLE isolation valve for purposes of compliance with the initial ACTION statement of Specification 3.6.3 which states "maintain at least one isolation valve OPERABLE. . ." Further action under Specification 3.6.3 is still required to isolate the affected penetration or to shut down in accordance with Actions a, b, c, or d. Reopening of a valve declared INOPERABLE to comply with Tech Spec ACTIONS is allowed to permit surveillance testing to demonstrate its operability or the operability of other equipment per Specification 3.0.5. The following guidance is provided for complying with the follow-up action requirement, specified in Actions a, b, or c, to isolate the penetration: • Either the INOPERABLE valve must be closed (and de-activated, if applicable for power-operated valves), OR • A valve having the same safety class and seismic design class in series with the INOPERABLE valve must be closed (and de-activated, if applicable for power-operated valves). If the piping branches, each branch must be isolated. • Check valves may not be used to isolate a penetration beyond the four-hour period.

<< Containment Isolation Valves >>

TABLE NOTATIONS	
(9)	For relief valves, the "REDUNDANT VALVE(s)" column applies to the action statement whenever the relief is unable to isolate (that is, excessive leakage or failed open). When a relief is unable to open or cannot otherwise adequately relieve design overpressure conditions, the penetration must be isolated, and the penetration and closed system (if Table Notation 6 or 8 is applicable) drained to eliminate the potential for an over-pressurization event.
(10)	For this valve, no redundant valve or closed system is available for purposes of compliance with the initial ACTION statement of Specification 3.6.3 which states "maintain at least one isolation valve OPERABLE. . ." Immediate action to isolate the penetration must be initiated or Technical Specification 3.0.3 should be applied.
(11)	Deleted
(12)	Engineering review required when maximum isolation time exceeds 8 seconds to ensure the corresponding Engineered Safety Features Response Time meets the ten second requirement.
(13)	If the valve is INOPERABLE, perform Technical Specification Surveillance Requirement 4.6.1.1.a at least once every 31 days.

<< Containment Penetration Conductor Overcurrent Protective Devices >>

NOTE

The following general notes apply to this table:

- If a breaker that provides containment penetration conductor over-current protection (T.S. 3.8.4.1) is found tripped and there is no reason to suspect it is INOPERABLE as a protection device, then it is not necessary to enter T.S. 3.8.4.1. A breaker found in the tripped condition does not automatically make it INOPERABLE for this Technical Specification.
- If it is suspected that the breaker may not be capable of performing its over-current protection function, then it should be declared INOPERABLE per T.S. 3.8.4.1. For example, if a breaker is tripped free but the trip coil has burned up, then T.S. 3.8.4.1 must be entered.
- Amperage rating of the device is enclosed in parentheses ( ). Fuses are listed for table completeness only; there are no surveillance requirements. The number listed with a fuse is the control wiring diagram (2166-B-401 series) on which the fuse is shown. Circuits with fuses for both primary and secondary protection are not included. When the two protection devices are on separate load centers, the item is duplicated in the table.

NOTE

DC Control power is required for breakers to function. A relay trips the breaker.

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
	6900-V Breakers	
6.9-kV Bus 1A-NNS		
REACTOR COOLANT PUMP 1A-SN	1A-5:002 (REDUNDANT TRIP COILS)	1A-3:012 (UAT A to Aux Bus 1A) and 1A 1:010 (SUT A to Aux Bus 1A)
6.9-kV Bus 1B-NNS		
REACTOR COOLANT PUMP 1B-SN	1B-9:002 (REDUNDANT TRIP COILS)	1B-5:011 (UAT B to Aux Bus 1B) and 1B-3:012 (SUT B to Aux Bus 1B)
6.9-kV Bus 1C-NNS		
REACTOR COOLANT PUMP 1C-SN	1C-2:002 (REDUNDANT TRIP COILS)	1A-4:007 (Aux Bus 1A to Aux Bus 1C Tie Bkr) and 1B-10:007 (Aux Bus 1B to Aux Bus 1C Tie Bkr)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

NOTE
DC Control power is required for breakers to function.

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
	480-V Bus Breakers	
480-V Bus 1D1		
CONTAINMENT FAN COOLER AH-39 1A-NNS	1D1-1A:003 (1600 A w/200 A Sensor)	FUSE FU7/2715 (400 A) FU8/2715 (400 A) FU9/2715 (400 A)
CONTAINMENT FAN COOLER AH-38 1A-NNS	1D1-1B:003 (1600 A w/200 A Sensor)	FUSE FU7/2709 (400 A) FU8/2709 (400 A) FU9/2709 (400 A)
CONTAINMENT FAN COOLER AH-37 1A-NNS	1D1-1C:003 (1600 A w/200 A Sensor)	FUSE FU7/2703 (400 A) FU8/2703 (400 A) FU9/2703 (400 A)
480-V Bus 1E1		
CONTAINMENT FAN COOLER AH-37 1B-NNS	1E1-7A:003 (1600 A w/200 A Sensor)	FUSE FU7/2704 (400 A) FU8/2704 (400 A) FU9/2704 (400 A)
CONTAINMENT FAN COOLER AH-38 1B-NNS	1E1-7B:003 (1600 A w/200 A Sensor)	FUSE FU7/2710 (400 A) FU8/2710 (400 A) FU9/2710 (400 A)
CONTAINMENT FAN COOLER AH-39 1B-NNS	1E1-7C:003 (1600 A w/200 A Sensor)	FUSE FU7/2716 (400 A) FU8/2716 (400 A) FU9/2716 (400 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
	480-V Bus Breakers	
480-V Bus 1A3-SA		
CONT. FAN COOLER AH-3 (1A-SA)	1A34-SA-1A:002 (225 A) (300 A after completion of WO 2270769)	1A3-SA-4A:002 (1600 A)
CONT. FAN COOLER AH-3 (1B-SA)	1A34-SA-2A:002 (225 A) (300 A after completion of WO 2270772)	1A3-SA-4A:002 (1600 A)
CONT. FAN COOLER AH-2 (1B-SA)	1A22-SA-1A:002 (225 A) (300 A after completion of WO 2270762)	1A3-SA-4B:002 (1600 A)
CONT. FAN COOLER AH-2 (1A-SA)	1A22-SA-2A:002 (225 A) (300 A after completion of WO 2270765)	1A3-SA-4B:002 (1600 A)
480-V Bus 1B3-SB		
CONT. FAN COOLER AH-1 (1A-SB)	1B22-SB-2A:002 (225 A) (300 A after completion of WO 2270775)	1B3-SB-4C:002 (1600 A)
CONT. FAN COOLER AH-1 (1B-SB)	1B22-SB-3A:002 (225 A) (300 A after completion of WO 2270779)	1B3-SB-4C:002 (1600 A)
CONT. FAN COOLER AH-4 (1A-SB)	1B34-SB-2A:002 (225 A) (300 A after completion of WO 2270781)	1B3-SB-5A:002 (1600 A)
CONT. FAN COOLER AH-4 (1B-SB)	1B34-SB-3A:002 (225 A) (300 A after completion of WO 2270783)	1B3-SB-5A:002 (1600 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #		Secondary Protection EDBS Tag #
480-V Motor Control Centers and Power Panels			
PHPP-1A			
PRESSURIZER HEATERS GROUP A	FUSE	FU1/176 (100 A) FU2/176 (100 A) FU3/176 (100 A)	PHPP-1A-1 (90 A)
	FUSE	FU4/176 (100 A) FU5/176 (100 A) FU6/176 (100 A)	PHPP-1A-2 (90 A)
	FUSE	FU7/176 (100 A) FU8/176 (100 A) FU9/176 (100 A)	PHPP-1A-3 (90 A)
	FUSE	FU10/176 (100 A) FU11/176 (100 A) FU12/176 (100 A)	PHPP-1A-4 (90 A)
	FUSE	FU1/177 (100 A) FU2/177 (100 A) FU3/177 (100 A)	PHPP-1A-5 (90 A)
	FUSE	FU4/177 (100 A) FU5/177 (100 A) FU6/177 (100 A)	PHPP-1A-6 (90 A)
	FUSE	FU7/177 (100 A) FU8/177 (100 A) FU9/177 (100 A)	PHPP-1A-7 (90 A)
	FUSE	FU10/177 (100 A) FU11/177 (100 A) FU12/177 (100 A)	PHPP-1A-8 (90 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #		Secondary Protection EDBS Tag #
480-V Motor Control Centers and Power Panels			
PHPP-1B			
PRESSURIZER HEATERS GROUP B	FUSE	FU1/178 (100 A) FU2/178 (100 A) FU3/178 (100 A)	PHPP-1B-1 (90 A)
	FUSE	FU4/178 (100 A) FU5/178 (100 A) FU6/178 (100 A)	PHPP-1B-2 (90 A)
	FUSE	FU7/178 (100 A) FU8/178 (100 A) FU9/178 (100 A)	PHPP-1B-3 (90 A)
	FUSE	FU10/178 (100 A) FU11/178 (100 A) FU12/178 (100 A)	PHPP-1B-4 (90 A)
	FUSE	FU1/179 (100 A) FU2/179 (100 A) FU3/179 (100 A)	PHPP-1B-5 (90 A)
	FUSE	FU4/179 (100 A) FU5/179 (100 A) FU6/179 (100 A)	PHPP-1B-6 (90 A)
	FUSE	FU7/179 (100 A) FU8/179 (100 A) FU9/179 (100 A)	PHPP-1B-8 (90 A)
	FUSE	FU10/179 (100 A) FU11/179 (100 A) FU12/179 (100 A)	PHPP-1B-1 (90 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection	EDBS Tag #	Secondary Protection EDBS Tag #
480-V Motor Control Centers and Power Panels			
PHPP-1C			
PRESSURIZER HEATERS GROUP C	FUSE	FU1/174 (100 A) FU2/174 (100 A) FU3/174 (100 A)	PHPP-1C-1 (90 A)
	FUSE	FU4/174 (100 A) FU5/174 (100 A) FU6/174 (100 A)	PHPP-1C-2 (90 A)
	FUSE	FU7/174 (100 A) FU8/174 (100 A) FU9/174 (100 A)	PHPP-1C-3 (90 A)
	FUSE	FU10/174 (100 A) FU11/174 (100 A) FU12/174 (100 A)	PHPP-1C-4 (90 A)
	FUSE	FU1/175 (100 A) FU2/175 (100 A) FU3/175 (100 A)	PHPP-1C-5 (90 A)
	FUSE	FU4/175 (100 A) FU5/175 (100 A) FU6/175 (100 A)	PHPP-1C-6 (90 A)
PHPP-1D			
PRESSURIZER HEATERS GROUP D	FUSE	FU1/180 (100 A) FU2/180 (100 A) FU3/180 (100 A)	PHPP-1D-1 (90 A)
	FUSE	FU4/180 (100 A) FU5/180 (100 A) FU6/180 (100 A)	PHPP-1D-2 (90 A)
	FUSE	FU7/180 (100 A) FU8/180 (100 A) FU9/180 (100 A)	PHPP-1D-3 (90 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1A21-SA		
REACTOR SUPPORTS COOLING FAN S-4 1A-SA	1A21-SA-1D:002 (100 A)	1A21-SA-1D:005 (100 A)
1CC-249 CCW FROM RCP THERMAL BARRIER HXS	1A21-SA-3C:002 (15 A)	1A21-SA-3C:006 (15 A)
1SI-248 ACCUMULATOR C DISCHARGE	1A21-SA-3D:002 (40 A)	1A21-SA-3D:003 (40 A)
1CS-470 RCP SEAL WTR RETURN 1A21-SA-4B:002 (15A) 1A21-SA-4B:005 (15 A) & EXCESS LTDN	1A21-SA-4B:002 (15 A)	1A21-SA-4B:005 (15 A)
PRIMARY SHIELD COOLING FAN S-2 1A-SA	1A21-SA-4C:002 (100 A)	1A21-SA-4C:003 (100 A)
1CC-297 CCW FROM RCP BEARING OIL HXS	1A21-SA-5B:002 (15 A)	1A21-SA-5B:005 (15 A)
1SI-246 ACCUMULATOR A DISCHARGE	1A21-SA-5C:002 (40 A)	1A21-SA-5C:003 (40 A)
1ED-94 CNMT SUMP PUMP DISCH	1A21-SA-7A:002 (15 A)	1A21-SA-7A:005 (15 A)
1RH-2 RHR PUMP A SUCTION FROM HOT LEG	1A21-SA-7B:002 (15 A)	1A21-SA-7B:003 (15 A)
1SI-300 CONTAINMENT SUMP TO RHR PUMP A	1A21-SA-7C:002 (30 A)	1A21-SA-7C:003 (30 A)
1RH-40 RHR PUMP B SUCTION FROM HOT LEG	1A21-SA-8A:002 (15 A)	1A21-SA-8A:003 (15 A)
1CT-105 CONTAINMENT SUMP TO CT PUMP A	1A21-SA-8C:002 (15 A)	1A21-SA-8C:005 (15 A)
LIGHTING PANEL LP-106	1A21-SA-14ER:002 (50A)	1A21-SA-14EL:002 (50A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1A21-SA (continued)		
1RH-1 ALTERNATE POWER SUPPLY	1A21-SA-6A:002 (15 A)	1A21-SA-6A:003 (15A)
480-V MCC 1A22-SA		
CONTAINMENT FAN COOLER AH-2 1B-SA	1A22-SA-1A:002 (225 A) (300 A after completion of WO 2270762-03)	1A3-SA-4B:002 (1600 A)
CONTAINMENT FAN COOLER AH-2 1A-SA	1A22-SA-2A:002 (225 A) (300 A after completion of WO 2270765-03)	
480-V MCC 1A24		
1RC-113 PRESSURIZER PORV BLOCK	1A24-3A:002 (15 A)	1A24-3A:005 (15 A)
RCP 1A-SN BEARING OIL LIFT PUMP	1A24-6A:002 (30 A)	1A24-6A:005 (30 A)
RCP 1C-SN BEARING OIL LIFT PUMP	1A24-6B:002 (30 A	1A24-6B:005 (30 A)
CRDM FAN E-80 1A-NNS	1A24-7B:002 (100 A)	1A24-7B:003 (100 A)
CRDM FAN E-81 1A-NNS	1A24-7C:002 (100 A)	1A24-7C:003 (100 A)
480-V MCC 1A34-SA		
CONTAINMENT FAN COOLER AH-3 1A-SA	1A34-SA-1A:002 (225 A) (300 A after completion of WO 2270769-03)	1A3-SA-4B:002 (1600 A)
CONTAINMENT FAN COOLER AH-3 1B-SA	1A34-SA-2A:002 (225 A) (300 A after completion of WO 2270772-03)	1A3-SA-4A:002 (1600 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1B21-SB		
PRIMARY SHIELD COOLING FANS-2 1B-SB	1B21-SB-4B:002 (100 A)	1B21-SB-4B:003 (100 A)
REACTOR SUPPORTS COOLING FAN S-4 1B-SB	1B21-SB-4C:002 (100 A)	1B21-SB-4C:004 (100 A)
1RH-1 RHR PUMP A SUCTION FROM HOT LEG	1B21-SB-5B:002 (15 A)	1B21-SB-5B:003 (15 A)
1SI-247 ACCUMULATOR B DISCHARGE	1B21-SB-5C:002 (40 A)	1B21-SB-5C:003 (40 A)
1CT-102 CONTAINMENT SUMP TO CT PUMP B	1B21-SB-6A:005 (15 A)	1B21-SB-6A:002 (15 A)
1RH-39 RHR PUMP B SUCTION FROM HOT LEG	1B21-SB-11A:002 (15 A)	1B21-SB-11A:003 (15 A)
RHR ISOL VA 1-8701B ALTERNATE PWR SUPPLY FROM AN "SB" SOURCE	1B21-SB-8B:002 (15 A)	1B21-SB-8B:006 (15 A)
1SI-301 CONTAINMENT SUMP TO RHR PUMP B	1B21-SB-11B:002 (30 A)	1B21-SB-11B:003 (30 A)
LIGHTING PANEL LP-107	1B21-SB-13AR:002 (50A)	1B21-SB-13AL:002 (50 A)
480-V MCC 1B22-SB		
CONTAINMENT FAN COOLER AH-1 1A-SB	1B22-SB-2A:002 (225 A) (300 A after completion of WO 2270775-03)	1B3-SB-4C:002 (1600 A)
CONTAINMENT FAN COOLER AH-1 1B-SB	1B22-SB-3A:002 (225 A) (300 A after completion of WO 2270779-03)	1B3-SB-4C:002 (1600 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1B24		
CRDM FAN E-80 1B-NNS	1B24-1B:002 (100 A)	1B24-1B:003 (100 A)
CRDM FAN E-81 1B-NNS	1B24-2B:002 (100 A)	1B24-2B:003 (100 A)
1RC-115 PRESSURIZER PORV BLOCK	1B24-2C:002 (15 A)	1B24-2C:003 (15 A)
1RC-117 PRESSURIZER PORV BLOCK	1B24-3B:002 (15 A)	1B24-3B:003 (15 A)
RCP 1B-SN BEARING OIL LIFT PUMP	1B24-4B:002 (30 A)	1B24-4B:005 (30 A)
480-V MCC 1B34-SB		
CONTAINMENT FAN COOLER AH-4 1A-SB	1B34-SB-2A:002 (225 A) (300 A after completion of WO 2270781-03)	1B3-SB-5A:002 (1600 A)
CONTAINMENT FAN COOLER AH-4 1B-SB	1B34-SB-3A:002 (225 A) (300 A after completion of WO 2270783-03)	1B3-SB-5A:002 (1600 A)
480-V MCC 1D11		
CNMT BLDG SUMP PUMP 1A-NNS	1D11-1B:002 (50 A)	1D11-1B:003 (50 A)
POWER RECEPTACLES 1-1 & 1-5	1D11-1C:002 (60 A)	1D11-1C:003 (60 A)
POWER RECEPTACLES 1-9 & 1-13	1D11-1D:002 (60 A)	1D11-1D:003 (60 A)
POWER RECEPTACLES 1-2 & 1-6	1D11-1E:002 (60 A)	1D11-1E:003 (60 A)
AIRBORNE RAD REMOVAL FAN S-1 1A-NNS	1D11-2B:002 (90 A)	1D11-2B:003 (90 A)
CNMT BLDG ELEVATOR	1D11-2C:002 (100 A)	1D11-2C:003 (100 A)
CNMT ACCESS HOIST 5-TON MONORAIL	1D11-2D:002 (50 A)	1D11-2D:003 (50 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1D11 (continued)		
POWER RECEPTACLES 1-18 & 1-75	1D11-3C:002 (60 A)	1D11-3C:003 (60 A)
CNMT JIB CRANE RECEPTACLES	1D11-3D:002 (60 A)	1D11-3D:003 (60 A)
POWER RECEPTACLES 1-10 & 1-14	1D11-4A:002 (60 A)	1D11-4A:003 (60 A)
REACTOR COOLANT DRAIN TANK PUMP 1A-NNS	1D11-4C:002 (90 A)	1D11-4C:003 (90 A)
POWER RECEPTACLE 1-76	1D11-4DR:002 (60 A)	1D11-4DL:002 (60 A)
LIGHTING PANEL LP-105	LP-105:003 (150 A)	1D11-4EL:002 (70 A)
LIGHTING PANEL LP-101	LP-101:003 (125 A)	1D11-4ER:002 (60 A)
LIGHTING PANEL LP-102	LP-102:003 (125 A)	1D11-5DL:002 (60 A)
POWER RECEPTACLES 1-17 & 1-74	1D11-5E:002 (60 A)	1D11-5E:003 (60 A)
CNMT POLAR BRIDGE CRANE	1D11-6D:002 (225 A)	1D11-6D:003 (225 A)
480-V MCC 1D12		
IRVH CABLE BRIDGE HOIST	1D12-7BR:002 (15 A)	1D12-7BL:002 (15 A)
480-V MCC 1E11		
FUEL TRANSFER MANIPULATOR CRANE	1E11-1A:002 (30 A)	1E11-1A:003 (30 A)
FUEL TRANSFER RX SIDE CONTROL (PUMP MOTOR)	1E11-1B:002 (15 A)	1E11-1B:003 (15 A)
AIRBORNE RAD REMOVAL FAN S-1 1B-NNS	1E11-1E:002 (90 A)	1E11-1E:003 (90 A)
REACTOR COOLANT DRAIN TANK PUMP 1B-NNS	1E11-2E:002 (90 A)	1E11-2E:003 (90 A)
POWER RECEPTACLES 1-77 & 1-78	1E11-3BR:002 (60 A)	1E11-3BL:002 (60 A)
LIGHTING PANEL LP-103	LP-103:003 (110 A)	1E11-3DL:002 (50 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
480-V MCC 1E11 (continued)		
LIGHTING PANEL LP-123	LP-123:003 (125 A)	1E11-3DR:002 (60 A)
LIGHTING PANEL LP-104	LP-104:003 (150 A)	1E11-4CR:002 (70 A)
CNMT BLDG SUMP PUMP 1B-NNS	1E11-5C:002 (50 A)	1E11-5C:003 (50 A)
INCORE INSTRUMENT DRIVE ASSEMBLES	1E11-5D:002 (15 A)	1E11-5D:003 (15 A)
POWER RECEPTACLES 1-12 & 1-16	1E11-6AR:002 (60 A)	1E11-6AL:002 (60 A)
POWER RECEPTACLES 1-4 & 1-8	1E11-6BR:002 (60 A)	1E11-6BL:002 (60 A)
POWER RECEPTACLES 1-3 & 1-7	1E11-6CR:002 (60 A)	1E11-6CL:002 (60 A)
POWER RECEPTACLES 1-11 & 1-15	1E11-6DR:002 (60 A)	1E11-6DL:002 (60 A)
IRVH STUD TENSIONER HOIST (B1263)	1E11-6ER:002 (15 A)	1E11-6EL:002 (15 A)
RCCA CHANGE FIXTURE HOIST	1E11-6FR:002 (15 A)	1E11-6FL:002 (15 A)
120-V Breakers		

Motor Space Heaters with Breakers in 6900-V and 480-V Bus Breakers

NOTE: The primary protection and/or secondary protection for the fan heaters are breakers located at the bus breaker cubicle corresponding to the listed equipment.

Reactor Coolant Pumps

RCP 1A-SN MOTOR SPACE HEATER	1A-5:028 (15 A)	PP-1D213-4 (30 A/phase)
RCP 1B-SN MOTOR SPACE HEATER	1B-9:030 (15 A)	PP-1E213-4 (30 A/phase)
RCP 1C-SN MOTOR SPACE HEATER	1C-2:031 (15 A)	PP-1D213-10 (20 A/phase)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Containment Fan Coil Units			
AH-37 1A-NNS MOTOR SPACE HEATER	PP-1D213-40-1 (15 A)		PP-1D213-40-2 (15 A)
AH-38 1A-NNS MOTOR SPACE HEATER	PP-1D213-40-3 (15 A)		PP-1D213-40-4 (15 A)
AH-39 1A-NNS MOTOR SPACE HEATER	PP-1D213-40-5 (15 A)		PP-1D213-40-6 (15 A)
AH-37 1B-NNS MOTOR SPACE HEATER	PP-1E213-22-1 (15 A)		PP-1E213-22-2 (15 A)
AH-38 1B-NNS MOTOR SPACE HEATER	PP-1E213-22-3 (15 A)		PP-1E213-22-4 (15 A)
AH-39 1B-NNS MOTOR SPACE HEATER	PP-1E213-22-5 (15 A)		PP-1E213-22-6 (15 A)
PP-1 (Rod Position Indication)			
DRPI PANELS 1A AND 1B POWER	PP-1:002 (100 A)	and	PP-1E211-18 (100 A)
		OR	
	PP-1:003 (100 A)	and	PP-1D211-18 (100 A)
PP-1A211-SA			
AH-2 DAMPER CV-D3SA	FUSE-L1/2674 (6 A)		PP-1A211-SA-7 (20 A)
AH-3 DAMPER CV-D5SA	FUSE-L3/2742 (6 A)		PP-1A211-SA-7 (20 A)
AH-2 1A-SA MOTOR SPACE HEATER	PP-1A211-SA-35 (15 A)		PP-1A211-SA-33 (15 A)
AH-2 1B-SA MOTOR SPACE HEATER	PP-1A211-SA-36 (15 A)		PP-1A211-SA-34 (15 A)
PP-1A212-SA			
AH-3 1A-SA MOTOR SPACE HEATER	PP-1A212-SA-9 (15 A)		PP-1A212-SA-7 (15 A)
AH-3 1B-SA MOTOR SPACE HEATER	PP-1A212-SA-10 (15 A)		PP-1A212-SA-8 (15 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
PP-1A212-SA (continued)		
CNMT HYDROGEN ANALYZER SA VALVES	PP-1A212-SA-16 (20 A)	PP-1A212-SA-14 (20 A)
S-4 1A-SA MOTOR SPACE HEATER	PP-1A212-SA-39 (15 A)	PP-1A212-SA-37 (15 A)
S-2 1A-SA MOTOR SPACE HEATER	PP-1A212-SA-40 (15 A)	PP-1A212-SA-38 (15 A)
INLET ISOL VA 1-8701A INDICATING LIGHT CIRCUIT	Fuse FU3/325 (3 A)	PP-1A212-SA-30 (15 A)
INLET ISOL VA 1-8701B INDICATING LIGHT CIRCUIT	Fuse FU3/326 (3 A)	PP-1A212-SA-32 (15 A)
INLET ISOL VA 1-8702A INDICATING LIGHT CIRCUIT	Fuse FU3/336 (3 A)	PP-1A212-SA-34 (15 A)
PP-1A311-SA		
1RC-900 RX VESSEL HEAD VENT	Fuse-L3/134 (6 A)	PP-1A311-SA-4 (20 A)
1RC-902 PRZ STEAM SPACE VENT	Fuse-L4/136 (6 A)	PP-1A311-SA-4 (20 A)
1SP-12 & 941 HYDROGEN ANALYZER A	Fuse-J1/701 (6 A)	PP-1A311-SA-4 (20 A)
1SP-916 & 918 RADIATION MONITOR REM-3502A	Fuse-L1/703 (6 A)	PP-1A311-SA-4 (20 A)
1SP-944 RCS HOT LEG LOOP C SAMPLE	Fuse-L4/2435 (6 A)	PP-1A311-SA-4 (20 A)
1CP-5 & 9 CNMT NORMAL PURGE DISCHARGE INLET	Fuse-L4/2706 (6 A)	PP-1A311-SA-14 (20 A)
1CP-10 & 4 CNMT P/E PURGE INLET/DISCHARGE	Fuse-L5/2713 (6 A)	PP-1A311-SA-14 (20 A)
1CM-2 HYDROGEN PURGE INLET	Fuse-L5/3074 (6 A)	PP-1A311-SA-14 (20 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
PP-1A312-SA		
1RC-904 RVH & PRZ VENT TO ATMOS	Fuse-FU5/137 (6 A)	PP-1A312-SA-11 (20 A)
PP-1B211-SB		
AH-1 DAMPER CV-D1SB	Fuse-L1/2670 (6 A)	PP-1B211-SB-14 (20 A)
AH-4 DAMPER CV-D7SB	Fuse-L6/2746 (6 A)	PP-1B211-SB-14 (20 A)
AH-1 1A-SB MOTOR SPACE HEATER	PP-1B211-SB-35 (15 A)	PP-1B211-SB-33 (15 A)
AH-1 1B-SB MOTOR SPACE HEATER	PP-1B211-SB-36 (15 A)	PP-1B211-SB-34 (15 A)
PP-1B212-SB		
AH-4 1A-SB MOTOR SPACE HEATER	PP-1B212-SB-9 (15 A)	PP-1B212-SB-7 (15 A)
AH-4 1B-SB MOTOR SPACE HEATER	PP-1B212-SB-10 (15 A)	PP-1B212-SB-8 (15 A)
CNMT HYDROGEN ANALYZER SB VALVES	PP-1B212-16 (20 A)	PP-1B212-SB-14 (20 A)
S-4 1B-SB MOTOR SPACE HEATER	PP-1B212-SB-39 (15 A)	PP-1B212-SB-37 (15 A)
S-2 1B-SB MOTOR SPACE HEATER	PP-1B212-SB-40 (15 A)	PP-1B212-SB-38 (15 A)
INLET ISOL VA 1-8702A INDICATING LIGHT CIRCUIT	Fuse FU3/327 (3 A)	PP-1B212-SB-30 (15 A)
INLET ISOL VA 1-8702B INDICATING LIGHT CIRCUIT	Fuse FU3/328 (3 A)	PP-1B212-SB-32 (15 A)
INLET ISOL VA 1-8701B INDICATING LIGHT CIRCUIT	Fuse FU3/337 (3 A)	PP-1B212-SB-34 (15 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #		Secondary Protection EDBS Tag #
PP-1B311-SB			
AH-1 DAMPER CV-D1SB & CV-D2SB REMOTE SHUTDOWN POWER)	Fuse-FU1A-15/2670 (5 A)		PP-1B311-SB-16 (20 A)
AH-4 DAMPER CV-D7SB & CV-D8SB REMOTE SHUTDOWN POWER)	Fuse-1A-16/2746 (5 A)		PP-1B311-SB-16 (20 A)
PP-1B312-SB			
1RC-905 RVH & PRZ VENT TO ATMOS	Fuse-FU21/138 (6 A)		PP-1B312-SB-11 (20 A)
PP-1D121			
INST. RACK C1-R1 (CNMT COOLER SW TEMP)	PP-1D121-13 (20 A)		PP-1D121-13-BU (20 A)
INST. RACK C1-R-18	PP-1D121-16 (15 A)		PP-1D121-16-BU (15 A)
PP-1D211			
ELEVATOR EQUIP RM EXHAUST FAN E-3 1X-NNS	PP-1D211-2 (20 A)		PP-1D211-2BU (20 A)
DRPI PANELS 1A AND 1B POWER	PP-1:002 (100 A)	and	PP-1E211-18 (100 A)
		OR	
	PP-1:003 (100 A)	and	PP-1D211-18 (100 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #		Secondary Protection EDBS Tag #
PP-1D212			
CNMT FAN COOLERS (NNS) DAMPERS CV-D9, D11 & D13	Fuse-L1/2728 (6 A)		PP-1D212-8 (15 A)
CNMT FAN COOLERS (NNS) DAMPERS CV-D10, D12 & D14	Fuse-L2/2729 (6 A)		PP-1D212-8 (15 A)
1SI-246 & 248 ACCUMULATORS A&C ALARM CKT	Fuse-FU-41/448 (3 A)		PP-1D212-16 (15 A)
1SI-247 ACCUMULATOR B ALARM CKT	Fuse-FU-41/447 (3 A)		PP-1D212-18 (15 A)
PP-1D213			
RCP 1A-SN MOTOR SPACE HEATER	1A-5:028 (15 A)		PP-1D213-4 (30 A/phase)
RCP 1C-SN MOTOR SPACE HEATER	1C-2:031 (15 A)		PP-1D213-10 (20 A/phase)
PP-1E121			
FUEL TRANSFER CONSOLE REACTOR SIDE POWER	PP-1E121-1 (20 A)		PP-1E121-1-BU (20 A)
PP-1E211			
E-81 1B-NNS MOTOR SPACE HEATER	PP-1E211-10 (20 A)		PP-1E211-10-BU (20 A)
E-80 1B-NNS MOTOR SPACE HEATER	PP-1E211-17 (20 A)		PP-1E211-17-BU (20 A)
DRPI PANELS 1A AND 1B POWER	PP-1:002 (100 A)	and	PP-1E211-18 (100 A)
		OR	
	PP-1:003 (100 A)	and	PP-1D211-18 (100 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
PP-1E212		
E-80 1A-NNS CRDM FAN DAMPER	Fuse-L1/472 (6 A)	PP-1E212-10 (15 A)
E-80 1B-NNS CRDM FAN DAMPER	Fuse-L2/473 (6 A)	PP-1E212-10 (15 A)
E-81 1A-NNS CRDM FAN DAMPER	Fuse-L3/474 (6 A)	PP-1E212-10 (15 A)
E-81 1B-NNS CRDM FAN DAMPER	Fuse-L4/475 (6 A)	PP-1E212-10 (15 A)
S-1 1A-NNS FLOW SW FIS-1AR-7647A	Fuse-L1/2661 (6 A)	PP-1E212-10 (15 A)
S-1 1B-NNS FLOW SW FIS-1AR-7647B	Fuse-L2/2662 (6 A)	PP-1E212-10 (15 A)
S-1 1A-NNS DAMPER AR-D3	Fuse-L3/2663 (6 A)	PP-1E212-10 (15 A)
S-1 1B-NNS DAMPER AR-D4	Fuse-L4/2664 (6 A)	PP-1E212-10 (15 A)
S-1 1A-NNS TEMP DET TAS-1AR-7644AV	Fuse-L2/2677 (6 A)	PP-1E212-10 (15 A)
S-1 1B-NNS TEMP DET TAS-1AR-7644BV	Fuse-L3/2678 (6 A)	PP-1E212-10 (15 A)
PP-1E213		
RCP 1B-SN MOTOR SPACE HEATER	1B-9:030 (15 A)	PP-1E213-4 (30 A/phase)
PP-1-4A10221		
S-1 1B-NNS MOISTURE DET MIS-1AR-7644BV	PP-1-4A10221-2 (20 A)	PP-1-4A10221-2-BU (20 A)
PP-1-4A111		
S-1 1A-NNS MOISTURE DET MIS-1AR-7644AV	PP-1-4A111-36 (15 A)	PP-1-4A111-34 (15 A)

<< Containment Penetration Conductor Overcurrent Protective Devices >>

Equipment Description	Primary Protection EDBS Tag #	Secondary Protection EDBS Tag #
PP-1-4B241		
Deleted		

<< Motor-Operated Valves Thermal Overload Protection Bypass >>

VALVE NO. CP&L (EBASCO)	FUNCTION
1AF-55 (AF-V10)	AUXILIARY FEEDWATER TO SG A *
1AF-74 (AF-V23)	AUXILIARY FEEDWATER TO SG C *
1AF-93 (AF-V19)	AUXILIARY FEEDWATER TO SG B *
1AF-137 (AF-V116)	AUXILIARY FEEDWATER TO SG A *
1AF-143 (AF-V117)	AUXILIARY FEEDWATER TO SG B *
1AF-149 (AF-V118)	AUXILIARY FEEDWATER TO SG C *
1CC-99 (CC-B19)	CCW A NONESSENTIAL SUPPLY ISOL
1CC-113 (CC-B20)	CCW B NONESSENTIAL SUPPLY ISOL
1CC-127 (CC-B6)	CCW A NONESSENTIAL RETURN ISOL
1CC-128 (CC-B5)	CCW B NONESSENTIAL RETURN ISOL
1CC-147 (CC-V165)	RHR HX A CCW OUTLET
1CC-167 (CC-V167)	RHR HX B CCW OUTLET
1CC-176 (CC-V172)	CCW TO RCDT & EXCESS LTDN HEAT EXCHS
1CC-202 (CC-V182)	CCW FROM RCDT & EXCESS LTDN HEAT EXCHS
1CC-207 (CC-V169)	CCW TO RCPS
1CC-208 (CC-V170)	CCW TO RCPS
1CC-249 (CC-V191)	CCW FROM RCP THERMAL BARRIER HXS
1CC-251 (CC-V190)	CCW FROM RCP THERMAL BARRIER HXS
1CC-297 (CC-V184)	CCW FROM RCP BEARING OIL HXS
1CC-299 (CC-V183)	CCW FROM RCP BEARING OIL HXS
1CS-165 (CS-L520)	VCT TO CSIPS ISOLATION
1CS-166 (CS-L521)	VCT TO CSIPS ISOLATION
1CS-168 (CS-V588)	CSIP SUCTION ISOLATION
1CS-169 (CS-V589)	CSIP SUCTION ISOLATION
1CS-170 (CS-V587)	CSIP SUCTION ISOLATION
1CS-171 (CS-V590)	CSIP SUCTION ISOLATION
1CS-182 (CS-V600)	CSIP A MINIFLOW ISOLATION
1CS-196 (CS-V602)	CSIP B MINIFLOW ISOLATION
1CS-210 (CS-V601)	CSIP C MINIFLOW ISOLATION
1CS-214 (CS-V585)	CSIPS MINIFLOW ISOLATION
1CS-217 (CS-V604)	CSIP DISCHARGE ISOLATION
1CS-218 (CS-V605)	CSIP DISCHARGE ISOLATION
1CS-219 (CS-V603)	CSIP DISCHARGE ISOLATION
1CS-220 (CS-V606)	CSIP DISCHARGE ISOLATION
1CS-235 (CS-V609)	CVCS NORMAL CHARGING
1CS-238 (CS-V610)	CVCS NORMAL CHARGING
1CS-240 (CS-V611)	RCP SEAL WATER INJECTION
1CS-278 (CS-V586)	BORIC ACID PUMPS TO CSIPS
1CS-291 (CS-L523)	RWST TO CSIPS ISOLATION

<< Motor-Operated Valves Thermal Overload Protection Bypass >>

VALVE NO. CP&L (EBASCO)	FUNCTION
1CS-292 (CS-L522)	RWST TO CSIPS ISOLATION
1CS-341 (CS-V522)	CVCS SEAL WATER TO RCP A
1CS-382 (CS-V523)	CVCS SEAL WATER TO RCP B
1CS-423 (CS-V524)	CVCS SEAL WATER TO RCP C
1CS-470 (CS-V516)	CVCS SEAL WATER RETURN & EXCESS LTDN
1CS-472 (CS-V517)	CVCS SEAL WATER RETURN & EXCESS LTDN
1CS-745 (CS-V758)	CSIP MINIFLOW TO RWST
1CS-746 (CS-V757)	CSIP MINIFLOW TO RWST *
1CS-752 (CS-V759)	CSIP MINIFLOW TO RWST *
1CS-753 (CS-V760)	CSIP MINIFLOW TO RWST
1CT-11 (CT-V88)	NAOH ADDITIVE ISOLATION
1CT-12 (CT-V85)	NAOH ADDITIVE ISOLATION
1CT-24 (CT-V8)	CNMT SPRAY PUMP A EDUCTOR TEST
1CT-26 (CT-V2)	CNMT SPRAY PUMP A RWST SUPPLY
1CT-25 (CT-V145)	CNMT SPRAY PUMP B EDUCTOR TEST
1CT-47 (CT-V25)	CNMT SPRAY HDR A RECIRC
1CT-50 (CT-V21)	CONTAINMENT SPRAY A
1CT-71 (CT-V3)	CNMT SPRAY PUMP B RWST SUPPLY
1CT-88 (CT-V43)	CONTAINMENT SPRAY B
1CT-95 (CT-V49)	CNMT SPRAY HDR B RECIRC
1CT-102 (CT-V7)	CONTAINMENT SUMP TO CT PUMP B
1CT-105 (CT-V6)	CONTAINMENT SUMP TO CT PUMP A
1ED-94 (MD-V36)	CNMT SUMP PUMPS DISCH
1ED-95 (MD-V77)	CNMT SUMP PUMPS DISCH
1MS-70 (MS-V8)	MAIN STEAM B TO AUXILIARY FW TURBINE*
1MS-72 (MS-V9)	MAIN STEAM C TO AUXILIARY FW TURBINE*
1RH-1 (RH-V502)	RHR PUMP A SUCTION FROM HOT LEG
1RH-2 (RH-V503)	RHR PUMP A SUCTION FROM HOT LEG
1RH-25 (RH-V507)	RHR TO CSIP SUCTION
1RH-31 (RH-F513)	RHR PUMP A MINIFLOW
1RH-39 (RH-V500)	RHR PUMP B SUCTION FROM HOT LEG
1RH-40 (RH-V501)	RHR PUMP B SUCTION FROM HOT LEG
1RH-63 (RH-V506)	RHR TO CSIP SUCTION
1RH-69 (RH-F512)	RHR PUMP B MINIFLOW
1SI-3 (SI-V505)	SI-HIGH HEAD TO COLD LEGS
1SI-4 (SI-V506)	SI-HIGH HEAD TO COLD LEGS
1SI-52 (SI-V502)	SI-HIGH HEAD TO COLD LEGS
1SI-86 (SI-V501)	SI-HIGH HEAD TO HOT LEGS
1SI-107 (SI-V500)	SI-HIGH HEAD TO HOT LEGS
1SI-246 (SI-V537)	ACCUMULATOR A DISCHARGE ISOLATION

<< Motor-Operated Valves Thermal Overload Protection Bypass >>

VALVE NO. CP&L (EBASCO)	FUNCTION
1SI-247 (SI-V536)	ACCUMULATOR B DISCHARGE ISOLATION
1SI-248 (SI-V535)	ACCUMULATOR C DISCHARGE ISOLATION
1SI-300 (SI-V571)	CONTAINMENT SUMP TO RHR PUMP A
1SI-301 (SI-V570)	CONTAINMENT SUMP TO RHR PUMP B
1SI-310 (SI-V573)	CONTAINMENT SUMP TO RHR PUMP A
1SI-311 (SI-V572)	CONTAINMENT SUMP TO RHR PUMP B
1SI-322 (SI-V575)	RWST TO RHR PUMP A ISOL
1SI-323 (SI-V574)	RWST TO RHR PUMP B ISOLATION
1SI-326 (SI-V577)	SI-LOW HEAD CROSS CONNECT
1SI-327 (SI-V576)	SI-LOW HEAD CROSS CONNECT
1SI-340 (SI-V579)	SI-LOW HEAD TO COLD LEGS
1SI-341 (SI-V578)	SI-LOW HEAD TO COLD LEGS
1SI-359 (SI-V587)	SI-LOW HEAD TO HOT LEGS
1SW-39 (SW-B5)	NORMAL SW HDR A SUPPLY ISOLATION
1SW-40 (SW-B6)	NORMAL SW HDR B SUPPLY ISOLATION
1SW-91 (SW-B45)	SERVICE WATER TO FAN COOLER AH-2
1SW-92 (SW-B46)	SERVICE WATER TO FAN COOLER AH-3
1SW-97 (SW-B47)	SERVICE WATER FROM FAN COOLER AH-3
1SW-98 (SW-B48)	SERVICE WATER FROM FAN COOLER AH-1
1SW-109 (SW-B49)	SERVICE WATER FROM FAN COOLER AH-2
1SW-110 (SW-B50)	SERVICE WATER FROM FAN COOLER AH-4
1SW-121 (SW-B74)	SW HDR A TO AFW PUMP A
1SW-123 (SW-B75)	SW HDR A TO AFW PUMP A
1SW-124 (SW-B70)	SW HDR A TO AFWTD PUMP
1SW-126 (SW-B71)	SW HDR A TO AFWTD PUMP
1SW-127 (SW-B72)	SW HDR B TO AFWTD PUMP
1SW-129 (SW-B73)	SW HDR B TO AFWTD PUMP
1SW-130 (SW-B76)	SW HDR B TO AFW PUMP B
1SW-132 (SW-B77)	SW HDR B TO AFW PUMP B
1SW-225 (SW-B52)	SERVICE WATER TO FAN COOLER AH-1
1SW-227 (SW-B51)	SERVICE WATER TO FAN COOLER AH-4
1SW-270 (SW-B15)	SW HDR A TO AUX RSVR ISOLATION
1SW-271 (SW-B16)	SW HDR B TO AUX RSVR ISOLATION
1SW-274 (SW-B14)	NORMAL SW HDR B RETURN ISOLATION
1SW-275 (SW-B13)	NORMAL SW HDR A RETURN ISOLATION
1SW-276 (SW-B8)	NORMAL SW HDR RETURN ISOLATION
1AV-11 (AC-B1)	RAB SWGR A EXHAUST
1AV-12 (AC-B2)	RAB SWGR B EXHAUST
1AV-13 (AC-B3)	RAB SWGR B EXHAUST
1AV-1 (AV-B1)	RAB EMER EXHAUST A INLET

<< Motor-Operated Valves Thermal Overload Protection Bypass >>

VALVE NO. CP&L (EBASCO)	FUNCTION
1AV-2 (AV-B2)	RAB EMER EXHAUST A OUTLET
1AV-3 (AV-B3)	RAB EMER EXHAUST A BLEED**
1AV-4 (AV-B4)	RAB EMER EXHAUST B INLET
1AV-5 (AV-B5)	RAB EMER EXHAUST B OUTLET
1AV-6 (AV-B6)	RAB EMER EXHAUST B BLEED**
1CZ-1 (CZ-B1)	CONTROL ROOM NORMAL SUPPLY ISOLATION*
1CZ-2 (CZ-B2)	CONTROL ROOM NORMAL SUPPLY ISOLATION*
1CZ-3 (CZ-B3)	CONTROL ROOM NORMAL EXHAUST ISOLATION*
1CZ-4 (CZ-B4)	CONTROL ROOM NORMAL EXHAUST ISOLATION*
1CZ-5 (CZ-B5)	RAB ELEC PROT INLET*
1CZ-6 (CZ-B6)	RAB ELEC PROT INLET*
1CZ-7 (CZ-B7)	RAB ELEC PROT EXHAUST*
1CZ-8 (CZ-B8)	RAB ELEC PROT EXHAUST*
1CZ-9 (CZ-B9)	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
1CZ-10 (CZ-B10)	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
1CZ-11 (CZ-B11)	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
1CZ-12 (CZ-B12)	CNTL RM EMER FLTR OUTSIDE AIR INTAKE*
1CZ-13 (CZ-B13)	CONTROL ROOM PURGE EXHAUST*
1CZ-14 (CZ-B14)	CONTROL ROOM PURGE EXHAUST*
1CZ-17 (CZ-B17)	CONTROL ROOM PURGE MAKE UP*
1CZ-18 (CZ-B18)	CONTROL ROOM PURGE MAKE UP*
1CZ-19 (CZ-B19)	CONTROL ROOM EMER FLTR DISCHARGE*
1CZ-20 (CZ-B20)	CONTROL ROOM EMER FLTR DISCHARGE*
1CZ-21 (CZ-B21)	CONTROL ROOM EMER FLTR DISCHARGE*
1CZ-22 (CZ-B22)	CONTROL ROOM EMER FLTR DISCHARGE*
1CZ-23 (CZ-B23)	CONTROL ROOM EMER FLTR A INLET*
1CZ-24 (CZ-B24)	CONTROL ROOM EMER FLTR B INLET*
1CZ-25 (CZ-B25)	CONTROL ROOM NORMAL SUPPLY A DISCHARGE*
1CZ-26 (CZ-B26)	CONTROL ROOM NORMAL SUPPLY B DISCHARGE*
1CZ-32 (CZ-B32)	RAB ELEC PROT PURGE MAKE-UP
1CZ-33 (CZ-B33)	RAB ELEC PROT PURGE MAKE-UP
1CZ-34 (CZ-B34)	RAB ELEC PROT PURGE INLET
1CZ-35 (CZ-B35)	RAB ELEC PROT PURGE INLET
1FV-2 (FV-B2)	FUEL HANDLING EXHAUST A INLET*
1FV-4 (FV-B4)	FUEL HANDLING EXHAUST B INLET*

<< Motor-Operated Valves Thermal Overload Protection Bypass >>

TABLE NOTATIONS	
*	Thermal Overload Protection Bypass for these valves is accomplished by contacts on the safeguards actuation circuit relays. These actuation relays are tested as a part of the Engineered Safety Features Actuation System instrumentation in accordance with the requirements of Technical Specification Table 4.3-2, or as part of FHB Emergency Exhaust System in accordance with the requirements of Specification 4.9.12.d.2. Verification of the Thermal Overload Protection Bypass circuit for each of these MOVs requires stroking the valve during safeguards actuation testing with normal control circuit power interrupted to simulate a thermal overload condition.
**	Valves are functional; however their ability to open and close is no longer required for system operability due to the blind flanges installed by EC 260241.

ATTACHMENT 8
Page 1 of 2

<< Safeguards Systems Isolation Valves in Closed Systems >>

VALVE NO. CP&L (EBASCO)	FUNCTION	REDUNDANT ISOL. VALVE NO. CP&L (EBASCO)	APPLICABLE MODES
1BD-1 (BD-P6)	SG A BLDN ISOL	1BD-11 (BD-V11)	1-3
1BD-20 (BD-P7)	SG B BLDN ISOL	1BD-30 (BD-V15)	1-3
1BD-39 (BD-P8)	SG C BLDN ISOL	1BD-49 (BD-V19)	1-3
1SP-214 (SP-V90)	SG A SHELL SAMPLE	1SP-217 (SP-V120)	1-3
1SP-216 (SP-V91)	SG A TUBE SAMPLE	1SP-217 (SP-V120)	1-3
1SP-219 (SP-V85)	SG B SHELL SAMPLE	1SP-222 (SP-V121)	1-3
1SP-221 (SP-V86)	SG B TUBE SAMPLE TUBE SHT ISO	1SP-222 (SP-V121)	1-3
1SP-224 (SP-V80)	SG C SHELL SAMPLE SHELL ISO	1SP-227 (SP-V122)	1-3
1SP-226 (SP-V81)	SG C TUBE SAMPLE TUBE SHT ISO	1SP-227 (SP-V122)	1-3
1FW-133 (FW-F3)	SG A MF REG VALVE	1FW-159 (FW-V26)	1-4
1FW-140 (FW-F6)	SG A MF BYPASS REG VALVE	1FW-159 (FW-V26)	1-4
1FW-249 (FW-F4)	SG B MF REG VALVE	1FW-277 (FW-V27)	1-4
1FW-256 (FW-F7)	SG B MF BYPASS REG VALVE	1FW-277 (FW-V27)	1-4
1FW-191 (FW-F5)	SG C MF REG VALVE	1FW-217 (FW-V28)	1-4
1FW-198 (FW-F8)	SG C MF BYPASS REG VALVE	1FW-217 (FW-V28)	1-4

<< Safeguards Systems Isolation Valves in Closed Systems >>

All safeguards systems isolation valves shall be functional in the modes specified.

With one or more valves non-functional, maintain the redundant isolation valve(s) functional for each line which is not isolated. Restore the non-functional valve to functional status or isolate the affected line within 72 hours. If the valve(s) is not restored or the associated line is not isolated then place the unit in a mode for which the isolation function is not required within the next 48 hours.

The following guidance is provided for complying with the requirement to isolate the penetration:

- Either the non-functional valve, or all valves listed under the Redundant Valves column must be closed (and de-activated, if applicable for power-operated valves), or a valve having the same safety class and seismic design class in series with the non-functional valve must be closed (and de-activated, if applicable for power-operated valves). If the piping branches, each branch must also be isolated.

- Penetration flow path(s) may be un-isolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated.

<< Harris Unit 1 Core Operating Limits Report >>

The Harris Unit 1 Core Operating Limits Report (COLR) is now located in a cycle-specific Engineering Instruction. The typical numbering and naming format for this document is as follows:

HNEI-0400-xxxx, Harris Unit 1 Cycle xx Core Operating Limits Report (COLR)

The following example shows the appropriate document number and title to search in Fusion to find a cycle-specific COLR (note that the document number and cycle number will change from cycle to cycle):

EXAMPLE
HNEI-0400-0012, Harris Unit 1 Cycle 22 Core Operating Limits Report (COLR)

Also, a simple Fusion search in released document space for document number "HNEI-0400-" will yield a list of the appropriate core design documents, among which the current cycle COLR will be found.

<< Instrument Uncertainties Associated with Reactor Vessel Pressure - Temperature Limits and Low Temperature Overpressure Protection System >>

1. Reactor Vessel P-T Limits:

   The P-T limits provided in Technical Specification Figures 3.4-2 and 3.4-3 do not include instrument uncertainties. Plant procedures implementing the P-T limits shall include the following minimum pressure and temperature uncertainties in a conservative manner.

	ERFIS	MCB/RECORDERS
Pressure (psig) Temperature (°F)	± 80 ± 11.5	± 120 ± 20.5

   Instrumentation used to monitor the above RCS variables for compliance with Technical Specifications 3.4.9.1 and 3.4.9.2 shall have a maximum loop instrument uncertainty (element to indication) at or within the values stated above.

2. Low Temperature Overpressure Protection System (LTOPS):

   The LTOPS setpoints provided in Technical Specification Figure 3.4-4 are the maximum nominal setpoints. They do not include instrument uncertainties. LTOP instrument uncertainty (element to actuation/arming) shall be at or within the following values:

   - Pressure (actuation): ± 65 psi
   - Temperature (actuation): ± 5°F
   - Temperature (arming): ± 7°F

<< Instrument Uncertainties Associated with Reactor Vessel Pressure - Temperature Limits and Low Temperature Overpressure Protection System >>

3. Temperature Range for Heatup and Cooldown Rates:

   The temperature ranges provided in Technical Specification Table 4.4-6 assume ERFIS temperature indication, which is normally available for plant heatup and cooldown, is utilized. If MCB/Recorder temperature indication is utilized, the instrument uncertainties are increased. The following table provides temperature ranges when utilizing MCB/Recorder temperature indication during heatup and cooldown:

HEATUP OR COOLDOWN	RATE (°F/HR)	RCS COLD LEG TEMPERATURE (°F)	
		USING ERFIS	USING MCB OR RECORDER
HU	50	≤350 [2]	≤ 350 [2]
CD	50	350-120 [2]	350-129 [2]
CD	30	< 120 [1]	< 129 [1]

Footnotes:

1. The lower bound temperature which LTOPS may be credited is 90°F using ERFIS per TS 3.4.9.4(b) and 99°F using MCB or recorder indication.

2. Instrument uncertainties at an indicated temperature of 350°F were also considered in the development of the heatup and cooldown pressure and temperature limits using either ERFIS or MCB indication.

<< Reactor Trip System Instrumentation Setpoint As-Found/As-Left Tolerances >>

	FUNCTIONAL UNIT	TRIP SETPOINT	AS-FOUND TOLERANCE	AS-LEFT TOLERANCE
1.	Manual Reactor Trip	N/A	N/A	N/A
2.	Power Range, Neutron Flux			
	a. High Setpoint	≤ 108% of RTP	1.12% Span	0.50% Span
	b. Low Setpoint	≤ 25% of RTP	1.12% Span	0.50% Span
3.	Power Range, Neutron Flux High Positive Rate	≤ 5% of RTP with a time constant ≥ 2 seconds	1.12% Span	0.50% Span
4.	Power Range, Neutron Flux High Negative Rate	≤ 5% of RTP with a time constant ≥ 2 seconds	1.12% Span	0.50% Span
5.	Pressurizer Water Level - High	≤ 87% of instrument span	Sensor Only (LT-459, LT-461/Barton): 1.46% Span Sensor Only (LT-460/Rosemount): 0.86% Span [7.3.22] Rack Only: 1.14% Span	Sensor Only(LT-459, LT-461/Barton): 0.50% Span Sensor Only (LT-460/Rosemount): 0.2% Span [7.3.22] Rack Only: 0.50% Span

NOTE: The following general note applies to the above table:

A discussion of the As-Found/As-Left tolerances is located in the Technical Specification Bases for the Reactor Trip System Instrumentation Setpoints. In the Harris Technical Specifications, the term Trip Setpoint is analogous to the Nominal Trip Setpoint (NTSP). Notes 7 and 8 have been added to Technical Specification Table 2.2-1 for the Power Range Neutron Flux and Pressurizer Water Level High Instruments that require verifying both the trip setpoint setting as-found and as-left values during surveillance testing. Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the trip setpoint in Table 2.2-1 of the Technical Specifications. The actual field setpoint and the associated as-found/as-left tolerances for the Nuclear Instrumentation Power Range and Pressurizer Water Level High channels are specified in this attachment. The bases for the as-found/as-left tolerances are contained within calculation HNP-I/INST-1010. The bases for adding Notes 7 and 8 is to provide implementation of the TSTF-493 recommendations (Option A) for the Nuclear Instrumentation Power Range and Pressurizer Water Level High setpoints.

<< Diesel Fuel Oil Testing Program >>

New Fuel Oil

Sample new fuel oil in accordance with ASTM-D4057-81 prior to addition to storage tanks and:

1. Verify, in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks, that the sample has:
   a. An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 26 degrees but less than or equal to 38 degrees.
   b. A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if the gravity was not determined by comparison with the supplier's certification;
   c. A flash point equal to or greater than 125°F; and
   d. A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.
2. Within 31 days following the addition of the new fuel oil to storage tanks, verify that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

Stored Fuel Oil

Every 31 days, obtain a sample of fuel oil from the storage tank, in accordance with ASTM-D2276-78, and verify that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A.

Main Fuel Oil Storage Tank Cleaning

At the frequency specified in the Surveillance Frequency Control Program, drain each main fuel oil storage tank, remove the accumulated sediment, and clean the tank using a sodium hypochlorite solution or other appropriate cleaning solution.

R
Reference
Use

SHEARON HARRIS NUCLEAR POWER PLANT

Plant Operating Manual

VOLUME 1

PART 2

Plant Programs (PLP)

PLP-114

Relocated Technical Specifications and Design Basis Requirements

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R 1.0 PURPOSE

This procedure identifies and controls regulatory requirements previously contained in the Shearon Harris Technical Specifications. The relocation of those requirements to the attachments contained in this procedure has been approved by the NRC through the issuance of license amendments. Since these requirements have been relocated to this licensee-controlled document, any future changes will require a 10CFR50.59 review.

Additionally, this procedure identifies and controls other design basis requirements identified by CP&L and which are not otherwise contained in the Technical Specifications, plant procedures, or other CP&L controlled documents.

2.0 REFERENCES

1. SHNPP Technical Specifications
2. SHNPP FSAR
3. Amendment No. 55 to Facility Operating License No. NPF-63
4. Amendment No. 61 to Facility Operating License No. NPF-63
5. Amendment No. 62 to Facility Operating License No. NPF-63
6. AD-DC-ALL-0201
7. AD-LS-ALL-0008
8. AD-LS-ALL-0019
9. AD-PI-ALL-0100
10. AD-LS-ALL-0005
11. AD-EG-ALL-1202
12. PLP-621
13. ESR-95-00687
14. ESR-94-00198
15. Letter (HNP-95-083) dated September 23, 1995 from CP&L to NRC, Reactor Auxiliary Building Emergency Exhaust System Design and Licensing Basis Issues
16. PNSC Meeting Minutes No. 92-36

2.0 REFERENCES (continued)

17. Amendment No. 64 to Facility Operating License No. NPF-63

18. ESR 96-00284

19. ESR 96-00126

20. ESR 00-00046, SFP Heatload Analysis for RFO9 and Cycle 10

21. ESR 00-00137, Engineering Disposition regarding CSIP Room Temperatures

22. NRC Regulatory Guide 1.133, Loose-Part Detection Program For The Primary System of Light-Water-Cooled Reactors

23. ESR 94-00001

24. EC 248760

25. EC 251445

26. Amendment No. 115 to Facility Operating License No. NPF-63

27. EC 248676

28. EC 272144

29. EC 274914

30. NRC Inspection Manual Part 9900: Technical Guidance

31. EC 300656, Revise Temperature Limit for 286' EL. A PIC Room

32. EC 402748, Increase RAB Battery Room Upper Temp Limit from 79 °F to 85 °F

33. AD-OP-ALL-0105, Operability Determinations and Functionality Assessments

34. Amendment No. 159 to Facility Operating License No. NPF-63

35. AD-LS-ALL-0007, Applicability Determination Process

36. AD-LS-ALL-0008, 10 CFR 50.59 Review Process

3.0 DEFINITIONS

None

4.0 GENERAL

1. Specific attachments to this procedure have been generated to fulfill the requirements of license amendments to Facility Operating License NPF-63 which allows the removal and relocation of specific Technical Specifications requirements for the Shearon Harris Nuclear Power Plant into licensee-controlled documents. Additionally, other attachments have been generated to control other design basis requirements identified by CP&L and not controlled elsewhere.

2. In addition to the reviews required by AD-DC-ALL-0201, changes to this procedure may also require a concurrent change to the FSAR per AD-LS-ALL-0005.

3. This procedure is incorporated by reference into the Harris Final Safety Analysis Report (FSAR) and is therefore made a part of the FSAR, which is part of the Harris current licensing basis. This procedure cannot be exempted from 50.59 review.

4. Operability of Structures, Systems, or Components (SSC) described in Technical Specifications:

   The scope of the SSCs considered in the operability determination process includes the following:

   a) SSCs required to be operable by Tech Specs. These SSCs may perform required support functions for other SSCs required to be operable by Tech Specs (example – emergency diesel and service water)

   b) SSCs that are not explicitly required to be operable by Tech Specs, but that perform required support functions (as specified by the TS definition of operability) for SSCs required to be operable by Tech Specs.

5.0 IMPLEMENTATION

5.1 Criteria for Implementing Relocated Technical Specifications and Design Basis Requirements

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

5.2 Additions, Changes, and Deletions to the Attachments

If the attachments to this procedure are revised or deleted or if new attachments are added, the following process shall be performed. This process exceeds the normal procedure review process as indicated in AD-DC-ALL-0201, Procedure Review and Approval.

1. Prepare a revision to the procedure per AD-DC-ALL-0201 for review by plant personnel.

2. Before approval by the General Manager - Harris Plant, Licensing/Regulatory Programs shall:

   - Ensure procedure revisions or new procedures are prepared to incorporate changes made to the attachments, or

   - Establish action requests per AD-PI-ALL-0100 for changes not requiring immediate implementation.

4. The procedure revision shall be submitted to the General Manager - Harris Plant for final approval along with an FSAR revision (if required) which incorporates the changes made to the Attachments per AD-LS-ALL-0005.

5. Upon final approval of the procedure revision, ensure that a task sheet is revised or created per AD-EG-ALL-1202, if appropriate, for testing and tracking purposes.

6.0 ATTACHMENTS

Attachment 1 - Turbine Overspeed Protection

Attachment 2 - Refueling Operations

R Attachment 3 - ECCS Leakage (Design Basis)

Attachment 4 - Area Temperature Monitoring

Attachment 5 - Gas Storage Tanks

Attachment 6 - Seismic Instrumentation

Attachment 7 - Meteorological Instrumentation

Attachment 8 - Metal Impact Monitoring System

Attachment 9 - Explosive Gas Monitoring Instrumentation

Attachment 10 - Feedwater Leading Edge Flow Meter (LEFM) Calorimetric

Attachment 11 - Liquid Holdup Tanks

Attachment 12 - Explosive Gas Mixture

Turbine Overspeed Protection

1.0 OPERATIONAL REQUIREMENTS

1.1 At least one Turbine Overspeed Protection System shall be functional.

APPLICABILITY: MODES 1, 2, and 3

NOTE: Not applicable in MODE 2 or 3 with all main steam isolation valves and bypass valves in the closed position and all other steam flow paths to the turbine isolated.

ACTION:

a. With one throttle valve or one governor valve per high pressure turbine steam line non-functional and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line non-functional, restore the non-functional valve(s) to functional status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise non-functional, within 6 hours isolate the turbine from the steam supply.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The above required Turbine Overspeed Protection System shall be demonstrated functional:

* a. At least once semiannually by direct observation of the movement of each of the following valves through at least one complete cycle from the running position:

(1) Four high pressure turbine throttle valves,

(2) Four high pressure turbine governor valves,

(3) Four low pressure turbine reheat stop valves, and

(4) Four low pressure turbine reheat intercept valves.

* The provision of Section 5.1, Criteria for Implementing Relocated Technical Specifications and Design Basis Requirements, maximum allowable extension not to exceed 25% of the specified surveillance interval does not apply.

Turbine Overspeed Protection

2.0 SURVEILLANCE REQUIREMENTS (continued)

b. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

c. Every refueling outage at least two low pressure turbine inlet valves, two governor valves, and two throttle valves will be swapped with rotational spares. The removed valves will be dismantled and inspected during non-outage periods. The non-outage dismantle inspections will as a minimum consist of surface examinations on the disks and stems. These non-outage inspected valves will then be used as rotational spares for the next refueling outage. During the non-outage inspection, if any attribute is found that would cause the valve function to be lost, then plans will be made for the upcoming refueling outage to inspect the other valves of the same group. Valve seats of the removed governor valves and throttle valves will be inspected for flaws during the refueling outage. If any attribute of the governor or throttle valve seats are found that would cause the valve function to be lost, the other remaining governor and throttle valve seats will be inspected during the same refueling outage. The low pressure turbine inlet valves do not have seats and are designed with some clearance between the disk and valve body. Under this turbine valve inspection schedule, all low pressure turbine inlet valves will be inspected over a four cycle period and all throttle valves and governor valves will be inspected over a two fuel cycle period.

Refueling Operations

1.0 OPERATIONAL REQUIREMENTS - DECAY TIME

1.1 The reactor shall be subcritical for a minimum period of time as determined by Table A.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for a time less than determined by Table A, suspend all operations involving movement of irradiated fuel in the reactor vessel. Fuel movement in the reactor vessel may continue provided the minimum decay time is greater than the time shown on Table A.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The reactor shall be determined to have been subcritical for a minimum period of time as determined using Table A by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

Refueling Operations

2.2 CCW temperature shall be monitored every 12 hours during the movement of fuel in the reactor vessel to ensure the temperature used to determine decay time is not exceeded.

Table A

Time from Reactor Subcritical (Hours)	Effective CCW Temperature (°F)
100	**96.9**
120	**99.3**
144	**101.7**
168	**103.8**
192	**105.6**
216	**107.2**
240	**108.6**

NOTE 1: - Linear interpolation between listed points is acceptable.
NOTE 2: - These delay times are applicable to end of cycle full core off-loads only. A mid-cycle core off-load assumes two CCW and Fuel Pool Cooling trains available and does NOT require compliance with these limits.
NOTE 3: - Effective CCW temperature refers to actual CCW heat exchanger outlet temperature plus 5°F.
NOTE 4: - The table assumes the core off-load duration is 30 hours or greater. Spent Fuel Pool Cooling analysis assumes full core off-load occurs no sooner than the earliest allowed time to start core off-load after reactor subcritical (based on CCW temperature) plus 30 hours.

Refueling Operations

3.0 OPERATION REQUIREMENTS - COMMUNICATIONS

3.1 Direct communications shall be maintained between the control room and personnel at the refueling station in containment.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

4.0 SURVEILLANCE REQUIREMENTS:

N/A

Refueling Operations

5.0 OPERATIONAL REQUIREMENTS - REFUELING MACHINE

5.1 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be functional with:

a. The refueling machine, used for movement of fuel assemblies, having:

1. A minimum capacity of 4000 pounds, and

2. An automatic overload cutoff limit less than or equal to 2700 pounds.

b. The auxiliary hoist, used for latching and unlatching drive rods, having:

1. A minimum capacity of 3000 pounds, and

2. A 0 - 2000 pound digital load indicator that shall be used to monitor loads to prevent lifting more than 600 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for the refueling machine and/or auxiliary hoist functionality not satisfied, suspend use of any non-functional refueling machine and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

6.0 SURVEILLANCE REQUIREMENTS

6.1 The refueling machine used for movement of fuel assemblies within the reactor vessel shall be demonstrated functional, within 30 days prior to the start of such operations, by performing a load test of at least 4000 pounds and demonstrating an automatic load cutoff at less than or equal to 2700 pounds.

6.2 The auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated functional within 30 days prior to the start of such operations by performing a load test of at least 900 pounds.

Refueling Operations

7.0 OPERATIONAL REQUIREMENTS - CRANE TRAVEL / FUEL HANDLING BUILDING

7.1 Loads in excess of 2300 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With irradiated fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

8.0 SURVEILLANCE REQUIREMENTS

8.1 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2300 pounds over fuel assemblies shall be demonstrated functional within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

9.0 OPERATIONAL REQUIREMENTS

9.1 Spent Fuel Pool loads used for plant operations scenarios assume that the refueling outage duration (reactor shutdown to re-synchronization) is no shorter than 15 days.

10.0 SURVEILLANCE REQUIREMENTS

10.1 Prior to Entry into MODE 1 following a refueling outage, it must be confirmed that the duration of a refueling outage is greater than 15 days.

R

ECCS Leakage (Design Basis)

1.0 OPERATIONAL REQUIREMENTS

1.1 External leakage from the Emergency Core Cooling System (ECCS) recirculation flow path pressure boundary located outside the Reactor Auxiliary Building (RAB) Emergency Exhaust System boundary shall be limited to a cumulative total of 2 gallons per hour.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With the above limit exceeded, reduce the leakage rate to within the limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2.0 SURVEILLANCE REQUIREMENTS

2.1 Areas outside the RAB Emergency Exhaust System boundary, excluding the CVCS filter room, that contain components (valves, flanges, strainers) in the ECCS recirculation flow path pressure boundary shall be inspected for indications of external leakage at least every 72 hours.

2.2 Components in the CVCS filter room shall be inspected for indications of external leakage at least every 31 days.

2.3 Any noticeable external leakage found while performing 2.1 or 2.2 above shall be measured or estimated to ensure total leakage is within the limit.

2.4 This portion of the ECCS system shall be leak tested in accordance with the Leakage Reduction Program requirements of PLP-621 at refueling cycle intervals or less.

REFERENCE:

ESR-95-00687

P&L Letter (HNP-95-083) to NRC, dated September 23, 1995

Area Temperature Monitoring

1.0 OPERATIONAL REQUIREMENTS

1.1 The temperature of each area shown in Table A shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be functional.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table A for more than 8 hours, prepare within 30 days an evaluation to demonstrate the continued functionality of the affected equipment.

b. With one or more areas exceeding the temperature limit(s) shown in Table A by more than 30°F, prepare an evaluation as required by Action a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) non-functional.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The temperature in each of the areas shown in Table A shall be determined to be within its limit at least once per 12 hours.

Area Temperature Monitoring

TABLE A

	AREA	MAXIMUM TEMPERATURE LIMIT (°F)
	REACTOR AUXILIARY BUILDING	
1.	Control Room Envelope (El 305')	75
2.	Process I&C, Room (El 305')	80
3.	Rod Control Cabinets Area (El 305')	104
4.	Auxiliary Relay Cabinet Room (El 305')*	80
5.	AH-15 Ventilation Room (El 305')*	104
6.	A&B Battery Rooms (El 286')	85***
7.	A&B Switchgear Rooms (El 286')	88****
8a.	Process I&C Room A (El 286')*	85
8b.	Process I&C Room B (El 286')*	85
9.	Auxiliary Transfer Panel Room (El 286')*	104
10.	Auxiliary Control Panel Room (El 286)*	88
11.	Main Steam, Feedwater Pipe Tunnel	122
12.	SA&SB Electrical Penetration Areas (El 261' & 286')	104
13.	E-6 Rooms (El 261')*	104
14.	Area with MCC 1A35SA and 1B35SB (El 261')	104
15.	HVAC Chillers, Auxiliary FW Piping & Valve Area (El 261')	104
16.	CCW Pumps, CCW Hx, Auxiliary FW Pumps Area (El 236')	104
17.	1A-SA, 1B-SB, and 1C-SAB Charging Pump Rooms (El 236')	104**
18.	Service Water Booster Pump 1B-SB (El 236')	104
19.	Mechanical and Electrical Penetration Areas (El 236')	104
20.	Containment Spray Additive Tank, and H&V Equipment Area (El 216')	104
21.	Trains A&B Containment Spray Pump, RHR Pump, H&V Equipment Areas (El 190')	104

See “Notes” on next page

Area Temperature Monitoring

TABLE A

	AREA	MAXIMUM TEMPERATURE LIMIT (°F)
	FUEL HANDLING BUILDING	
22.	Trains A&B Emergency Exhaust System Areas (El 261')	104
23.	Spent Fuel Pool Cooling Pump Room (El 236')	115.5
	WASTE PROCESSING BUILDING	
24.	H&V Equipment Room (El 236')	104
	MISCELLANEOUS	
25.	Tank Area (El 236')	104
26.	Diesel Fuel Oil Storage Building (El 242')	122
27.	Emergency Service Water Electrical Equipment Room	116
28.	Emergency Service Water Pump Room	122
29.	1A-SA & 1B-SB H&V Equipment Rooms (El 292')	122
30.	1A-SA & 1B-SB H&V Equipment Rooms (El 280')	118
31.	1A-SA & 1B-SB Electrical Rooms (El 261')	116
32.	1A-SA & 1B-SB Diesel Generator Rooms (El 261')	120

Notes:

* Areas 4, 5, 8, 9, 10, and 13 were added per PNSC Meeting 92-36.

** An Engineering Disposition has been performed regarding the selection of these setpoints. Reference ED ESR 00-00137 for additional information. This note added per AR# 3744.

*** Battery Room temperature of 85°F was established per EC 402748.

**** Ambient temperature of up to 104°F in the “A & B” Switchgear Room is acceptable whenever ventilation is not available during maintenance outage activities. (Ref. EC 278851)

Gas Storage Tanks

1.0 OPERATIONAL REQUIREMENTS

1.1 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 1.05×10^5 Curies of noble gases (considered as Xe-133 equivalent).

NOTE: The limit on the quantity of radioactive material stored in each gas storage tank will also be used as the limit on the total curie content in any portion of the Waste Gas System that is interconnected. Therefore, prior to interconnecting portions of the system, the influents to the system shall be isolated and the total curie content (considered as Xe-133 equivalent) in the tanks to be interconnected shall be determined to be within the limit.

The 1.05×10^5 curie limit is based on restricting the quantity contained in each gas storage tank to provide assurance that in the event of an uncontrolled release of the tank's contents the resulting whole body exposure to a member of the public will not exceed 0.5 rem. This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981. Interpretation of the limit to apply to any portion of the system that could be released due to single failure allows flexibility in operation of the system while ensuring any uncontrolled releases meet the above limit.

APPLICABILITY: At all times.

ACTION:

With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend any additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.1.4.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once every 7 days whenever radioactive materials are added to the tank, and at least once every 24 hours during primary coolant system degassing operations.

Seismic Instrumentation

1.0 OPERATIONAL REQUIREMENTS

1.1 The seismic monitoring instrumentation shown in Table A Attachment 6 shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments non-functional for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and plans for restoring the instrument(s) to functional status.

2.0 SURVEILLANCE REQUIREMENTS

2.1 Each of the above required seismic monitoring instruments shall be demonstrated functional by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and an ANALOG CHANNEL OPERATION TEST at the frequencies shown in Table B Attachment 6.

2.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to 0.01 g shall be restored to functional status within 24 hours and a CHANNEL CALIBRATION performed within 10 days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

Seismic Monitoring Instrumentation

TABLE A

Instruments and Sensor Locations	Measurement Range	Minimum Instruments Functional
1. Triaxial Time-History Accelerographs		
a. Containment Mat (El 221 ft)	0.01-1.0 g	1**
b. Containment (El 286 ft)	0.01-1.0 g	1**
c. Diesel Fuel Oil Storage Tank Building (El 242 ft)	0.01-1.0 g	1**
2. Triaxial Peak Accelerograph Recorders		
a. Reactor Coolant Pipe (Loop B)	±10 g	1
b. Steam Generator 1A Pedestal (El 238 ft)	± 2 g	1
c. Reactor Auxiliary Building (El 236 ft)	± 10 g	1
3. Triaxial Seismic Switches		
a. Starter Unit for Time History Accelerograph System - Containment Mat (El 221 ft)	0.005 - 0.05	1*
b. Triaxial Seismic Switch- Containment Mat (El 221 ft)	0.025 - 0.25	1*
4. Triaxial Response-Spectrum Recorders		
a. Steam Generator 1B Pedestal (El 238 ft)	± 2 g	1
b. Reactor Auxiliary Building (El 216 ft)	± 2 g	1
c. Diesel Fuel Oil Storage Tank Building (El 242 ft)	± 2 g	1
d. Containment Building (El 221 ft)	± 2 g	1*

* With Main Control Room Indication

** With Main Control Room Recording

Seismic Monitoring Instrumentation Surveillance Requirements

TABLE B

Instruments and Sensor Locations	Channel Check	Channel Calibration	Analog Channel Operational Test
1. Triaxial Time-History Accelerographs			
a. Containment Mat (El 221 ft)	M*	R	SA***
b. Containment (El 286 ft)	M*	R	SA***
c. Diesel Fuel Oil Storage Tank Building (El 242 ft)	M*	R	SA***
2. Triaxial Peak Accelerograph Recorders			
a. Reactor Coolant Pipe (Loop B)	N.A.	R	N.A.
b. Steam Generator 1A Pedestal (El 238 ft)	N.A.	R	N.A.
c. Reactor Auxiliary Building (El 236 ft)	N.A.	R	N.A.
3. Triaxial Seismic Switches			
a. Starter Unit for Time History Accelerograph System - Containment Mat (El 221 ft)**	M	R	SA***
b. Triaxial Seismic Switch- Containment Mat (El 221 ft)**	M	R	SA***

See NOTES on Next Page

Seismic Monitoring Instrumentation Surveillance Requirements

TABLE B (continued)

Instruments and Sensor Locations	Channel Check	Channel Calibration	Analog Channel Operational Test
4. Triaxial Response-Spectrum Recorders			
a. Containment Building (Active) (El 221 ft)**	M	R	SA***
b. Steam Generator (Passive) 1B Pedestal	N.A.	R	N.A.
c. Reactor Auxiliary Building (Passive) (El 216 ft)	N.A.	R	N.A.
d. Diesel Fuel Oil Storage Tank Building (Passive)(El 242 ft)	N.A.	R	N.A.

* Except seismic starter unit

** With Main Control Room Alarms

*** The bistable trip setpoint need not be determined during the performance of a channel operational test

Meteorological Instrumentation

1.0 OPERATIONAL REQUIREMENTS

1.1 The meteorological monitoring instrumentation channels shown in Table A Attachment 7 shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels non-functional for more than 7 days:

(1) Within 24 hours, obtain Plant Manager or designee approval for plans to restore meteorological monitoring instrumentation channel(s) to functional status.

AND

(2) Initiate a Corrective Action Program (CAP) document in order to document cause(s) and corrective action(s).

2.0 SURVEILLANCE REQUIREMENTS

2.1 Each of the above meteorological monitoring instrumentation channels shall be demonstrated functional by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table B Attachment 7.

Meteorological Monitoring Instrumentation

Table A

Instrument	Location	Minimum Functional
1. Wind Speed		
	Nominal Elev. 12.5 meters	1
	Nominal Elev. 61.4 meters	1
2. Wind Direction	Nominal Elev. 12.5 meters	1
	Nominal Elev. 61.4 meters	1
3. Air Temperature - Differential Temperature	11.0 meter and 59.9 meters	1

Meteorological Monitoring Instrumentation Surveillance Requirements

Table B

Instrument	Channel Check	Channel Calibration
1. Wind Speed		
a. Nominal Elev. 12.5 meters	D	SA
b. Nominal Elev. 61.4 meters	D	SA
2. Wind Direction		
a. Nominal Elev. 12.5 meters	D	SA
b. Nominal Elev. 61.4 meters	D	SA
3. Differential Air Temperature Between 11.0 meters and 59.9 meters	D	SA

Metal Impact Monitoring System

1.0 OPERATIONAL REQUIREMENTS

1.1 The metal impact monitoring system shall be functional.

APPLICABILITY: Modes 1 and 2.

ACTION:

a. With both metal impact monitoring system channels in a region* non-functional for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and plans for restoring the channel(s) to functional status.

2.0 SURVEILLANCE REQUIREMENTS

2.1 Each channel of the metal impact monitoring system shall be demonstrated functional by the performance of:

a. A channel check at least once per 24 hours.

b. An analog channel operational test, except for verification of setpoint, at least once per 31 days, and a channel calibration at least once per 18 months.

* Note: MIMS regions are defined as follows:

- Reactor Vessel Upper
- Reactor Vessel Lower
- Steam Generator “A”
- Steam Generator “B”
- Steam Generator “C”

Explosive Gas Monitoring System

1.0 OPERATIONAL REQUIREMENTS

1.1 The explosive gas monitoring instrumentation channels shown in Table A Attachment 9 shall be functional with their alarm/trip Setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table A Attachment 9.

ACTION:

a. With an explosive gas effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification declare the channel non-functional and take the action shown in Table A Attachment 9.

b. With the number of functional explosive gas monitoring instrumentation channels less than the minimum channels functional , take action shown in Table A Attachment 9. Restore the non-functional instrumentation to functional status within 30 days, and if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this non-functionality was not corrected in a timely manner.

2.0 SURVEILLANCE REQUIREMENTS

Each explosive gas monitoring instrumentation channel shall be demonstrated functional by the performance of a Channel Check, Channel Calibration and Analog Channel operational test at the frequencies shown in Table B Attachment 9.

Explosive Gas Monitoring Instrumentation

TABLE A

Instrument	Minimum Channels Functional	Applicability	Action
1. Gaseous Waste Processing System - Hydrogen and Oxygen Analyzers			
a. Recombiner Outlet Hydrogen Monitor	1/recombiner	*	50
b. Recombiner Outlet Oxygen Monitor	1/recombiner	*	48
c. Compressor Discharge Oxygen Monitor	1	*	48

Table Notations

*During GASEOUS WASTE PROCESSING SYSTEM Operation

Action Statements

Action 45 (Not used)

Action 46 (Not used)

Action 47 (Not used)

Action 48 With the number of channels functional less than the minimum channels functional requirement, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 12 hours during other operations.

Action 49 (Not used)

Action 50 With the number of channels functional one less than required by the minimum channels functional requirement, suspend oxygen supply to the recombiner.

Action 51 (Not used)

Action 52 (Not used)

Explosive Gas Monitoring Instrumentation Surveillance Requirements

TABLE B

Instrument	Channel Check	Channel Calibration	Analog Channel Operational Test	Modes for Which Surveillance is required
1. Gaseous Waste Processing System-Hydrogen and Oxygen Analyzers				
a. Recombiner Outlet Hydrogen Monitor	D	Q(4)	M	*
b. Recombiner Outlet Oxygen Monitor	D	Q(5)	M	*
c. Compressor Discharge Oxygen Monitor	D	Q(5)	M	*

Table Notations

*During GASEOUS WASTE PROCESSING SYSTEM operation.

(1) (Not Used)

(2((Not Used)

(3) (Not Used)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing hydrogen and nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing oxygen and nitrogen.

Feedwater Leading Edge Flow Meter (LEFM) Calorimetric

1.0 OPERATIONAL REQUIREMENTS

1.1 The Feedwater LEFM Calorimetric shall be functional with:

a. The Feedwater LEFM System in the Normal mode.

b. The ERFIS calorimetric program functional.

APPLICABILITY: MODE 1 with THERMAL POWER > 2900 MWt (98.4% RTP)

ACTION:

a. With the Feedwater LEFM System in the Fail mode, change the calorimetric program from the Feedwater LEFM System to the Normalized Feedwater Venturi System within 1 hour and either:

(1) Restore the Feedwater LEFM System to the Normal mode within 72 hours, or reduce THERMAL POWER to ≤ 2900 MWt (98.4% RTP) and change the calorimetric program from the Normalized Feedwater Venturi System to the Feedwater Venturi System. If the plant experiences a power decrease below 2900 MWt (98.4% of RTP) during the 72 hour allowed outage time, the maximum permitted power level will be 2900 MWt until the LEFM is restored to either Normal or Maintenance mode operation.

OR

(2) Restore the Feedwater LEFM System to the Maintenance mode within 72 hours and reduce THERMAL POWER to ≤ 2943 MWt (99.86% RTP). The plant can operate at this power level indefinitely.

b. With the Feedwater LEFM System in the Maintenance mode, restore the Feedwater LEFM System to the Normal mode within 72 hours, or reduce THERMAL POWER to ≤ 2943 MWt (99.86% RTP). The plant can operate at this power level indefinitely.

c. With the ERFIS calorimetric program non-functional for reasons other than the LEFM System, restore the ERFIS calorimetric program to functional status prior to performing the next required power range channel calorimetric heat balance comparison per TS Table 4.3-1, channel calibration D2, or reduce THERMAL POWER to ≤ 2900 MWt (98.4% RTP) by monitoring alternate power indications.

2.0 SURVEILLANCE REQUIREMENTS

2.1 Perform the power range channel calorimetric heat balance comparison per TS Table 4.3-1, channel calibration D2 using the Feedwater LEFM System prior to exceeding 2900 MWt (98.4% RTP) and once per 24 hours thereafter.

2.2 Perform Channel Calibration of the Feedwater LEFM System instrumentation once per 18 months.

Liquid Holdup Tanks

NOTE: Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

1.0 OPERATIONAL REQUIREMENTS

1.1 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases:

a. Outside temporary tank, excluding demineralizer vessels and liners used to solidify or to dewater radioactive wastes.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Annual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents within 7 days following any addition of radioactive material to the tank.

Explosive Gas Mixture

1.0 OPERATIONAL REQUIREMENTS

1.1 The concentration of oxygen in the Gaseous Waste Processing System downstream of the hydrogen recombiners shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the Gaseous Waste Processing System downstream of the hydrogen recombiners greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the Gaseous Waste Processing System downstream of the hydrogen recombiners greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

2.0 SURVEILLANCE REQUIREMENTS

2.1 The concentrations of hydrogen and oxygen in the Gaseous Waste Processing System shall be determined to be within the above limits by monitoring, at least once per 12 hours, the waste gases in the Gaseous Waste Processing System.

Revision Summary

General (Revision 028)

This revision (PRR 2170123) incorporates the following:

PRR 2144288: Wording in PLP-114, Purpose (section 1) states: This procedure identifies and controls regulatory requirements previously contained in the Shearon Harris Technical Specifications. The relocation of those requirements to the attachments contained in this procedure has been approved by the NRC through the issuance of license amendments. Since these requirements have been relocated to this licensee-controlled document, any future changes will require 10CFR50.59 evaluation. Recommended change is to replace "10CFR50.59 evaluation" with "10CFR50.59 review".

Section 4.3 states: This procedure is incorporated by reference into the Harris Final Safety Analysis Report (FSAR) and is therefore made a part of the FSAR, which is part of the Harris current licensing basis. This procedure cannot be exempted from 50.59 screening review. Recommended change is to replace "50.59 screening" with "50.59 review".

Reason for Change: The current language that requires 50.59 screening or evaluation is outdated by the current 50.59 process. Per AD-LS-ALL-0007, a 50.59 review is a more accurate description of the expectation to review procedure changes under 50.59 (the previous legacy progress procedure required screens for everything that was not exempted, whereas the current procedure requires an additional step, known as the AD process).

PRR 2129830: AP-013 was superseded by AD-LS-ALL-0019, On-Site Review Committee (ORC). Review of PLP-114 by ORC is not required. Delete requirements for PNSC review.

PRR 2170123: The proposed activity modifies PLP-114, Attachment 2, Refueling Operations, as follows: PLP-114, Attachment 2, Section 3.0 Operations Requirement - Communications, step 3.1 contains an action that states that when direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. Attachment 2, Section 4.0 Surveillance Requirements, step 4.1 states that direct communications between the control room and personnel at the refueling station in containment shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during core alternations. The communication capability between the control room and the refueling station personnel is required to ensure that any significant changes in the facility status or core reactivity conditions during core alternations observed on the control room instrumentation can be communicated to refueling station personnel. The change to this instruction will continue to require that direct communications between the control room and personnel at the refueling station in containment is demonstrated during core alterations, but will remove the surveillance requirements.

Reason for Change: This change was initially pursued based upon review of Oconee communications requirements during core alternations. This PLP-114 change is acceptable since this change does not impact HNP UFSAR requirements for the Onsite Communications System or how this system is used to support the Fuel Handling System during refueling operations.

Revision Summary

Description of Changes

Page	Section	Change Description
3	1.0	Changed 50.59 evaluation to 50.59 review. (PRR 214428)
3	2.0.8	Replaced AP-013 with AD-LS-ALL-0019. (PRR 2129830)
4	2.0.35	Added AD-LS-ALL-0007, Applicability Determination Process to References. (PRR 214428)
4	2.0.36	Added AD-LS-ALL-0008, 10 CFR 50.59 Review Process to References. (PRR 214428)
5	4.0.2	Deleted PNSC review requirements. (PRR 2129830)
5	4.0.3	Changed 50.59 screening to 50.59 review. (PRR 214428)
6	5.2.2 5.2.4	Deleted 5.2.2 and revised 5.2.4: Deleted PNSC review requirements. (PRR 2129830)
12	Attachment 2 Sheet 3	The SURVEILLANCE REQUIREMENT was changed to 'N/A'. (PRR 2170123)

DUKE ENERGY CORPORATION

TOPICAL REPORT

Quality Assurance Program Description

Operating Fleet

DUKE-QAPD-001 -A-

QUALITY ASSURANCE PROGRAM

POLICY STATEMENT

Duke Energy Corporation (DEC) designs, procures, constructs and operates its nuclear plants in a manner that ensures the health and safety of the public and workers. These activities are performed in compliance with the requirements of the Code of Federal Regulations (CFR), the applicable Nuclear Regulatory Commission (NRC) Facility Operating Licenses, and applicable laws and regulations of the state and local governments.

The applicable Quality Assurance Program (QAP) is the Quality Assurance Program Description (QAPD) contained or referenced in each nuclear plant's Updated Final Safety Analysis Report and the associated implementing documents. Together they provide for control of DEC activities that affect the quality of safety-related nuclear plant structures, systems, and components (SSCs) and include all planned and systematic activities necessary to provide adequate confidence that such SSCs will perform satisfactorily in service. The QA Program may also be applied to certain equipment and activities that are not safety-related, but support safe plant operations, or where other NRC guidance establishes program requirements.

The QAPD is the top-level policy document that establishes DEC’s overall philosophy regarding achievement and assurance of quality. Implementing documents assign detailed responsibilities and requirements and define the organizational interfaces involved in conducting activities within the scope of the QAP. Compliance with the QAP is mandatory for individuals involved directly or indirectly with its implementation.

DEC personnel have authority commensurate with their responsibility, including the authority to stop work that does not conform to established requirements. This stop work authority may be exercised in accordance with established nuclear system procedures.

Figure 17-1, Duke Energy Corporation Quality Assurance Policy Statement

Summary of Changes

Changes since last NRC update at Amendment 42

DRR #	Description of Change
02125077	Revised the 3rd paragraph of Section 17.3.2.13 to provide clarity of requirements. This change split the paragraph to separate the first sentence from the remainder of the paragraph that was transferred to this section in Amendment 42. The following administrative improvements and clarifications identified during the preparation of Amendment 43 are included in this evaluation for this DRR: Addition of Regulatory Guide 1.164 and EPRI 3002002982 for Dedication of Commercial Grade Items in Table 17-1 (RG 1.123 and Generic Letter 89-02) and Section 17.3.2.4.1. Clarifications in Section 17.3.3.3 to reduce repetition in the section. Clarifications in Attachment A Section A17.3 under QA Standards and Guides to remove content that needlessly duplicates Standard content. Similar clarifications are made in Attachments B, C, and D.
02139232	Revised Brunswick exception 10 for Regulatory Guide 1.33 in Attachment A Table A17-1 to establish a consistent fleet approach for exception to the ANSI N18.7 requirement for procedure biannual reviews. Content is added to Section 17.3.2.14 (final two paragraphs) and removed from Section D17.3.2.14 describing the administrative controls referenced in the NRC letter dated January 31, 1995 approving this exception. Conforming changes were made to Tables A17-1, C17-1, and D17-1 to state the exception consistently.
02147631	Revised Section 17.3.3.2.1.3 to clarify the On-Site Review Committee review requirements. Relocated paragraph from 17.3.3.2.1.4.
02147698	Revised Tables A17-1, B17-1, C17-1, and D17-1 to standardize the exception for Regulatory Guide 1.58 and ANSI N45.2.6 to allow task qualification for inspectors rather than requiring receipt inspectors to be qualified by levels.

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17 QUALITY ASSURANCE

17.1 QA DURING DESIGN AND CONSTRUCTION

NOTE: Not included, this description of the Quality Assurance Program follows Standard Review Plan Section 17.3 for format and content.

17.2 OPERATIONAL QA

NOTE: Not included, this description of the Quality Assurance Program follows Standard Review Plan Section 17.3 for format and content.

17.3 QUALITY ASSURANCE PROGRAM DESCRIPTION

INTRODUCTION

The Duke Energy Corporation Quality Assurance Program (QAP) Policy Statement in Figure 17-1 describes the corporate policy and assigns responsibility for implementation of the QAP.

Duke Energy Corporation maintains full responsibility for assuring its nuclear power plants are designed, constructed, tested and operated in conformance with good engineering practices, applicable regulatory requirements and specified design bases and in a manner to protect the public health and safety. To this end Duke Energy Corporation has established and implemented a Quality Assurance Program which conforms to the criteria established in Appendix B to Title 10 Code of Federal Regulations (10 CFR), Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" published June 27, 1970 (35 F. R. 10499), amended September 17, 1971 (36 F. R. 18301), amended January 20, 1975 (40 F. R. 3210D), and amended August 28, 2007 (72 F. R. 49505).

This document follows the format and content guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 17.3, "Quality Assurance Program Description," except that the Duke Energy Corporation QAP is based on ANSI N18.7 and the ANSI N45.2 series standards in lieu of ANSI/ASME NQA-1 and NQA-2. This document is applicable to Duke Energy Corporation operating nuclear power stations as referenced by Chapter 17 of each station's UFSAR for those systems, components, items, and services that have been determined to be nuclear safety related.

This document is organized with a generic description of the organization and overview of the QAP in the main body of the document. Site specific details for the Quality Assurance Program Description along with conformance to the regulatory positions of the NRC QA Regulatory Guides are addressed in separate attachments as follows:

- Attachment A, Brunswick Specific QAPD
- Attachment B, Harris Specific QAPD
- Attachment C, Robinson Specific QAPD
- Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Each Attachment follows the section numbering in the main body of the document. The Brunswick, Harris, and Robinson attachments contain the conformance to the QA related Regulatory Guides, identified in Table 17-1, transferred from Chapter 1 of each respective UFSAR. Each attachment also contains supplemental descriptions transferred from each respective UFSAR Chapter 17, Section 17.3 when detail was included beyond the generic text

in the main body. Attachment D contains the conformance to the QA related Regulatory Guides, identified in Table 17-1, transferred from Amendment 40 of the Duke Energy Carolinas Topical Report Quality Assurance Program. Attachment D also contains supplemental descriptions from the Duke Energy Carolinas Topical Report Quality Assurance Program when detail was included beyond the generic text in the main body.

As discussed herein, the Quality Assurance Program (QAP) includes the description contained in this document and the controlled documents providing implementation of the requirements of this document, including the requirements of industry standards to the degree identified in Table 17-1, Conformance with QA Regulatory Guides and Industry Standards, and Table 17-2, Site Specific Response to Regulatory Guides and Industry Standards. The QAP provides a method of applying graded controls to certain non-safety related systems, components, items, and services (such as fire protection and radioactive waste structures, systems, and components).

Subsequent changes to the Duke Energy Corporation QAP are incorporated in this document as identified in Section 17.3.1.7. The QAP controlled implementing documents are used and updated as necessary to assure the nuclear generating units are managed such that they will be operated and maintained in a safe manner.

DEFINITIONS

The following definitions are applicable to terms used in this report. Refer to ANSI N45.2.10, "Quality Assurance Terms and Definitions" for definition of terms not included below.

Audit – The following modifications are applied to the definition in ANSI N45.2.10:

> Internal Audit - An activity to determine through investigation the adequacy of, and adherence to, established procedures, instructions, specifications, codes, and licensing requirements, and the effectiveness of implementation of the Duke Energy Corporation QAP.
>
> Supplier Audit - A documented activity performed in accordance with written procedures or checklists to verify, by examination and evaluation of objective evidence, that applicable elements of the supplier's QA program has been developed, documented and implemented in accordance with specified requirements.

Basic Component – See 10 CFR Part 21.

Commercial Grade Items - See 10 CFR Part 21.

Deficiency - Any condition considered to be adverse to quality including inadequacies of personnel, procedures, systems, methods, or items.

Engineering Change (Modification) - A planned change in plant design accomplished in accordance with the requirements and limitations of applicable codes, standards, specifications, licenses and predetermined safety restrictions.

Hold Point - That point in the manufacturing, preparation, development, installation and construction, inspection, or testing process that requires witness or review by qualified personnel.

Inspector - Any individual certified to the requirements identified in Table 17-1 for Regulatory Guide 1.58 who performs required inspections, tests or examinations.

Pre-award Survey - A documented activity performed in accordance with written procedures or checklists to verify, by examination and evaluation of objective evidence, that the supplier's QA

program has been developed, documented, and implemented in accordance with specified requirements.

Quality Assurance (QA) - The planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.

QA Records - Those records which furnish documentary evidence of the quality of items and of activities affecting quality.

QA Requirements - Those inspection, test, examination, certification and documentation requirements which are imposed to provide objective evidence of the conformance of an item or activity to established design, engineering, standards, and code requirements.

Services - The performance by a supplier of activities such as calibration, design, investigation, inspection, nondestructive examination, software applications, and installation.

EXPLANATION OF "QUALITY ASSURANCE"

Quality Assurance (QA) as used in this document includes:

1) Performance of planned and systematic actions necessary to provide assurance of the safety and integrity of the facility.

   The QAP is founded on the principle that the line organization has the primary responsibility for quality and safety. Self-assessment practices are used to ensure the desired levels of quality and safety are achieved and maintained. Each individual is responsible to ensure the plant is operated in a safe, reliable, and efficient manner.

2) Quality verifications performed by those independent of the performers.

   When required, verification of conformance to established program requirements is accomplished by qualified individuals who do not have responsibility for performing or directly supervising the work. Nuclear Oversight (NOS) evaluates the performance, compliance, and effectiveness of plant programs, processes, and personnel. The activities of NOS are intended to detect deficiencies in the desired levels of performance and quality, communicating these conditions to those responsible for the activities, appropriate management and the Chief Nuclear Officer, and ensuring adequate action is taken to correct these conditions.

QA STANDARDS AND GUIDES

The Duke Energy Corporation QAP conforms to Appendix B of 10 CFR 50. This description of the QA Program is formatted per NUREG-0800 Section 17.3, "Quality Assurance Program Description;" however, the Duke Energy Corporation QAP continues to use the ANSI N45.2 series standards in lieu of ANSI/ASME NQA-1 and NQA-2.

Table 17-1 identifies the QA program Regulatory Guides and other NRC program guidance for which conformance is addressed in this description of the QA Program. Changes to conformance for the Regulatory Guides in Table 17-1 are controlled in accordance with 10 CFR 50.54(a) and are incorporated in this document as identified in Section 17.3.1.7.

Table 17-2 identifies additional Regulatory Guides that relate to QA program implementation but where the subject matter closely relates to UFSAR technical content. Conformance for those Regulatory Guides is site specific and addressed with each site's UFSAR.

Together, Tables 17-1 and 17-2 indicate where conformance is identified for the regulatory guidance documents referenced in NUREG-0800 Section 17.3.

Table 17-1. Conformance with QA Regulatory Guides and Industry Standards

Generic Exception:

Table 17-1 addresses Duke Energy Corporation's Conformance of the Quality Assurance Program to certain NRC Regulatory Guides. In so doing, specific editions of industry standards are identified for compliance with exceptions and alternatives. Those identified standards include references to other industry standards for activities. Those referenced industry standards are considered to be guidance documents for details of how activities may be accomplished. The actual standard to be used in such cases is controlled by each station's current licensing and design bases (e.g. ANSI N18.7-1976 Section 3.4.2 identifies American National Standard for Selection and Training of Nuclear Power Plant Personnel, N18.1-1971. The actual standard used is site specific as identified in Table 17-2 for Regulatory Guide 1.8.).

Regulatory Guide 1.28, Quality Assurance Program Requirements (Design and Construction)
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.28 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.30 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation)
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.33 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Table 17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.37 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.38 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.39, Housekeeping Requirements for Water-Cooled Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.39 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.58 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Table 17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.64 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.74, Quality Assurance Terms and Definitions
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.74 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
The Duke Energy program for storage of records on microfilm, dual storage or in electronic format meets the preservation requirement for the retention of QA Records. For management of electronic records, the appropriate controls on quality are summarized as follows: a) The Electronic Records Management (eRM) system does not allow deletion or modification of records. (NOTE: Authorized deletion of records per the Record Retention Rules is controlled.) b) The eRM system provides redundancy (i.e., system backup, dual storage, etc.). c) The legibility of each record is verified prior to acceptance into the eRM system. d) The media used by the eRM system is maintained to ensure the records are acceptably copied onto a new media before the manufacturer's certified useful life of the media is exceeded. This includes verification of the records so copied. e) Periodic random inspections of records are performed to verify that there has been no degradation of record quality. f) If the eRM system in use is to be replaced by new system, the records stored on the old system are acceptably converted into the new system before the old system is taken out of service. This includes verification of the records so copied. To implement those controls, Duke Energy Corporation uses the following Nuclear Information and Records Management Association (NIRMA) standards: • NIRMA TG 11-2011 "Authentication of Records and Media" • NIRMA TG 15-2011, "Management of Electronic Records," • NIRMA TG 16-2011, "Software Quality Assurance Documentation and Records" • NIRMA TG 21-2011, "Required Records Protection, Disaster Recovery and Business Continuation"

Table 17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.88 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD
Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.94 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD
Regulatory Guide 1.116, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.116 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD
Regulatory Guide 1.123, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
Reference table content for Generic Letter (GL) 89-02 applicable to the procurement of Commercial Grade Items and services. For the procurement of commercial grade calibration and/or testing services, Duke Energy Corporation uses NEI 14-05A, Revision 0, "Guidelines for the Use of Accreditation In Lieu of Commercial Grade Surveys for Procurement of Laboratory Calibration and Test Services." The conditions for the use of this process, consistent with NRC Safety Evaluation dated April 1, 2016 to Union Electric Company, Callaway Plant (ADAMS Accession # ML16089A167), are identified in Sections 17.3.2.4 and 17.3.2.5.

Regulatory Guide 1.123, Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
Note: Well defined and documented measurement assurance techniques or uncertainty analysis may be used to verify the adequacy of the measurement process. If such techniques are not used, the collective uncertainty of the measurement standards shall not exceed 25% of the acceptable tolerance for each characteristic being calibrated. (This is typically referred to as the four-to-one ratio.) The Duke Energy Corporation QAP conforms to Regulatory Guide 1.123 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.144 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.146, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
The Duke Energy Corporation QAP conforms to Regulatory Guide 1.146 as identified in: • Attachment A, Brunswick Specific QAPD • Attachment B, Harris Specific QAPD • Attachment C, Robinson Specific QAPD • Attachment D, Catawba, McGuire, and Oconee Specific QAPD

Regulatory Guide 1.152 Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants
Conformance to Regulatory Guide 1.152 was not addressed during the licensing of the operating Duke Energy Corporation Nuclear plants.

Table 17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 7.10, Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material
Duke Energy Corporation does not conform to Regulatory Guide 7.10. This QAPD is used to satisfy applicable Quality Assurance requirements for packaging and transportation of radioactive material.
Generic Letter 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products
Duke Energy complies with the provisions of Generic Letter (GL) 89-02. GL 89-02 was issued in March 1989. This generic letter provides the staff's perspective on good practices in procurement and dedication and the NRC's conditional endorsement of an industry standard (EPRI NP-5652, Revision 0) on the methods of commercial-grade item procurement and dedication. Consistent with that guidance, Duke Energy complies with EPRI NP-5652, *"Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications (NCIG-07)"*. When NRC publishes additional guidance for the dedication of Commercial Grade Items, Duke Energy may utilize that guidance in the completion documentation provided any clarifications identified by the NRC are followed. Regulatory Guide 1.164, Dedication of Commercial-Grade Items for Use in Nuclear Power Plants, Revision 0 issued June 2017 Duke Energy also complies with the provisions of Regulatory Guide 1.164, which endorses in part, with exceptions or clarifications, EPRI 3002002982, Revision 1 to EPRI NP-5652 and TR-102260, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Items in Nuclear Safety-Related Applications" with respect to acceptance of commercial-grade dedication of items and services to be used as basic components for nuclear power plants. Regulatory Guide 1.231, Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants, Revision 0 issued January 2017 Duke Energy complies with the provisions of Regulatory Guide 1.231 which approves for use, with clarifications, EPRI Technical Report 1025243, "Plant Engineering: Guideline for the Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Nuclear Safety-Related Applications," Revision 1.
Quality assurance for Fire Protection from Positions 2 & 4 of Branch Technical Position CMEB 9.5-1 (Attachment to NUREG 0800 Section 9.5.1 Revision 3)
Quality assurance controls for non-Nuclear Safety Related components Important to Fire Protection are in accordance with the intent of Positions 2 & 4 of Branch Technical Position CMEB 9.5-1. Identification of items Important to Fire Protection is site specific consistent with each site's Fire Protection Program.

Table 17-2. Site Specific Response to Regulatory Guides and Industry Standards

Table 17-2 identifies additional Regulatory Guides addressing subjects related to implementation of the QAP but the implementation is site specific and addressed with each site's UFSAR.

Regulatory Guide 1.8, Personnel Selection and Training
Personnel selection and training is site specific addressing requirements beyond nuclear safety related applications.
Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
Quality group classifications and standards trace to the original design and construction of the nuclear power plant and therefore are site specific.
Regulatory Guide 1.29, Seismic Design Classification
Seismic design classification trace to the original design and construction of the nuclear power plant and therefore is site specific.
Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel
Nonmetallic thermal insulation for austenitic stainless steel trace to the original design and construction of the nuclear power plant and therefore is site specific.
Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
Requirements for protective coatings applied to water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific.
Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
Design of radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific.
Regulatory Guide 1.155, Station Blackout
Addressing Station Blackout is site specific.

Table 17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment
Requirements for radiological monitoring program (normal operations) – effluent streams and the environment is site specific.

17.3.1 MANAGEMENT

17.3.1.1 Methodology

The Chief Nuclear Officer (CNO) is the corporate executive responsible for quality assurance (QA) and is the highest level of management responsible for establishing Duke Energy Corporation's QA policies, goals, and objectives.

The QAP Policy Statement, shown in Figure 17-1, requires compliance with the QAP implementing documents in nuclear safety related matters. Organizations performing quality affecting activities are bound by this Policy Statement. The QAP has been developed in accordance with this Policy Statement. The QAP applies to individuals and organizations responsible for operating and supporting the nuclear plants in the performance of activities affecting quality (e.g., operation, maintenance, modification, and refueling). The implementing documents define responsibilities and authorities, prescribe measures for the control and accomplishment of activities for the operation of nuclear safety related structures, systems, and components and requires appropriate verification of conformance to established requirements to an extent consistent with their importance to safety. The individuals who constitute Nuclear Generation have full personal and corporate responsibility to assure that nuclear power plants are designed, constructed, tested and operated in a manner to protect the public health and safety. The comprehensive program to assure this began with initial design and continues throughout the life of the station. The Duke Energy Corporation QAP assures that the necessary quality requirements for nuclear safety related structures, systems, components and materials are achieved. All special equipment, environmental conditions, skills and processes that are determined to be nuclear safety related will be provided within the scope of the QAP.

Nuclear safety related structures, systems, and components (SSCs) are specified by approved design documents. Each nuclear plant has a controlled system for identifying items and activities to which the QAP applies. Controls and responsibilities for maintaining the system are prescribed in procedures.

The QAP applies to the nuclear safety related portions of the plant. The program is applied, in whole or in part, to other selected items based on the item’s or activity’s importance to safety. This application includes but is not limited to control and accomplishment of activities for radioactive waste, fire protection, seismically designed/restrained SSCs whose continued functions are not required during and after a seismic event, and License Renewal non-safety-related SSCs that are subject to an aging management review. Procedures provide a graded application of this QAP to non-safety related systems, components, items, and services by prescribing measures for the control and accomplishment of activities for their operation. For example, aging effects of non-safety related SSCs that were determined to be within the scope of License Renewal Aging Management Program as identified in Chapter 18 of the applicable site UFSAR, are included in the QAP for the administrative controls, corrective actions and confirmation processes described in Sections 17.3.1.6 and 17.3.2.13, Corrective Action, and 17.3.2.14, Document Control.

The QAP is founded on the principle that the line organization has the primary responsibility for quality and safety. Self-assessment practices are used to ensure the desired levels of quality and safety are achieved and maintained. This consists of each individual being involved with plant performance to ensure the plant is operated in a safe, reliable, and efficient manner. The Nuclear Oversight (NOS) Department evaluates the compliance, and effectiveness of plant programs, processes, personnel, and the line organization's self-assessment.

17.3.1.2 Organization

This section provides a generic functional description of the organization. The actual organization in-place is defined in a controlled implementing document containing the fleet operating model.

Plant specific details for the organization responsible for the safe plant operation are described in Chapter 13 of the UFSAR for each plant and in implementing documents. The term "line organization" refers to the production organization reporting to the CNO and the interfacing department staff supporting the Nuclear Generation as identified in Section 17.3.1.2.3, Department Interfaces. "Line organization" does not include the independent verification functions of the Nuclear Oversight organization.

17.3.1.2.1 Corporate Organization

The Chairman, President and Chief Executive Officer has overall responsibility for Design, Construction, Operation, and Decommissioning of generation facilities. Reporting to the Chairman, President and Chief Executive Officer is the Executive Vice President and Chief Operating Officer, who is responsible for generation and transmission including nuclear operations, nuclear development and nuclear decommissioning. Reporting to the Executive Vice President and Chief Operating Officer are the Senior Vice President and Chief Nuclear Officer (CNO), who has the overall authority and responsibility for Nuclear Generation, and the executive for Operations Support, whose responsibilities include Nuclear Decommissioning. Nuclear Decommissioning is controlled under a separate description of the quality assurance program as identified in the Defueled Safety Analysis Report for that facility.

As described in Section 17.3.1.2.3, Nuclear Generation receives support services from other organizations, reporting to the Chief Operating Officer, having responsibilities for supply chain, environmental, health and safety and non-nuclear generation activities including: fossil and hydro generation; coal combustion product strategic management; and fuels and system optimization. Services also are provided to Nuclear Generation by Group Executives, reporting to the President and Chief Executive Officer, responsible for the following: electrical distribution; support for the emergency response communications; and Information Technology Services. The interfaces with organizations providing those activities are described in Section 17.3.1.2.3. As such, the attainment of quality rests with those assigned the responsibility of performing the activity. The verification of quality is assigned to qualified personnel independent of the responsibility for performance or direct supervision of the activity. The degree of independence varies commensurate with the activity's importance to safety.

The policies described in this document are implemented through departmental program manuals and procedures, and are, thereby, available to all levels of management.

17.3.1.2.2 Nuclear Generation

Nuclear Generation has direct line responsibility for Duke Energy Corporation nuclear station operations. Nuclear Generation is responsible for achieving quality results during engineering, preoperational testing, operation, testing, maintenance and modification of the Corporation's nuclear stations and with complying with applicable codes, standards and NRC regulations. The functions of Nuclear Generation are directed by the CNO.

The CNO formulates, recommends, and carries out plans, policies, and programs related to the nuclear generation of electric power. The CNO is informed of significant problems or

occurrences relating to safety and QA through established administrative procedures and participates directly in their resolution, where necessary.

Nuclear Generation is organized into three divisions. The activities of each division are directed by an executive who reports to the CNO. The divisions are Nuclear Corporate, Nuclear Oversight, and Nuclear Operations.

The CNO has the organizational flexibility to reassign responsibilities, within the limits specified in the following section, between the standard divisions to provide added focus on areas determined to need increased management attention. This flexibility includes both the ability to consolidate divisions or to identify new divisions. The actual organization in-place is defined in a controlled document containing the fleet operating model.

a) NUCLEAR CORPORATE

The senior executive(s) reports to the CNO and is responsible for Corporate Governance and providing support functions to the Nuclear Sites in the following areas: Nuclear Engineering; Nuclear Regulatory Affairs; Nuclear Support Services; Nuclear Protective Services; Nuclear Operations; Nuclear Corporate Organizational Effectiveness; Nuclear Training; and Emergency Preparedness.

The organizational structure for these functions may vary based on near-term activities and the strategic importance of our fleet initiatives, in our continuing efforts to set and achieve industry-leading operational and outage performance. These functions are primarily off-site located in the Nuclear General Office (NGO).

NUCLEAR ENGINEERING

Nuclear engineering provides broad engineering leadership and technical support to the nuclear sites with emphasis on generic issues and consistent practices, providing expertise in safety assessment with technical support in the areas of risk assessment, radiological engineering, and safety analysis; fuel management with leadership and technical support in the areas of fuel supply, spent fuel management, reactor core mechanical and thermal hydraulic analysis; the fleet electrical and procurement engineering with technical support in the areas of procurement engineering, nuclear process systems, and electrical systems and analysis; and programs and components support in the areas of steam generator inspections and maintenance, engineering programs, component engineering, material failure analysis and materials science, equipment reliability, and ASME Code inspections and testing. Nuclear engineering provides support to Site engineering for contracts and engineering related to fleet and nuclear site major project modifications.

Nuclear engineering provides record storage and document management services, technology planning, project control and technical support for information technology applications and systems such as equipment databases, applications, infrastructure, and plant process information systems.

Nuclear engineering is also responsible for Nuclear Development, which includes the licensing actions needed in support of new nuclear site development under 10 CFR Part 52. Responsibilities also include engineering oversight of contractors, site layout, staffing and program development, and operational readiness. Nuclear Development activities are controlled under a separate description of the quality assurance program as identified in the UFSAR for those facilities.

NUCLEAR MAJOR PROJECTS

Nuclear major projects provides project management for select projects critical to the success of the Nuclear Generation Department. This responsibility includes scope development,

estimating, planning and scheduling, project controls, timely and accurate financial reporting, contract management, and execution of assigned projects.

NUCLEAR REGULATORY AFFAIRS

Nuclear regulatory affairs provides fleet support to and governance of the site regulatory affairs and licensing activities to help improve overall fleet performance.

NUCLEAR SUPPORT SERVICES

Nuclear support services provides fleet support to the nuclear sites for laboratory, calibration, and select maintenance and refueling activities.

NUCLEAR PROTECTIVE SERVICES

Nuclear protective services provides access authorization support to the nuclear sites security organization. Nuclear protective services is responsible for governance of the site security functions, providing assistance to help improve overall fleet performance.

NUCLEAR OPERATIONS

Nuclear operations is responsible for governance of the nuclear site operating organizations, providing assistance to promote improvements to overall fleet performance.

NUCLEAR CORPORATE ORGANIZATIONAL EFFECTIVENESS

Nuclear corporate organizational effectiveness is responsible for governance of the nuclear site performance improvement organizations, providing assistance to promote improvements to overall fleet performance through the corrective action and self-assessment programs. This group also supports implementation of the corrective action and self-assessment programs by the Nuclear Corporate Organization.

NUCLEAR TRAINING

Nuclear training is responsible for governance of the nuclear site training organizations, providing assistance to promote improvements to overall fleet performance. This group also supports implementation of the training programs by the Nuclear Corporate Organization.

EMERGENCY PREPAREDNESS

Emergency preparedness is responsible for governance of the nuclear site emergency response organizations, providing assistance to promote improvements to overall fleet performance.

b) NUCLEAR OVERSIGHT

The executive for Nuclear Oversight (NOS) reports to the CNO and is located in the NGO. NOS consists of both site assigned and NGO located personnel. NOS provides oversight of the NGO, Departmental Interfaces, and the nuclear sites with QA program audits, vendor quality, and quality control. In addition, NOS coordinates the off-site review board, which provides an advisory function to senior management. NOS also provides oversight of Nuclear Development and Nuclear Decommissioning through QA program audits. The NOS executive has the authority and organizational freedom to: identify quality problems, initiate, recommend or provide solutions to quality problems through designated channels, verify the implementation of solutions to quality problems, and ensure cost and schedule do not influence decision making involving quality. This includes full access to Nuclear Development and Nuclear Decommissioning and all levels of management up to and including the Chief Executive Officer.

The NOS executive has primary ownership of the department QA program description (this document) and is responsible for interpretation and resolution of QA issues.

If significant quality problems are identified, NOS personnel have the authority to stop work as discussed in Section 17.3.1.4 pending satisfactory resolution of the identified problem.

Also reporting to the executive for NOS is Employee Concerns, which investigates concerns identified through the Employee Concerns Program to determine their validity and initiate corrective actions as appropriate. Employee Concerns also promotes the Safety Conscious Work Environment (SCWE) Program and is sensitive to SCWE concerns during investigations.

c) NUCLEAR OPERATIONS

The executive for Nuclear Operations reports to the CNO and is located in the NGO. This executive is responsible for the safe operation of the nuclear stations. Reporting to this executive are the executives for the operation of the nuclear stations.

The organization structure for each site is controlled by the site's UFSAR, which may vary from the following generic description. Reporting to the site executive for each nuclear station is a Nuclear Plant Manager who is assigned the direct responsibility for the safe operation of the facility including operations, maintenance, work management, radiation protection, chemistry, and environmental services. Also reporting to the site executive is a site Engineering manager; a site Training manager; and an Organization Effectiveness manager, typically having responsibility for regulatory affairs, emergency preparedness, performance improvement, and procedures. Each site executive also has a Security manager assigned to provide services to the site. The qualification requirements for the Nuclear Plant personnel are in accordance with the provisions of ANSI N18.1 or ANS 3.1 as identified in each site's UFSAR and Technical Specifications.

17.3.1.2.3 Department Interfaces

Quality related activities performed by departments other than Nuclear Generation are identified by and conducted in accordance with controls identified in approved departmental interface agreements. The following are generic descriptions of those other corporate departments and the services they provide. These generic organizations are referred to, as appropriate, within this document; however, approved departmental interface agreements establish and define the applicability of the QAP to the services they provide.

CORPORATE COMMUNICATIONS

Corporate Communications provides support for the nuclear site emergency response organization.

ENVIRONMENTAL HEALTH AND SAFETY

Environmental, Health and Safety provides occupational safety and environmental and laboratory support services.

NUCLEAR FINANCE

Nuclear Finance provides support for the nuclear sites in the areas of financial planning.

INFORMATION TECHNOLOGY

Information Technology provides a variety of services and technical support to Nuclear Generation for information technology applications and systems such as equipment databases, applications, and infrastructure including the electronic document management system and telecommunication systems.

CUSTOMER OPERATIONS

Customer Operations provides electrical distribution and switchyard engineering, as well as providing electrical maintenance and testing support.

NUCLEAR SUPPLY CHAIN

Nuclear Supply Chain provides procurement services including receipt inspection/testing, storage, and inventory control of materials, parts, and components.

17.3.1.3 Responsibility

The primary responsibility for quality performance, including the identification and effective correction of problems potentially affecting the safe and reliable operation of the Company's nuclear facilities, resides with the line organization. The individuals who constitute Nuclear Generation have full personal and corporate responsibility to assure nuclear power plants are designed, constructed, maintained, tested and operated in a manner to protect the public health and safety; and to assure the effectiveness of the QAP.

Appropriate procedures are developed, approved by the responsible implementing manager, issued for use, and used at the location where the prescribed activity is performed, where appropriate. Managers assure that their personnel are adequately trained for their jobs and they have the experience and education required to carry out their assigned responsibilities. These managers ensure that adequate resources and procedures are available for correctly implementing the work activities. Sufficient personnel, including necessary resources, are available and trained prior to performing activities that affect quality.

Independent inspections are conducted to verify specific critical quality attributes. Individuals performing these inspections have access to necessary information to ensure that activities and equipment meet established acceptance criteria.

NOS is responsible for monitoring and auditing activities that are performed by the line organization for, or in support of, Duke Energy Corporation’s Nuclear Plants and Nuclear Generation. These activities include those performed at the individual plant sites, corporate offices, and other Nuclear Generation locations. NOS performs audits to verify that applicable elements of the quality assurance and other regulatory required programs have been developed, documented and effectively implemented in accordance with specified requirements. NOS monitors supplier performance to assure implementation of the applicable quality assurance program requirements. A periodic briefing of NOS activities, along with any potential findings and recommendations, is presented to the CNO.

The CNO is responsible for ensuring that the results and effectiveness of the nuclear oversight program are regularly evaluated as discussed in Section 17.3.3.3.6, Independent Audit of QA Functions.

17.3.1.4 Authority

Personnel involved in quality activities have the authority and responsibility to stop work if they discover deficiencies in quality.

Personnel performing the QA functions have the authority and responsibility to stop unsatisfactory work and to assure the item/activity is controlled to prevent further processing, delivery, installation, or use until authorized by appropriate management.

Procedures outline the methodology for resolution of disputes involving quality and nuclear safety issues arising from a difference of opinion between identifying personnel and other groups.

17.3.1.5 Personnel Training and Qualification

Both on-site and off-site personnel who perform activities affecting quality (implement requirements of the QAP) are indoctrinated and trained such that they are knowledgeable and capable of performing their assigned tasks.

Training programs and reviews ensure that proficiency of personnel performing activities affecting quality is achieved and maintained by training, examining, and/or certifying, as appropriate.

Training programs are modified to reflect station engineering changes and changes in procedures.

Personnel training and qualification records are to be maintained in accordance with procedures.

Personnel within the Operating organization performing duties of a licensed operator are indoctrinated, trained, and qualified as required by 10 CFR Part 55 Operators' Licenses.

17.3.1.6 Corrective Action

It is the policy of Duke Energy Corporation to seek improvement in each nuclear plant's performance as well as in the performance of supporting Departments. Duke Energy Corporation has established a corrective action process whereby all personnel are expected to assure conditions adverse to quality are promptly identified, controlled, and corrected. Individuals are encouraged to voluntarily report events, near misses, and potential problems. In the case of significant conditions adverse to quality, the process assures that the cause of the condition is determined and action be taken to preclude repetition. This process also provides for trending of problems to detect adverse trends in quality performance, including reporting of results to appropriate levels of management.

Management will emphasize to all levels in the organization the importance of identifying and effectively correcting situations that can adversely affect human and equipment performance. An important aspect of this program is the assignment of qualified personnel to accurately evaluate equipment/human performance problems, implement appropriate corrective actions, and verify corrective action adequacy.

Management is responsible for fostering a positive environment that encourages the self-identification of adverse conditions and trends. This includes assuring the process is administered to correct the problem rather than to establish blame or fault.

License Renewal non-safety-related SSCs that are subject to an aging management review are included in the scope of the corrective action program.

Section 17.3.2.13, Corrective Action provides additional detail.

17.3.1.7 Regulatory Commitments

The operation of nuclear plants is accomplished in accordance with the U.S. Nuclear Regulatory Commission (NRC) regulations specified in Title 10 of the U.S. Code of Federal Regulations.

The operation of the Company's nuclear power plants is in accordance with the terms and conditions of the facility operating license issued by the NRC.

The QAP provides for compliance with QA regulatory guides and the related codes and standards as identified in Table 17-1, Conformance with QA Regulatory Guides and Industry Standards.

The requirements of this section (17.3) may provide additional details for implementation of exceptions to these Regulatory Guides and codes and standards.

Changes to the description of the QAP contained in this document are controlled in accordance with 10 CFR 50.54(a).

Table 17-2, Site Specific Response to Regulatory Guides and Industry Standards, identifies additional Regulatory Guides that relate to implementation of the QAP but the implementation is site specific and controlled with each site's UFSAR in accordance with 10 CFR 50.59.

17.3.2 PERFORMANCE/VERIFICATION

17.3.2.1 Methodology

Personnel performing work activities are responsible for achieving the acceptable level of quality.

Personnel performing verification activities are responsible for verifying the achievement of acceptable quality.

Work is accomplished and verified using instructions, procedures, or appropriate means that are of a detail commensurate with the activity's complexity and importance to safety. The implementing manager is responsible to ensure instructions and procedures provide adequate detail for achieving an acceptable level of quality.

Criteria that define acceptable quality are specified in procedures and/or other documents, and verification, when required is performed against these criteria.

17.3.2.2 Design Control

In order to provide for the continued safe and reliable operation of a nuclear station's nuclear safety related structures, systems and components, design control measures commensurate with those applied to the original design are implemented during the operational phase to assure that the quality of such structures, systems and components is not compromised by engineering changes.

Nuclear Engineering is responsible for design activities during the operational phase of nuclear stations to Nuclear Generation. Nuclear Engineering will assure that the organization performing design has access to pertinent background information, including an adequate understanding of the requirements and intent of the original design, and that the organization has demonstrated competence in applicable design areas.

Procedures and instructions for design control during the operational phases for nuclear safety related items provide controls to assure the design is performed in accordance with approved criteria, and that deviations and nonconformances are controlled.

Procedures identify the responsibilities of the various individuals/organizations involved in nuclear safety related engineering changes. The assignment of responsibility for the evaluation and design of a particular engineering change to a specific individual/organization is documented. Procedures addressing the control, including the review, approval, release, and distribution of engineering changes, address the communication of information between internal and external individuals/organizations and, where appropriate, require documentation of such communications.

The procedures include measures to assure that the design selected to accomplish a necessary or desirable change does not create "new" problems in off-normal modes of operation or in adjacent inter-tied systems. For each proposed nuclear safety related engineering change, the individual/organization assigned responsibility for evaluation and design of the engineering change considers the following in the design of the engineering change:

a. Necessary design analyses, e.g., physics, stress, thermal, hydraulic, accident, etc.
b. Compatibility of materials.
c. Accessibility for operation, testing, maintenance, inservice inspection, etc.

d. Necessary installation and periodic inspections and tests, and acceptance criteria therefore.
e. The suitability of application of materials, parts, components, and processes that are essential to the function of the structure(s), system(s) and/or component(s) to be modified.
f. Materials, parts, and equipment which are commercial grade items or which have been previously approved for a different application are evaluated for suitability prior to selection.

Engineering changes are then executed in accordance with approved checklists, instructions, procedures, drawings, etc., appropriate to the nature of the work to be performed. These checklists, instructions, procedures, drawings, etc., include criteria for determining the acceptability of the engineering change.

Any errors or deficiencies found in the design process or the nuclear safety related design itself are documented and corrected using the corrective action program.

Prior to a structure, system, or component that has been modified by engineering change being declared operable and returned to service, the procedures governing the operation are reviewed and revised as necessary. If the engineering change significantly alters the function, operating procedure, or operating equipment, then additional training is administered as necessary.

Adequate identification and retrievable documentation of station engineering changes is retained for the life of the station.

Engineering changes are reviewed to determine whether or not the modification is a change in the facility as described in the UFSAR, involves a change to the Technical Specifications, or requires a license amendment in accordance with 10 CFR 50.59(c)(2). Engineering changes which are determined to require a license amendment are reviewed by the On-Site Review Committee and must be authorized by the NRC prior to implementation.

17.3.2.3 Design Verification

Procedures require that the adequacy of nuclear safety related designs and design changes be verified by the performance of design reviews, alternate calculations, or qualification testing. The control measures specified in the plan for control of design verification activities are as follows:

a. Personnel responsible for design verification do not include the original designer or the designer's immediate supervisor unless the immediate supervisor is the only one capable of verifying the design, in which case additional requirements apply as identified below.
b. Procedures identify the positions or organizations responsible for design verification and define their authority and responsibility. Procedures also provide guidelines as to the method of design verification to be used. Unless otherwise specified, design verification is performed by the method of independent design reviews and includes verification that UFSAR commitments have been addressed.
c. Qualification tests to verify the adequacy of the design are performed using the most adverse specified design conditions.
d. Design changes are reviewed to assure that design parameters are defined and that inspection and test criteria are identified.
e. Design verification is completed prior to relying upon the component, system or structure to perform its function or before its installation becomes irreversible.

The use of the originator's immediate supervisor for verification is:

1) restricted and justified to special situations where the immediate supervisor is the only individual capable of performing the verification
2) the need is individually documented and approved in advance by the supervisor's management and
3) the frequency and effectiveness of the supervisor's use as design verifier are independently verified to guard against abuse.

The individuals assigned to perform the design verification of a nuclear safety related document have full authority to withhold approval of the document until every question concerning the work has been resolved. If required, the matter can be carried up to the CNO for resolution.

17.3.2.4 Procurement Control

Duke Energy Corporation maintains a program for supplier evaluation, results of supplier evaluation, surveillance of suppliers, supplier furnished records, certificates of conformance, effectiveness of supplier quality control, and the purchase of spare or replacement parts. The Duke Energy Corporation QAP requires the control of nuclear safety related items or services purchased from a supplier, subsupplier, or consultant through appropriate processes and specific procurement documents.

Procedures identify the responsibilities and requirements for the control of procurement documents and ensure that purchased material and services are of acceptable quality. Procurement of QA items is to the quality program requirements in effect at the time of purchase.

Nuclear safety related material, equipment and services procured as basic components may only be procured from qualified suppliers. Supplier qualification is accomplished by NOS evaluation of the supplier QA program. An audit or pre-award survey is performed by NOS when required. The audit or pre-award survey is carried out in accordance with a comprehensive audit checklist to determine the ability of the supplier QA program and manual(s) to meet applicable criteria of 10 CFR 50, Appendix B; 10 CFR 21; the ASME Code, when required, and any other codes and standards determined to be appropriate for the prospective scope of supply.

The above requirements apply to procurement of services and items as basic components, including obtaining a Commercial Grade Item dedicated as basic component from an approved third party dedicator. The remainder of this section addresses alternate requirements for purchase of Commercial Grade Items or services.

17.3.2.4.1 Commercial Grade Dedication

When nuclear safety related items/services are not supplied as a basic component and meet the definition of Commercial Grade Item, the item may be procured without the performance of a supplier qualification audit or the existence of a documented supplier QA program. These Commercial Grade Items used in nuclear safety related applications require evaluation, dedication and approval by Nuclear Generation personnel. Commercial Grade Dedication is performed using NRC endorsed industry standards EPRI NP-5652, EPRI Technical Report 102260, EPRI 3002002982, and EPRI Technical Report 1025243 consistent with the NRC exceptions or clarifications identified in GL 89-02, RG 1.123, RG 1.164, and RG 1.231 providing the endorsements. Supplier selection for Commercial Grade Items is the responsibility of the responsible engineering personnel or designated supply chain personnel as identified in

procedures. These items are subject to the same verification and checking process for suitability of application as other nuclear safety related items.

17.3.2.4.2 Commercial Grade Dedication of Laboratory and Testing Services

As identified in NEI 14-05A, commercial grade calibration or testing services may be procured from commercial laboratories based on the laboratory's accreditation to ISO/IEC-17025 by an Accreditation Body (AB) which is a signatory to the International Laboratory Accreditation Cooperation (ILAC) Mutual Recognition Arrangement (MRA) without performing commercial grade surveys as part of commercial grade dedication provided all of the following are met:

1. A documented review of the supplier's accreditation is performed and includes a verification of the following:
    a. The calibration or test laboratory holds accreditation by an accrediting body recognized by the ILAC MRA. The accreditation encompasses ISO/IEC-17025:2005, "General Requirements for the Competence of Testing and Calibration Laboratories."
    b. For procurement of calibration services, the published scope of accreditation for the calibration laboratory covers the needed measurement parameters, ranges, and uncertainties.
    c. For procurement of testing services, the published scope of accreditation for the test laboratory covers the needed testing services including test methodology and tolerances/uncertainty.
2. The purchase documents require that:
    a. The service must be provided in accordance with their accredited ISO/IEC-17025:2005 program and scope of accreditation.
    b. As found calibration data must be reported in the certificate of calibration when calibrated items are found to be out-of-tolerance. (for calibration services only)
    c. The equipment/standards used to perform the calibration must be identified in the certificate of calibration. (for calibration services only)
    d. The customer must be notified of any condition that adversely impacts the laboratory's ability to maintain the scope of accreditation.
    e. Additional technical and quality requirements, as necessary, are specified for verification at receipt based upon a review of the procured scope of services, which may include, but are not necessarily limited to, tolerances, accuracies, ranges, and industry standards.
3. It is validated, at receipt inspection as part of the commercial grade dedication process, that the laboratory's documentation certifies that:
    a. The contracted calibration or test service has been performed in accordance with their ISO/IEC-17025:2005 program, and has been performed within their scope of accreditation, and
    b. The purchase order's requirements are met.

17.3.2.5 Procurement Verification

Duke Energy Corporation procurement documents are prepared, reviewed, approved, and controlled in accordance with procedures to assure that requirements are correctly stated, inspectable, verifiable, and controllable, and there are adequate acceptance/rejection criteria. Procurement documents are reviewed by personnel knowledgeable in applicable technical and quality requirements, and documentary evidence of that review and approval is retained and available for verification.

As required by procurement criteria, in order to assure that material and equipment are fabricated in accordance with applicable requirements, supplier reviews are performed by Vendor Quality. Those reviews may include witnessing of tests, observation of fabrication checkpoints, and documentation review.

Receipt inspections are performed by qualified inspectors in accordance with procedures to assure that:

1. Materials, equipment, or components are properly identified and correspond with associated documentation.
2. Inspection records or certificates of conformance attesting to the acceptance of materials, equipment, and components are completed and are available prior to installation or use.
3. Materials, equipment, and components are inspected and judged acceptable in accordance with predetermined inspection instructions prior to installation or use.
4. Items not meeting applicable requirements are identified and controlled until proper disposition is made.

The process ensures that required documentation of compliance is received and available on site and procurement, inspection, and testing requirements are satisfied before the item is placed in service.

As identified in Section 17.3.2.4.2, specific to the commercial grade dedication of Calibration Testing and Laboratory Services, receipt inspection verifies that:

- The laboratory's documentation certifies that:
    - contracted calibration or test service has been performed in accordance with their ISO/IEC-17025:2005 program,
    - has been performed within their scope of accreditation, and
    - the purchase order's requirements are met.
- Additional technical and quality requirements are met.

17.3.2.6 Identification and Control of Items

Procedures require spare or replacement parts to be subject to QAP controls, codes and standards, and technical requirements which ensure they are suitable for their intended service. Items accepted or released are identified as to their inspection status prior to forwarding them to a controlled storage area or releasing them for installation or further work. Bulk items will not require individual accept tags; however, status of unacceptable bulk items will be so indicated.

Identification requirements for materials, parts and components important to nuclear safety are stated in specifications, drawings and purchase documents.

Control of material, parts and components is governed by approved procedures.

Following QA receipt inspection, materials, parts and components which are determined to be acceptable are assigned an identifying designation such as a unique tracking number in order to provide traceability of each item. This traceability is maintained for nuclear safety related items. In the event that the identification of an item becomes lost or illegible, the item is considered nonconforming and not utilized until proper resolution of the nonconformance.

Consumables utilized in nuclear safety related structures, systems and components are subject to appropriate controls as described in procedures.

17.3.2.7 Handling, Storage, and Shipping

Procedures utilized by suitably trained individuals define requirements for the control of the handling, storage, and shipping of safety-related items. These procedures require measures to be taken to ensure special handling, storage, cleaning, packaging, shipping, and preservation requirements are established to control these activities in accordance with design and specification requirements to preclude damage, loss or deterioration by environmental conditions such as temperature or humidity. Nuclear safety related materials, parts and components are handled, stored, issued and shipped in such a manner that the serviceability and QA traceability of an item is not impaired.

Nonconforming items are identified, segregated, or otherwise controlled in such a manner as to preclude their inadvertent substitution for and use as conforming materials parts and components.

17.3.2.8 Test Control

The QAP addresses both preoperational and periodic (surveillance) testing. The program requires that such testing associated with nuclear safety related structures, systems and components demonstrate that the items will perform satisfactorily in service. Testing activities are accomplished in accordance with approved, written procedures. Testing schedules are provided and maintained in order to assure that all necessary testing is performed and properly evaluated on a timely basis. Test controls include requirements on the review and approval of test procedures, and on the review and approval of changes to such procedures, as discussed in Section 17.3.2.14, Document Control.

Modifications, repairs, and replacements are accomplished in accordance with the original design and testing requirements or acceptable alternatives.

17.3.2.9 Measuring and Test Equipment Control

The organizations performing nuclear safety related work activities have the responsibility to assure the required accuracy of tools, gauges, instruments, radiation measuring equipment, non-destructive testing equipment and other measuring and test devices affecting the proper functioning of nuclear safety related structures, systems and components and that a program of control and calibration for such devices is provided.

Procedures define requirements for the control of measuring and test equipment (M&TE) used. These procedures include requirements to establish procedures for the calibration technique and frequency, maintenance, and control of measuring and test equipment. The requirements include the following:

a. M&TE is assigned permanent, identifying designations. M&TE is identified and traceable to the calibration test data.
b. M&TE is calibrated at prescribed intervals against certified equipment having known, valid relationships to nationally recognized standards or where national standards do not exist, provisions are established to document the basis for the calibration. The calibration interval is based on the applicable manufacturer's recommendations. If experience shows that the manufacturer's recommendations are not appropriate, the calibration interval is changed as necessary. One or more of the following may be used to adjust intervals: 1. Technical Specifications; 2. Required accuracy; 3. Intended use; 4. Frequency of usage; 5. Stability characteristics; 6. Other conditions affecting

measurement. In lieu of specified intervals, infrequently used M&TE may be calibrated immediately before and after use.

c. Status of calibration for M&TE is provided through the use of tags, stickers, labels, routing cards, computer programs, or other suitable means. The status indicators indicate the date recalibration is due or the frequency of recalibration.
d. M&TE failing to meet calibration specifications is identified through the use of tags, stickers, labels, routing cards, computer programs, or other suitable means, showing the date of rejection, the reason for rejection and the identification of the individual rejecting the device. "Accepted" and "Rejected" calibration identification is sufficiently different to preclude confusion between them.
e. Items and processes determined to be acceptable based on measurements made with M&TE that subsequently cannot be demonstrated to meet calibration specifications are re-evaluated to determine the validity of previous inspections and test results and the results of the evaluation documented.
f. M&TE is stored under conditions which are in accordance with, or more conservative than, the applicable manufacturer's recommendations.
g. M&TE is issued under the control of responsible personnel so as to preclude unauthorized use.
h. M&TE is shipped in a manner that is in accordance with, or more conservative than, the applicable manufacturer's recommendations.
i. Records are maintained for each item of M&TE identifying the device designation, the calibration frequency and specifications. Records are maintained reflecting current calibration status, the date of calibration, the date the next calibration is due, and the identification of the individual who was responsible for performing the calibration.
j. As a rule, the calibration program achieves a minimum ratio of 4-to-1 calibration standard accuracy to measuring and test equipment accuracy is used. However, well defined and documented measurement assurance techniques or uncertainty analysis may be used to verify the adequacy of the measurement process. See site specific requirements for other exceptions to the 4:1 rule.

M&TE is selected to assure accurate measurement (i.e., to overcome inherent inaccuracies associated with environment, human error, equipment, etc.).

17.3.2.10 Inspection, Test, and Operating Status

Procedures define requirements for the identification and control of the inspection, test, and operating status of safety-related structures, systems, and components, to assure that equipment operating status is clearly evident, and to prevent inadvertent operation of nuclear safety related structures, systems and components which, if operated, could cause damage to other equipment/systems or to personnel

These measures include the use of checklists, computer programs, logs, stickers, tags, labels, record cards, and test records to indicate the acceptable operating status of installed equipment. Where appropriate, an independent verification of the correct implementation of such identification measures is performed.

When tags, labels or stamps are utilized for the identification of equipment status, the issuance and removal thereof is documented in order to assure proper control of such identification measures. Also, procedures require that the operability of an item removed from operation for maintenance or testing be verified prior to returning the item to normal service.

Selected plant procedures and subsequent revisions receive separate technical review to ensure required inspections, tests, and other critical operations are included.

17.3.2.11 Special Process Control

Procedures define requirements for the control of special processes, such as welding, heat treating, nondestructive examination (NDE), coatings, and chemical cleaning when the performance of such processes affects the proper functioning of nuclear safety related structures, systems, and components.

Procedures require that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures. These personnel and procedures are to be qualified in accordance with applicable codes, standards, and specifications as described in procedures.

Qualification records of special process procedures and personnel performing special processes are maintained and available for verification.

17.3.2.12 Inspection

Procedures define requirements for an inspection program to verify conformance to performance and quality requirements specified for nuclear safety related structures, systems, and components.

Inspections are performed by personnel who are not directly responsible for performing or supervising the activity being inspected. Inspection personnel are qualified in accordance with applicable codes and standards, and their qualifications and certifications are maintained current.

Inspections are performed in accordance with procedures or other documents, which provide for the following:

1. Identification of individuals or groups responsible for performing the inspections
2. Identification of characteristics and activities to be inspected
3. Acceptance criteria
4. Inspection techniques
5. Recording the results of the inspection, review of the results, and identification of the inspector
6. Indirect control by monitoring of processing methods, equipment, and personnel when direct inspection is not possible

Mandatory inspection hold points are included in the documents addressing the activities being performed, as necessary and work does not proceed until satisfactory completion of the required inspection.

When acceptance criteria are not met, the condition will be documented in accordance with the corrective action program procedures and work does not proceed until satisfactory disposition of any item not meeting the acceptance criteria and satisfactory completion of any required re-inspection.

Modification, repairs, and replacements are inspected in accordance with the original design and inspection requirements or acceptable alternatives.

17.3.2.13 Corrective Action

Station personnel are responsible for the implementation of the QAP as it pertains to the performance of their activities. Specific to this responsibility is the requirement for informing the

responsible supervisory personnel and/or for taking appropriate corrective action whenever any deficiency in the implementation of the requirements of the program is determined.

Procedures define requirements for a corrective action program that charges personnel working at or supporting the nuclear plants with the responsibility to identify adverse conditions (including conditions adverse to quality). Conditions adverse to quality are identified through inspections, assessments, tests, checks, and review of documents. Procedures require that conditions adverse to quality be corrected. In the case of significant conditions adverse to quality, the procedures assure that the cause of the condition is determined and action be taken to preclude repetition.

Significant conditions adverse to quality are reported to appropriate management for review and evaluation.

Violations of Technical Specifications, safety limit violations, and other reportable events are investigated to correct the condition and to support the reporting requirements of 10 CFR 50.73(b). Reports of such investigations are reviewed by a knowledgeable individual other than the individual who prepared the report.

Periodic reviews and evaluations of adverse conditions are performed to identify and correct adverse trends.

17.3.2.14 Document Control

Procedures define requirements for the development, review, approval, issue, use, revision, and control of documents. These procedures define the scope of which documents are to be controlled. These activities include measures to control the issuance of documents such as, instructions, procedures, and drawings, and changes thereto, which prescribe activities affecting quality.

A document control system has been established to identify the current revision number of instructions, procedures, specifications, and drawings. This system includes provisions to ensure that superseded documents are controlled to prevent inadvertent use.

Controlled documents are to be distributed to and used by the person performing the activity in accordance with procedures. These controlled documents are distributed electronically. Hardcopy distribution, if required, is by distribution indices.

Procedures require the identification of those individuals or organizations responsible for reviewing, approving, and issuing documents and revisions thereto. The required reviews include reviews verifying that changes to the procedures, tests or experiments do not involve a change in the Technical Specifications or otherwise require prior NRC approval.

In addition to procedures and engineering documents (e.g. specifications and drawings), the following are considered to be controlled documents:

- The station Facility Operating License and Technical Specifications
- Updated Final Safety Analysis Reports
- Process Control Program
- Offsite Dose Calculation Manual
- Radiological Effluent Controls of the UFSAR, and radwaste treatment systems

Procedures established for operational phase activities include:

1. Operating Procedures
2. Alarm Responses

3. Radiation Protection Procedures
4. Maintenance Procedures
5. Instrument Procedures
6. Chemistry Procedures
7. Process Control Program Implementing Procedures
8. Periodic Test Procedures
9. Abnormal Procedures
10. Emergency Procedures
11. Emergency Response Procedures
12. License Renewal Aging Management Program

In lieu of the two year procedure review prescribed by ANSI N18.7-1976 Section 5.2.15, Duke Energy Corporation has programmatic controls in place to continually identify procedure revisions which may be needed to ensure that procedures are appropriate for the circumstance and are maintained current. These controls include the following:

- The procedure revision process includes a mechanism for procedure users to request changes to the procedures.
- The modification process requires that procedures be reviewed to determine the effects of a planned plant modification.
- Procedures are reviewed for adequacy based upon lessons learned from the operating experience program, training programs, emergency plan reviews, drills and exercises, and normal use.
- The work control process includes pre job review process and a procedure adherence policy requiring that, if procedures cannot be implemented as written, the job be stopped and the procedure be revised or the situation resolved prior to work continuing.

The line organization performs a biennial self-assessment of the procedure process to assure their procedures are maintained current. This assessment includes a requirement to evaluate potential adverse trends in the procedure change process to ensure that changes required to maintain procedures current and technically accurate are being implemented in a timely manner.

17.3.2.15 Records

Each nuclear station is required to maintain adequate identifiable and retrievable QA records. The QAP requires that sufficient records be maintained to provide documentary evidence of the quality of items and the accomplishment of activities affecting quality.

Procedures define requirements for the identification, collection, and storage of quality assurance records.

The program for storage of records on microfilm, dual storage or in electronic format meets the preservation requirement for the retention of QA Records.

Media used for retention of records include (but are not limited to): microfilm, compact disk recordable (CD-R), and magnetic media including videotape, computer tape, optical disks, and hard disk storage. Electronic records retention is an integral component of the Record Retention Program, approved by the management position responsible for Nuclear Generation Department records. The format used must be capable of producing legible, accurate, and complete documents supporting the required retention period. Electronic approval and

authorization procedures are established to assure that only those persons authorized grant the required approvals.

For creation and maintenance of on-line electronic records, Duke Energy Corporation follows the Nuclear Information and Records Management Association (NIRMA) Technical Guides as identified in Table 17-1, Conformance with QA Regulatory Guides and Industry Standards.

There is no requirement to convert records stored on media including hardcopy, microfilm, compact disk recordable (CD-R), and magnetic media including videotape, computer tape, and optical disks to on-line electronic records. Those records may be maintained in their current form as long as retrieval technology and media life support the continued use of the media. If records stored on one media are to be converted to a new media, the records stored on the old system's media are acceptably converted into the new system before the old system is taken out of service. This includes verification of the records so copied are complete and accurate in the new system.

Records are identifiable and retrievable through the use of indexes and filing systems, which are required by the program.

Procedures are required to be developed to indicate responsibilities and retention periods.

The actual retention times for the various QA records are in accordance with corporate retention policies. The development of these retention policies includes consideration of applicable requirements, including those of the Code of Federal Regulations, a station's Technical Specifications, established national codes and standards, and regulatory guidance as listed in Table 17 1, Conformance with QA Regulatory Guides and Industry Standards.

The following is a list of typical QA Records retained for the operational phase:

1. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report. These include: drawings, design specifications, calculations, design analyses, and vendor documents for nuclear safety related structures, systems and components.
2. Records of new and irradiated fuel inventory, fuel transfers and assembly burn-up histories.
3. Radiation monitoring records, including records of radiation and contamination surveys.
4. Personnel radiation exposure records.
5. Records of radioactive releases and waste disposal, records of gaseous and liquid radioactive material released to the environs.
6. Records of component cyclic or transient limits established for the reactor coolant system, reactor vessel, and secondary coolant system.
7. Records of the qualifications, experience and training of appropriate station personnel
8. Records of quality control inspections.
9. Records of reviews performed for changes made to procedures or safety related SSCs or reviews of tests and experiments pursuant to 10 CFR 50.59.
10. Changes to station procedures; including review and approval documentation.
11. Records of meetings of the off-site review committee.
12. Records of Independent Review. These records include on-site review committee meeting minutes.
13. Records of reactor tests and experiments.
14. Records of inservice inspections performed pursuant to Technical Specifications and 10 CFR 50.55a(g).

15. Records of the service lives of all safety-related snubbers (required by Technical Specification) including the data at which the seal service life commences and associated installation and maintenance records.
16. Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date.
17. Records of secondary water sampling and water quality.
18. Records of reviews performed for changes made to the Off-Site Dose Calculation Manual, the Process Control Program, and Radwaste Treatment Systems.
19. Isotopic and physical inventory records of special nuclear materials.
20. Nuclear safety related preoperational testing records.
21. Records such as vendor documentation packages and inspection reports, piping isometric drawings, welding records, etc. compiled during the design and construction of a nuclear station.
22. Approved purchasing documents for items requiring QA certification.
23. Purchase specifications.
24. Records of special processes affecting nuclear safety related structures, systems and components.
25. Records of off-site environmental surveys.
26. Records of environmental qualification.
27. By-product material inventory records.
28. Radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation alarm/trip setpoints.
29. Records of reviews performed for changes made to Radiological Effluent Controls.
30. Records of reviews performed on the Fire Protection Program and implementing procedures.
31. Audit reports and required written responses.
32. Records and logs of facility operation covering time interval at each power level, including: switchboard record, reactor operator logbook, and shift supervisor logbook.
33. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
34. Reports of all reportable and other significant events.
35. Records of surveillance activities, inspections, and calibrations required by Technical Specifications.
36. Records of radioactive shipments.
37. Records of sealed source and fission detector leak tests and results.
38. Records of annual physical inventory of all sealed source material of record.
39. Calibration standard records and Measuring and Test Equipment (M&TE) calibration records.

Dry cask storage records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety must be maintained for the life of the storage mode.

17.3.3 SELF-ASSESSMENT

17.3.3.1 Methodology

Each site executive and the CNO are responsible for ensuring that an environment exists for a strong assessment program at each nuclear site and within Nuclear Generation, respectively.

The overall objective at Duke Energy Corporation is to encourage ownership, involvement, and dedication by each individual supporting Nuclear Generation. This involves continually looking for ways to improve the overall performance and safety at each plant. This approach of identifying and correcting conditions early, requires active support by management and employees.

The Duke Energy Corporation self-assessment process includes the line organization self-assessment activities, independent review activities, and an independent assessment process implemented by NOS that encompasses internal and supplier audits. NOS may perform in-plant reviews and other independent assessments requested by the CNO.

The managers of line organizations are responsible for ensuring that self-assessment activities and processes are implemented within their functions to promote continuous improvements. A process of self-assessment is an attitude by personnel that the Duke Energy Corporation Nuclear Generation is improving on a continual basis. This process, along with an effective corrective action program, ensures that conditions are identified early, corrected promptly and effectively before becoming significant quality or safety problems.

The independent review activities are discussed in Section 17.3.3.2.

As directed by the CNO, an off-site review board periodically performs independent reviews of matters involving the safe operation of its fleet of nuclear power plants. The review addresses matters that plant and corporate management determine warrant special attention, such as plant programs, performance trends, employee concerns, or other matters related to safe plant operations. The review is performed by a team consisting of personnel with experience and competence in the activities being reviewed, but independent (from cost and schedule considerations) from the organizations responsible for those activities. The review is supplemented by outside consultants or organizations as necessary to ensure the team has the requisite expertise and competence. Results are documented and reported to responsible management.

The independent assessment process is to confirm to management that activities affecting quality comply with the QAP and that the QAP has been implemented effectively. The assessment activities are performed in accordance with instructions and procedures by organizations independent of the areas being assessed. This process is discussed in detail in Section 17.3.3.3.

17.3.3.2 Independent Review

The independent review function is provided through a combination of the On-Site Review Committee, Nuclear Oversight, and the line organization executing quality assurance program required reviews as follows:

- Reviews of the independent review subjects are performed by the On-Site Review Committee as described in Section 17.3.3.2.1, On-Site Review Committee.

- Reviews of audit reports, identified in ANSI N18.7-1976 Section 4.5, are performed by management of the audited area and Nuclear Oversight instead of the independent review function.
- Reviews of the corrective actions for significant conditions adverse to quality are performed by appropriate management. Collectively, the On-Site Review Committee and the NOS audit function perform the independent review, identified in ANSI N18.7-1976 Section 5.2.11, for significant conditions adverse to quality.

17.3.3.2.1 On-Site Review Committee

The On-Site Review Committee is responsible to the Nuclear Plant Manager for advice on all plant-related matters concerning nuclear safety. The requirements for personnel, committee composition, meeting frequency, quorum and meeting records are identified in procedures. A general description of these areas is included below. (Note: Each plant may name this function differently. Regardless of the name, these requirements are met.)

In discharging its independent review responsibilities, the On-Site Review Committee keeps safety considerations paramount when opposed to cost or schedule considerations. Should a voting member at a particular meeting have direct responsibility for item under review where a conflict of such considerations is likely, that member is replaced (to fill the quorum) by another voting member not having such potential conflict.

17.3.3.2.1.1 Composition

The On-Site Review Committee is comprised of a minimum number of members as designated by the Plant Manager and detailed in procedures. All members are qualified in accordance with procedure requirements that meet site Technical Specifications. Membership includes representation from at least the following disciplines: Operations, Maintenance, Engineering, Radiation Protection and Chemistry. The On-Site Review Committee collectively has, or has access to, the experience and competence necessary to review the areas of (1) nuclear power plant operations, (2) nuclear engineering, (3) chemistry and radiochemistry, (4) metallurgy, (5) nondestructive testing, (6) instrumentation and control, (7) radiological safety, (8) mechanical and electrical engineering, (9) administrative controls and quality assurance practices, and (10) other fields associated with the unique characteristics of the plant. Consultants may be utilized to provide expert advice as needed.

Alternate chairmen and members may be appointed by the Nuclear Plant Manager to serve on a permanent or temporary basis.

17.3.3.2.1.2 Meetings

The On-Site Review Committee meets commensurate with the scope of activities, but minimal frequency requirements are specified in procedures.

Rules for a quorum are established and adhered to. However, no more than a minority of alternates may participate as voting members at any one time.

17.3.3.2.1.3 Review Topics

In performing its independent review responsibilities, the On-Site Review Committee reviews:

(1) Proposed changes to the facility as described in the UFSAR. This review is to confirm that the regulatory required written evaluation provides adequate bases for the determination that the change does not require a license amendment.
(2) Proposed changes to procedures as described in the UFSAR and tests or experiments not described in the UFSAR. This review is to confirm that the regulatory required written evaluation provides adequate bases for the determination that the test or experiment does not require a license amendment.
(3) Proposed Technical Specifications changes and license amendments, except in those cases where the change is identical to a previously reviewed proposed change.
(4) Licensee Event Reports that are required to be made to the NRC. This review includes results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.
(5) Any other matter related to nuclear safety requested by the Site executive, Plant Manager, selected by On-Site Review Committee members, or referred for review by other organizations.

In addition to reviews of license amendments addressed by (3) above, the On-Site Review Committee should be informed of changes to Site documents that are required to be reported to the NRC. When appropriate, the On-Site Review Committee conducts reviews of such changes to confirm the changes have been prepared and internally approved within license obligations and can be effectively implemented. These documents include the Offsite Dose Calculation Manual (ODCM), the Process Control Program (PCP), the Emergency Plan, and the Security Plan.

The On-Site Review Committee may establish subcommittees or designate organizational units to carry out the reviews. The subcommittees or organizational units report results of reviews for full committee consideration and may recommend items for full committee review as warranted. The reviews by the On-Site Review Committee recognize that the QA Program requires independent technical reviews to be completed including, but not limited to, design verification and reviews of procedures. Those independent technical reviews are conducted commensurate with the importance to nuclear safety of the item or activity. In conducting its review, the On-Site Review Committee is confirming the changes have been prepared and internally approved within license obligations and can be effectively implemented, not re-performing completed technical reviews.

The On-Site Review Committee conducts special reviews and investigations as requested by the Site executive or Nuclear Plant Manager.

17.3.3.2.1.4 Authority

The On-Site Review Committee:

- Recommends to the Nuclear Plant Manager approval or disapproval of items reviewed.
- Renders determinations with regards to whether items (1) through (3) adversely affect safety and if a Technical Specification change or NRC review is required.
- Provides written notification to the Site executive of any disagreements between the On-Site Review Committee and the Nuclear Plant Manager.

The On-Site Review Committee advises the Nuclear Plant Manager on matters related to safe operation and overall performance. The Committee has authority to obtain access to records and personnel as needed to conduct reviews.

17.3.3.2.1.5 Records

The On-Site Review Committee maintains written minutes of each Committee meeting, to include identification of items reviewed, and decisions and recommendations of the Committee. Copies of the minutes are provided to the Site executive, and to other onsite and offsite management responsible for the areas reviewed as necessary. On-Site Review Committee records are retained according to Section 17.3.2.15.

17.3.3.3 Independent Assessment

NOS is responsible for conducting independent assessments of functions and activities affecting the nuclear programs at Duke Energy Corporation locations. NOS monitors and assesses the Company's nuclear programs on a continuing basis. As part of this assessment process, NOS performs audits to verify that applicable elements of the quality assurance and other regulatory required programs have been developed, documented and effectively implemented in accordance with specified requirements. In this section, the words assess, assessment, and their various word forms are used generically to indicate the act of monitoring the performance of the line organization for indications of decline.

NOS, along with the line organization management, monitors functional areas to determine if the required levels of performance are being achieved.

The functions of NOS are to assess line organization performance including the self-assessment and corrective action process. NOS performs these monitoring activities for nuclear safety related functions in operations, engineering, and maintenance.

NOS evaluations, including the results and recommended corrective actions, are reported to senior management.

17.3.3.3.1 Organization

On an exception basis, personnel in NOS may provide assistance to the line organization by participating in emergency preparedness activities, ad hoc committees or analyzing technical issues, if such assistance is deemed to be in the overall best interest of safety and is approved in advance by NOS management.

NOS teams may include peers from other Duke Energy Corporation plants and from the nuclear utility industry, as appropriate, to lend expertise to the assessment process. When subject matter experts from the line organizations are utilized to add specific technical expertise to a specific audit team, the subject matter experts will work under the direction of the audit team leader and not evaluate any documentation for which they had direct responsibility.

Selection of personnel is based on experience and training that establishes that their qualifications are commensurate with the complexity or special nature of the area being audited. The process for qualification of personnel to perform audits is established in procedures.

17.3.3.3.2 Internal Assessment Process

The internal assessment process includes gathering data, analyzing data, focusing on selected issues and identifying deficiencies to desired performance. Data is gathered using performance based techniques during:

a) Observations of work activities
b) Interviews
c) Reviews of documents to gather information (including the use of NRC, INPO, and other agency evaluations)
d) Audits, and
e) Analysis of data and reports (including adverse condition reports, etc.)

NOS personnel have access to records, procedures, and line organization personnel to gather data.

NOS conducts observations of specific activities, and processes on the basis of their impact and importance relative to safety. The schedule is flexible and dynamic to allow the overall assessment process to be changed depending on plant conditions, events, or issues raised by senior management. Assessment activities can be focused on areas most in need of improvement.

Audits are a specific independent assessment activity performed to verify that applicable elements of the quality assurance and other regulatory required programs have been developed, documented and effectively implemented in accordance with specified requirements. Independent Audit activities are selected with flexibility based on various factors. These factors include but are not limited to: importance to safety and reliability, monitoring of performance indicators, time since last audit, plant management perspective, outside agency audits, and problem areas identified from industry and Duke Energy Corporation experience.

Audits are scheduled per the following section.

17.3.3.3.3 Internal Audit Program

The Duke Energy Corporation QAP requires a comprehensive system of planned and periodic internal audits for all phases of station operations and supporting activities.

Periodic audits of activities or records of processes (e.g., welding, maintenance, development of design, record management, or system testing), to verify compliance and effectiveness of the implementation of the QAP are performed. NOS audits are performance based and scheduled based on plant performance and importance to safety but at a frequency not to exceed twenty-four months with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.

The audit system is reviewed periodically and revised as necessary to assure coverage commensurate with current and planned activities. These audits encompass:

- The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions.
- The performance, training and qualifications of the Nuclear Generation Department.
- The results of actions taken to correct deficiencies occurring in facility equipment, structures, or systems that affect nuclear safety; or method of operation that affect nuclear safety.
- The performance of activities required by the QAP to meet the criteria of Appendix B to 10 CFR 50 for activities performed by the Nuclear Generation Department and the interfacing organizations.

- Any other area of nuclear generation considered appropriate by responsible management.
- The Radiological Environmental Monitoring Program and the results thereof.
- The Offsite Dose Calculation Manual and implementing procedures.
- The Process Control Program and implementing procedures for processing and packaging of radioactive wastes.
- The acceptability of a representative sample of station procedures, including the effectiveness of the procedure review and revision program.
- Independent Spent Fuel Storage Installation Activities (reference 10 CFR Part 72).
- Packaging of Radioactive Materials for Off-Site Shipment (reference 10 CFR Part 71).

The scope of each audit is determined by the responsible Lead Auditor, under the direction of NOS management. The lead auditor is responsible for completion of audit checklists and directing the audit team in the performance of the audit. The audit is conducted in accordance with checklists; the scope may be expanded upon by the audit team during the audit, if needed. One or more persons comprise an audit team, one of whom is a qualified lead auditor.

17.3.3.3.3.1 Other Reviews Prescribed by the Code of Federal Regulations

Other reviews prescribed by the Code of Federal Regulations are scheduled and performed per the CFR. The audit frequency extension provisions of Section 17.3.3.3.7 do not apply.

NOS performs the following reviews under the internal audit program:

a. Emergency Preparedness (per 10 CFR 50.54(t))
b. Security (per 10 CFR 50.54(p) and 10 CFR Part 73)
c. Fitness for Duty and Fatigue Rule (per 10 CFR Part 26)

The periodic review of the radiation protection program content and implementation required by 10 CFR 20.1101c may be performed by either the line organization or NOS.

17.3.3.3.3.2 Independent Audit of Fire Protection Program

For sites implementing the fire protection program under provisions of 10 CFR 50.48(c) National Fire Protection Association Standard NFPA 805:

- An independent fire protection audit is performed at least once per 36 months using an outside (external to Duke Energy Corporation) qualified fire protection engineer meeting education and experience requirements for a Professional Member of the Society of Fire Protection Engineers (SFPE).

For the remaining sites, audits of the following functions are completed within a period of 24 months:

- The Facility Fire Protection programmatic controls including the implementing documents.
- The fire protection equipment and program implementation utilizing either a qualified offsite fire protection engineer or an outside independent fire protection consultant. An outside (external to Duke Energy Corporation) qualified fire protection engineer meeting education and experience requirements for a Professional Member of the SFPE shall be used at least every 36 months.
- The audit scope may be combined into a single audit performed on a 24 month frequency with the inclusion of an outside independent qualified fire protection engineer.

17.3.3.3.4 Results

Adverse conditions are reported in accordance with the applicable corrective action program procedure.

Independent audit results are communicated to line management to allow for timely action to address potential problems or recognize strengths and superior performance.

Follow-up is accomplished to assure that corrective action is taken as a result of the audit and that deficient areas are re-audited, when necessary, to verify implementation of adequate corrective actions.

17.3.3.3.5 Supplier Oversight

Supplier QA programs are evaluated and monitored by NOS-Vendor Quality, to assure that QA requirements are met. Supplier QA programs require a system of periodic and planned supplier and sub-supplier audits conducted by persons not directly involved in the activity being audited. Supplier audits are performed on a three year frequency with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.

17.3.3.3.6 Independent Audit of QA Functions

As directed by the CNO, the executive for NOS initiates a program audit of the QA Functions performed by NOS. These functions include the internal audit program, the NOS portions of the supplier oversight program, and maintenance of this document (Quality Assurance Program Description). This program audit is performed within a period of two years with extensions as allowed in Section 17.3.3.3.7 Audit Frequency Extensions.

This audit team consists of qualified individuals, none of which is from the area audited.

The audit is performed with pre-approved checklists, instructions, or plans.

The audit team conducts a post-audit conference with the responsible management of the areas audited to discuss the audit results, including deficiencies. The audit team prepares checklists and the audit report. The report is sent to the executive for NOS.

The executive for NOS and/or responsible management of the area being audited determines the need for corrective action and re-evaluation. Necessary corrective action and re-evaluation are performed as required.

Pertinent correspondence and reports related to the audit are filed.

17.3.3.3.7 Audit Frequency Extensions

Except when the frequency is specified by regulation, the following criteria for extending audit intervals apply:

1) Schedules are based on the anniversary established for each audit.
2) A maximum extension not to exceed 25 percent of the audit interval may be allowed (e.g., audits on a two year frequency may not be extended beyond 30 months, audits on an annual frequency may not be extended beyond 15 months).
3) When an audit interval extension is used, the next audit for that particular audit area is scheduled from the original anniversary.

4) Provision 2) also applies to supplier audits and evaluations except that a total combined time interval for any three consecutive inspection or audit intervals should not exceed 3.25 times the specified inspection or audit interval.

17.3.4 ADMINISTRATIVE CONTROLS RELOCATED FROM TECHNICAL SPECIFICATIONS

Consistent with NRC Administrative Letter 95-06, certain administrative controls from the original station Technical Specifications have been relocated to the Quality Assurance Program. These relocated administrative controls included technical review, independent review, 10 CFR 50.59 review, record retention, and audit requirements. This section provides references to the sections of this document where the administrative controls have been integrated with QAP controls.

17.3.4.1 Technical Reviews

This content provided requirements for technical reviews of station modifications, procedures, tests, and experiments to assure adequacy of nuclear safety related SSCs and associated activities. Those reviews are embedded in the QAP and its committed Standards. See Sections 17.3.2.2, Design Control; 17.3.2.3, Design Verification; 17.3.2.8, Test Control; and 17.3.2.14, Document Control.

As identified by procedures, technical evaluations are performed by personnel qualified in the subject matter to determine the technical adequacy and accuracy of the proposed activity. If interdisciplinary evaluations are required to cover the technical scope of an activity, they will be performed. Technical review personnel are identified by the responsible manager or his designee for a specific activity when the review process begins.

17.3.4.2 10 CFR 50.59 Reviews

The review of station modifications, procedures, tests, and experiments against the requirements of 10 CFR 50.59 is to ensure that changes requiring prior NRC approval are submitted to and approved by the NRC prior to implementation. Provisions are included in Sections 17.3.2.3 Design Verification and 17.3.2.14 Document Control to amplify the need to complete these reviews.

The program for 10 CFR 50.59 reviews is in accordance with NEI 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Evaluations" as endorsed by Regulatory Guide 1.187, November 2000.

This program includes provisions to ensure that individuals have appropriate qualifications prior to completing these reviews. A list of individuals qualified to perform 50.59 evaluations is maintained for each site.

17.3.4.3 Record Retention

The list of typical operational phase records is in Section 17.3.2.15, Records.

17.3.4.4 Audit Types and Frequencies

These are addressed in Section 17.3.3.3.3, Internal Audit Program.

17.3.4.5 On-Site Review Committee

This is addressed in Section 17.3.3.2, Independent Review.

17.3.4.6 Reportable Event Action

Procedures are established to assure events are reviewed and notifications and reports are made as required by Regulations including, but not limited to, 10 CFR Part 21, 10 CFR 50.72, and 10 CFR 50.73.

These procedures require for significant incidents occurring during operation where a safety limit is exceeded, or which could otherwise be related to the nuclear safety of the station, the Site executive is notified, the event is investigated, and a report prepared. These reports:

a) Contain a summary description of the circumstances and information relating to the subject incident.
b) Contain an evaluation of the effects of the incident.
c) Describe corrective action taken or recommended as a result of the incident.
d) Describe, analyze and evaluate any significant nuclear safety related implications of the incident.

17.3.4.7 Independent Safety Engineering Group Functions

Independent Safety Engineering Group (ISEG) was addressed on a Site Specific basis for certain plants. See Site specific Attachments for additional requirements as follows:

- Attachment A, Brunswick Specific QAPD, Not Addressed.
- Attachment B, Harris Specific QAPD, Section B17.3.4.4, Independent Safety Engineering Group.
- Attachment C, Robinson Specific QAPD, Not Addressed
- Attachment D, Catawba, McGuire, and Oconee Specific QAPD, Section D17.3.4.7, Independent Safety Engineering Group

Attachment A, Brunswick Specific QAPD

Information presented in this attachment is specific to Brunswick and was contained in the UFSAR prior to Amendment 41.

Where a section contains no descriptive information beyond that in the generic text in the body of the document, a statement is made to that effect and no content is included. See A17.3.1.2, Organization for example.

A17. BNP SPECIFIC QUALITY ASSURANCE

A17.1 BNP QA DURING DESIGN AND CONSTRUCTION

See Brunswick UFSAR Chapter 17 for historic information from the description of the QA Program for design and construction.

A17.2 OPERATIONAL QA

Deleted

(NOTE: In April 1995, NRC approved the reformatting of the description of the Brunswick QA Program to follow Standard Revision Plan Section 17.3, replacing the content of 17.2.)

A17.3 BNP QUALITY ASSURANCE PROGRAM (QAP) DESCRIPTION

INTRODUCTION

This content is not addressed in SRP Section 17.3; therefore, the Brunswick description of the QA Program did not include this section.

DEFINITIONS

There are no Brunswick specific definitions.

EXPLANATION OF "QUALITY ASSURANCE"

There is no Brunswick specific content.

QA STANDARDS AND GUIDES

Table A17-1 and A17-2 address QAP conformance to the referenced regulatory and program guidance in NUREG-0800 Section 17.3.

The content of Table A17-1 was transferred from Table 1-6 of the Brunswick UFSAR. Changes to the content of Table A17-1 are controlled in accordance with 10 CFR 50.54(a). Subsequent changes to the QAP are incorporated in this document as identified in Section 17.3.1.7.

Table A17-2 addresses additional Regulatory Guides that relate to implementation of the QAP but the implementation is site specific and controlled with the Brunswick UFSAR in accordance with 10 CFR 50.59.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards

Generic Exception:

Table A17-1 addresses Brunswick's Conformance of the Quality Assurance Program to certain NRC Regulatory Guides. In so doing, specific editions of industry standards are identified for compliance with exceptions and alternatives. Those identified standards include references to other industry standards for activities including, but not limited to; design, fabrication, inspection, and testing. Those included reference industry standards are considered to be guidance documents for details of how activities may be accomplished. The actual standard to be used in such cases is controlled by each station's current licensing and design bases.

Regulatory Guide 1.28, Quality Assurance Program Requirements (Design and Construction) (Safety Guide 28 June 1972) (Rev. 0)
ANSI Standard N45.2-1971, Quality Assurance Requirements for Nuclear Power Plants
This guide, and the standard it endorses, have been superseded for operations activities by Regulatory Guide 1.33 and ANSI N18.7-1976, which it endorses. The Operational Quality Assurance Program complies with Regulatory Guide 1.33 and ANSI N18.7-1976 as stipulated in Appendix A to that Program; therefore, Regulatory Guide 1.28 (Safety Guide 28) and ANSI N45.2-1971, which it endorses, are not considered necessary and are not included as part of the program.

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment (Safety Guide 30, Revision 0, August 1972)
ANSI Standard N45.2.4-1972 (IEEE-336-1971), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations
BNP 1 and 2 complies with the provisions of Regulatory Guide 1.30, August 1972, as indicated below:
The installation, inspection, and testing of nuclear power plant instrumentation and electrical equipment at BNP will be in accordance with the applicable requirements of ANSI N45.2.4-1972 with the following exceptions: 1. Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in Brunswick commitment to Regulatory Guide 1.74. 2. Section 1.5 titled Reference Documents: Brunswick's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. 3. Section 2.5 titled Measuring and Test Equipment: Brunswick will implement the applicable portions of this Section as follows: The status of portable items of measuring and test equipment and reference standards shall be identified by use of status cards, computer schedules, or tags for the date recalibration is due. These items are in a calibration program which requires recalibration on a specified frequency or, in certain cases, prior to use.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment (Safety Guide 30, Revision 0, August 1972)
ANSI Standard N45.2.4-1972 (IEEE-336-1971), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations
BNP 1 and 2 complies with the provisions of Regulatory Guide 1.30, August 1972, as indicated below:

Instrumentation and electrical equipment in the categories listed below shall be in a calibration program. This program provides, by the use of status cards, computer schedules, or tags, for the date that recalibration is due and indicates the status of calibration. The identity of person(s) performing the calibration is provided on the calibration documents.

a. Instruments installed as listed in the BNP Technical Specifications
b. Installed instrumentation used to verify BNP Technical Specification parameters
c. Installed safety-related instruments and electrical equipment that provide an active function during operation or during shutdown; i.e., not a device being designated safety-related solely because the instrument is an integral part of a pressure retaining boundary.

4. Section 7 titled Data Analysis and Evaluation states in part, "Procedures shall be established for processing inspection and test data and their analysis and evaluation." At BNP 1 and 2, (data processing procedures per se have not been developed; instead, test data are recorded, processed, and analyzed in accordance with procedures and instructions in appropriate functional areas; e.g., maintenance, startup.

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation) (Safety Guide 33 November 1972)
ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
BNP 1 and 2 complies with the provisions of Regulatory Guide 1.33, November 1972, and the requirements and recommendations for administrative controls described in ANSI N18.7-1976 except as stated below:

1. The requirements of Paragraph 4.3 Independent Review Program are replaced by Section 17.3.3.2, Independent Review. This exception uses NRC Safety Evaluation dated January 13, 2005 to Nuclear Management Company (ADAMS ML050210276).
2. Deleted - see exception 1.
3. Paragraph 4.5 - Written audit reports are not formally reviewed as part of the Independent Review function.
4. Paragraph 4.5 - The CNO will assure that an independent assessment of the overall Nuclear Oversight Program is conducted at least once every 24 months. Results of the independent assessment will be reported directly to the CNO and entered into the Corrective Action Program for resolution. For scheduling consistency, the exceptions included in paragraph 5 of this section will be used as clarification for scheduling this independent assessment.
5. Paragraph 4.5, Audit Program - ANSI N18.7-1976/ANS-3.2, Section 4.5 is implemented with the following clarification: The audits of selected aspects of operational phase activities as identified in Section 17.3.3.3.3, Internal Audit Program, are scheduled based on plant performance and importance to safety but at a frequency not to exceed twenty-four months with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation) (Safety Guide 33 November 1972)
ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
BNP 1 and 2 complies with the provisions of Regulatory Guide 1.33, November 1972, and the requirements and recommendations for administrative controls described in ANSI N18.7-1976 except as stated below:

6. Section 5.2.2 titled Procedure Adherence: Temporary changes to approved procedures and proposed tests or experiments may be made provided; a) the intent of the original procedure, proposed test or experiment is not altered; b) the change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected; and c) the change is documented and, if appropriate, reviewed and approved for incorporation in the next revision of the procedure within 14 days of implementation of the temporary change.
7. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972, shall be established, implemented, and maintained as specified in the BNP 1 and 2 BNP Technical Specifications.
8. Paragraph 5.2.7 - BNP will comply with requirements of the first sentence of the second paragraph and provides the following clarification:
   a. "Documented Instructions" is defined as any credible information (e.g., vendor manuals, vendor recommendations, engineering direction, etc.) Used for work planning/execution which is reviewed and approved prior to use in accordance with approved procedures.
9. Paragraph 5.2.13, titled Procurement Document Control: When purchasing commercial grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Alternate requirements described in Section 1.8 for Regulatory Guide 1.123 may be implemented in lieu of imposing a quality assurance program consistent with ANSI N45.2-1971.
10. Section 5.2.15 titled Review, Approval and Control of Procedures, states that, "Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary. A revision to a procedure constitutes a procedure review." In lieu of this commitment, Duke Energy addresses programmatic controls in Section 17.3.2.14 to continually identify procedure revisions which may be needed to ensure that procedures are appropriate for the circumstance and are maintained current.
11. Section 5.2.17, second to the last sentence in the last paragraph, "Deviations, their cause, and any…," to be consistent with Paragraph 5.2.11 and 10CFR 50, Appendix B, the cause of the deviation will be determined for only significant conditions adverse to safety.
12. Paragraph 5.3.5(4) last sentence - BNP interprets the review requirements for "Supporting Maintenance Documents" which have not been incorporated in a procedure, be performed in an equivalent manner as described in approved procedures.
13. Section 5.3.9.1, titled Emergency Procedure Format and Content: Emergency procedures shall be in the format as committed to in NUREG-0737, TMI Action Plan.
14. ANSI N18.7-1976, Section 5.2.16. See Section A17.3.2.9 for clarification.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation) (Safety Guide 33 November 1972)

ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants

BNP 1 and 2 complies with the provisions of Regulatory Guide 1.33, November 1972, and the requirements and recommendations for administrative controls described in ANSI N18.7-1976 except as stated below:

15. Section 5.3.10, first paragraph - The requirement "Test and inspection results shall be documented...," will be implemented as follows:
    As an alternative to the records required for inspections outlined in paragraph 5.3.10, BNP shall provide the following as the method to document results of inspections.
    The results of inspections will be documented in appropriate records and those records shall, as a minimum, identify a) through f) below:
    a) Item inspected
    b) Date of inspection
    c) Inspector
    d) Type of observation
    e) Results or acceptability
    f) Reference to information on action taken in connection with non-conformances.

Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 1973)

ANSI Standard N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants

Those areas of the QA Program applicable to onsite cleaning of materials and components, cleanliness control, and pre-operation cleaning and layup of BNP 1 and 2 fluid systems, will be in accordance with ANSI N45.2.1-1973, with the following exceptions:

1. At BNP 1 and 2, a classification system similar to ANSI N45.2.1-1973 has been developed and is fully implemented for cleaning of fluid systems.
2. Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in BNP's commitment to Regulatory Guide 1.74.
3. Section 1.5 titled Referenced Documents: BNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document.

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging Shipping Receiving Storage and Handling of Items for Water-Cooled Nuclear Power Plants (March 1973)

ANSI Standard N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants

Packaging, shipping, receiving, storage, and handling of BNP items are in accordance with applicable requirements of ANSI N45.2.2-1972 with the following specific exceptions:

1. Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in BNP's commitment to Regulatory Guide 1.74.
2. Section 1.5 titled Referenced Documents: BNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging Shipping Receiving Storage and Handling of Items for Water-Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants
Packaging, shipping, receiving, storage, and handling of BNP items are in accordance with applicable requirements of ANSI N45.2.2-1972 with the following specific exceptions:

3. Section 2.7 titled Classification of Items and Section 6.1.2 titled Levels of Storage:
    a. Special electronic equipment and instrumentation received as assembled panels will be stored as recommended by the manufacturer and/or based on engineering evaluation to prevent damage, deterioration, or contamination, but not necessarily in a Level A storage area.
    b. Chemicals used at BNP 1 and 2 are stored at the point of use and/or in warehouse areas that satisfy the requirement of Level B storage. These storage areas have been evaluated and determined to be adequate for the limitations established by the manufacturer.
    c. Special nuclear materials are stored in areas specifically designed for such storage.
4. Paragraph 6.4.2, Care of Items: The following alternates are provided for indicated subparts:
    a. Space heaters in electrical equipment shall be energized unless a documented engineering evaluation determines that such space heaters are not required.
    b. Rotating electrical equipment, commensurate to safety or reliability, shall be given insulation resistance tests on a schedule basis, unless a documented evaluation determines that such tests are not required.
    c. Rotating equipment, commensurate to safety or reliability, shall be evaluated for shaft rotation requirements. The degree of turn shall be established so that the parts receive a coating of lubrication where applicable, and so that the shaft does not come to rest in a previous position. (90 deg. and 450 deg. rotations are examples.)
    d. Other maintenance requirements specified by the manufacturer's instructions shall be evaluated to determine applicability during storage of the item.
5. Section 7.3.4 - BNP intends to comply with the requirements of this Section with the following clarification: Test loads equal to or greater than the original crane rating shall not pass over locations where special nuclear material is stored or where reactor system components or high cost equipment are located.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.39, Housekeeping Requirements for Water-Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.3-1973, Housekeeping, During the Construction Phase of Nuclear Power Plants
The applicable operational phase requirements of N45.2.3-1973 are followed at BNP within the context of the established QA Program with the following specific exception -- the zone designations of Section 2.1 of N45.2.3 and the requirements associated with each zone are considered impractical for implementation, as stated, at BNP during the operations phase. Instead, procedures or instruction for housekeeping activities, which include the applicable requirements outlined in Section 2.1 of N45.2.3, and which take into account radiation control considerations, security considerations, and cleanliness requirements, are developed on a case by case basis for work to be performed.
Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (September 1980)
ANSI Standard N45.2.6-1978, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants"
BNP 1 and 2 complies with NRC Regulatory Guide 1.58, September 1980, which endorses ANSI N45.2.6-1978, with the following exceptions: 1. Section 1.2 titled Applicability: BNP elects not to apply the requirements of this guide to those personnel who are involved in the daily operations of surveillance, maintenance, and certain technical and support services whose qualifications are controlled by the BNP Technical Specifications or are controlled by other QA Program commitment requirements. Only personnel in the following listed categories will be required to meet ANSI N45.2.6-1978 requirements: a. Nondestructive examination (NDE) personnel b. QC inspection personnel c. Receipt Inspection personnel 2. The fourth paragraph of Section 1.2 requires that the Standard be imposed on personnel other than BNP employees. The applicability of the Standard to suppliers and contractors will be documented and applied, as appropriate, in the procurement documents for such suppliers and contractors or in interface agreements for Duke Energy non-nuclear organizations providing services identified in Section 17.3.1.2.3. 3. Section 1.4 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in BNP's commitment to Regulatory Guide 1.74. 4. Section 2.5 titled Physical: BNP will implement the requirements of this Section with the stipulation that, where no special physical characteristics are required, none will be specified. The converse is also true: if no special physical requirements are stipulated by BNP, none are considered necessary. BNP employees receive an initial physical examination to assure satisfactory physical condition; however, only the following listed personnel will receive an annual examination: a. NDE personnel b. QC inspection personnel c. Receipt inspection personnel This annual examination shall consist of the near visual acuity using the standard Jaeger's type chart or equivalent test.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (September 1980)
ANSI Standard N45.2.6-1978, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants"
BNP 1 and 2 complies with NRC Regulatory Guide 1.58, September 1980, which endorses ANSI N45.2.6-1978, with the following exceptions: 5. Section 3 titled Qualifications: Only personnel performing NDE (such as LP, MT, UT, and RT) are required to be grouped in levels of capability and certified as such. QC inspection personnel will be certified for inspection, review, and evaluation of inspection data, and reporting of inspection and test results. 6. Section 3.5 titled Education & Experience Recommendations: BNP will certify individual inspectors through training and experience to requirements appropriate to the specific assignment; however, except for NDE, personnel are not required to be classified by levels of capability. Inspection personnel may be qualified based on pre-established experience, education, on-the-job training, written examinations and proficiency tests associated with the specific activity. Proficiency tests are given to personnel performing independent QC inspections and documented acceptance criteria are developed to determine if individuals are properly trained and qualified. Certificates of qualification delineate the functions personnel are qualified to perform. Qualification records are maintained and performance evaluations conducted at least once every three years. If organizations elect to utilize qualifications by levels for non-NDE inspections, Level I inspectors receive a minimum of 4 months experience as Level I before being certified as Level II, in lieu of one year experience recommended by ANSI N45.2.6 Section 3.5.2(1). Organizations identify in their procedures if they qualify their inspectors by Level or by task qualifications. Inspectors are only assigned functions for which they have been qualified.
Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants (October 1973)
ANSI Standard N45.2.1.1-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants
Those areas of the QA Program for BNP 1 and 2 applicable to design or modification of the plant are in accordance with the applicable guidance of ANSI N45.2.11-1974, with the following exception: 1. Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in the BNP commitment to Regulatory Guide 1.74.
Regulatory Guide 1.74, Quality Assurance Terms and Definitions (February 1974)
ANSI Standard N45.2.1.0-1973, Quality Assurance Terms and Definitions
Comply with the provisions of Regulatory Guide 1.74, February, 1974.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (August 1974)
ANSI Standard N45.2.9-1974, Collection, Storage, and Maintenance of QA Records
The requirements for collection, storage, and maintenance of QA records at BNP will be in accordance with ANSI N45.2.9-1974 and 17.3.2.15, with the following specific exceptions:
See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records. 1. The document control facility at the BNP shall comply with the requirement of Regulatory Guide 1.88, October, 1976, Regulatory Position C.2 in that the facility has been specifically designed to protect the contents from fire in accordance with NFPA 232-1975, with the following exceptions/alternatives/comments: a. Records are classified as Class 1 - Vital Records in accordance with NFPA 232-1975, Chapter 5, Section 5222; however, the records that meet this classification include those determined to be QA records as defined in ANSI N45.2.9-1974, paragraph 1.4. b. The facility is constructed in accordance with NFPA 232-1975 requirements for a fire-resistive file room as defined in NFPA 232-1975, Chapter 3. The walls were designed and constructed equivalent to a four-hour barrier. The doors are four-hour rated vault doors. Penetrations for electrical service and ventilation are sealed to a rating of 3 hours to protect the vault from a fire originating outside the vault. c. Due to the construction of the facility and other safety measures described herein, the statement in NFPA 232-1975, Chapter 3, Section 3022(d), "Class 1 . . . records should not be subjected to these possibilities of destruction by fire" is deemed to be inappropriate. d. The facility is protected by a Halon fire extinguishing system, automatic door closures, and fire detection system. e. The floor of the file room is six inches higher than the floor areas outside the file room. f. The walls are reinforced concrete, ten inches thick. g. The exterior walls are totally enclosed and insulated from the outside environment and elements. h. The facility is constructed independently from the building. i. NFPA 232-1975, Chapter 3, Sections 332 and 333 describe methods for heating and ventilation. The facility will have penetrations in the wall for the purposes of heating and ventilation. The facility is equipped with a Heating, Ventilating and Air Conditioning system external to the file room with automatic closing dampers. "The temperature and humidity should be controlled between 65 and 75 degrees and 20 and 40 percent, respectively. Temporary operation outside this range is acceptable during extreme low outside relative humidity conditions which have shown to drop the vault humidity below 20 percent. Humidity above the upper 40 percent range is acceptable during maintenance. The above times are short in duration and the humidity change is gradual. Evaluation for out of tolerance conditions will be performed. Corrective actions or compensatory measures will be taken if required, to ensure there is no adverse impact on storage of records and to restore the vault to prescribed conditions." j. 120 VAC wall outlets are provided in the file room for emergency lighting and janitorial needs. These outlets may be de-energized from a disconnect box installed on the outer wall of the records storage facility. The lighting may be disconnected outside the room and is equipped with a red pilot light. k. BNP QA records not stored in the facility described above may be retained at off-site locations which meet the requirements (with approved exceptions as necessary) of Section 5.6, ANSI 45.2.9-1974.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.88, Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (August 1974)
ANSI Standard N45.2.9-1974, Collection, Storage, and Maintenance of QA Records
The requirements for collection, storage, and maintenance of QA records at BNP will be in accordance with ANSI N45.2.9-1974 and 17.3.2.15, with the following specific exceptions:

2. Paragraph 1.4, Definitions: The phrase "when the document has been completed" is clarified to mean when the document has received the final review performed by the organizational element responsible for generating or collecting the records. In the case of a record package made up of several individual documents, the package will be considered to be the document for the purpose of determining when the record is complete.
3. Paragraph 3.2.1, Generation of Quality Assurance Records: The phrase "completely filled out" is clarified to mean that sufficient information is recorded to fulfill the intended purpose of the record.
4. Paragraph 4.2, Timeliness: BNP's contractual agreement with its contractors and suppliers will constitute fulfillment of the requirements of this paragraph.
5. Paragraph 5.4, Preservation: The following clarification is substituted for the current subparagraph 5.4.2: "Records shall not be stored loosely. They shall be secured for storage in file cabinets or on shelving in containers." The following clarification is substituted for the current subparagraph 5.4.3: "Appropriate provisions shall be made for special processed records (such as radiographs, photographs, negatives, microfilm and magnetic media) to prevent or minimize damage for excessive light, stacking, electromagnetic fields, temperature and humidity, etc. Manufacturer's recommendations will be considered as appropriate."
6. Paragraph 5.6, Facility: This paragraph provides no distinction between temporary and permanent facilities. To cover temporary storage, the following clarification is added: "Complete records may be stored in one-hour fire rated file cabinets until transmitted for permanent storage. In general, records shall not be maintained in temporary storage by the generating organization for more than 90 days after completion. Any exceptions to this requirement must be justified, evaluated and approved by the Document Management Supervisor and documented. A list of exceptions shall be maintained and available for NRC review. Exceptions may include records needed on a continuing basis for an extended period of time at the location of the work group responsible for generating the records and records which are cumulative in nature and could best be turned over for storage for a designated period of time.
   In addition, Document Management & Information Services will store records in one- hour rated file cabinets while the records are being processed for permanent storage.
7. See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)
ANSI Standard N45.2.5-1974, Supplementary Quality Assurance Requirements for Installation Inspections and Testing of Structural Steel During the Contract Phase of Nuclear Power Plants
Regulatory Guide 1.94, Revision 1, April 1976 endorses ANSI N45.2.5-1974. BNP 1 and 2 do not commit to Regulatory Guide 1.94 but do endorse parts of ANSI N45.2.5-1974 as described below.
The original specification requirements, applicable guidance contained in ANSI N45.2.5-1974, or acceptable alternatives based on an engineering evaluation will be utilized in the event future structural work is to be performed which falls under the established requirements of the BNP QA Program.
Regulatory Guide 1.116, QA Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (June 1976)
ANSI Standard N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants
Regulatory Guide 1.116, June 1976, endorses ANSI N45.2.8-1975. BNP 1 and 2 does not commit to Regulatory Guide 1.116 but does endorse parts of ANSI N45.2.8-1975 as described below.
Within the context of the established QA Program, the applicable guidance contained in ANSI N45.2.8-1975 will be utilized in relation to mechanical maintenance or modification with the following exceptions: 1. Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in BNP's commitment to Regulatory Guide 1.74. 2. Section 1.5 titled Referenced Documents: BNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. 3. Section 2.8 titled Measuring and Test Equipment: BNP will implement the applicable portions of this Section as follows: a. The status of portable items of measuring and test equipment and reference standards shall be identified by use of status cards, computer schedules, or tags for the date recalibration is due. These items are in a calibration program which requires recalibration on a specified frequency or, in certain cases, prior to use. b. Instrumentation and electrical equipment in the categories listed below shall be in a calibration program. This program provides, by the use of status cards, computer schedules, or tags, for the date that recalibration is due and indicates the status of calibration. The identity of person(s) performing the calibration is provided on the calibration documents. 1) Instruments installed as listed in the BNP Technical Specifications 2) Installed instrumentation used to verify BNP Technical Specification parameters

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.116, QA Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (June 1976)
ANSI Standard N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants
Regulatory Guide 1.116, June 1976, endorses ANSI N45.2.8-1975. BNP 1 and 2 does not commit to Regulatory Guide 1.116 but does endorse parts of ANSI N45.2.8-1975 as described below.

3) Installed safety-related instruments and electrical equipment that provide an active function during operation or during shutdown; i.e., instead of being designated safety-related solely because the instrument is an integral part of a pressure retaining boundary,

4. Section 6 titled Data Analysis and Evaluation states in part, "Procedures shall be established for processing inspection and test data and their analysis and evaluation."

   At BNP 1 and 2, data processing procedures per se have not been developed; instead, test data are recorded, processed, and analyzed in accordance with procedures and instructions in appropriate functional areas; e.g., maintenance, startup.

Regulatory Guide 1.123, "Quality Assurance Requirement for Control or Procurement of Items and Services for Nuclear Power Plants"
ANSI Standard N45.2.13, "Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants" (Draft 2, Rev. 4, April 1974)
BNP does not commit to Regulatory Guide 1.123; however, the applicable guidance contained in ANSI N45.2.13 (Draft 2, Revision 4, April 1974) and ANSI N18.7-1976, will be utilized in relation to procurement of items and services performed under the established requirements of the QA Program.
See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and Services including, purchasing commercial-grade calibration services from calibration laboratories.

Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants (January 1979)
ANSI Standard N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
BNP will follow the requirements and recommendations of Regulatory Guide 1.144 and ANSI Standard N45,2.12, with the following clarifications:

1. BNP will follow the requirements and recommendations of Regulatory Guide 1.144, paragraphs C.1, C.2, C.3.a.2, C.3.b, and C.4. BNP's position on paragraph C.3.a.1 is as follows:

   Audits of operational phase activities, as outlined in Section 17.3.3.3 shall be performed at the frequencies stated in exception 5 for RG 1.33 in Table A17-1.

   See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants (January 1979)
ANSI Standard N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
BNP will follow the requirements and recommendations of Regulatory Guide 1.144 and ANSI Standard N45,2.12, with the following clarifications:

2. (Deleted)
3. BNP will comply with the last paragraph of Section 4.4 of ANSI N45.2.12 concerning issuing audit reports, with the following clarification: "Audit reports shall be issued within thirty working days after the last day of the audit. The last day of the audit shall be considered to be the day of the post-audit conference. If a post-audit conference is not held because it was deemed unnecessary, the last day of the audit shall be considered to be the date the post-audit conference was deemed unnecessary as documented in the audit report."
4. ANSI N45.2.12 Paragraph 4.3.1, Preaudit Conference: BNP will comply with the requirement of this paragraph by inserting the word "Normally" at the beginning of the first sentence. This clarification is required because, in the case of certain unannounced audits or audits of a particular operation or work activity, a preaudit conference might interfere with the spontaneity of the operation or activity being audited. In other cases, persons who should be present at a preaudit conference may not always be available. Such lack of availability should not be an impediment to beginning an audit. Even in the above examples, which are not intended to be all inclusive, the material set forth in paragraph 4.3.1 will normally be covered during the course of the audit.
5. ANSI N45.2.12 Paragraph 4.3.3, Post Audit Conference: BNP will substitute and comply with the following paragraphs:
   For all external audits, a post audit conference shall be held with management of the audited organization to present audit findings and clarify misunderstandings. Where no adverse findings exist, this conference may be waived by management of the audited organization. Such waiver shall be documented in the audit report. For all internal audits, unless unusual operating or maintenance conditions preclude attendance by appropriate managers/supervisors, an audit exit shall be held with managers/supervisors. If there are no adverse findings, management of the audited organization may waive the audit exit. Such waiver shall be documented in the audit report.
6. ANSI N45.2.12 Paragraph 4.4, Reporting:
   a. This paragraph requires that the audit report be signed by the audit team leader which is not always the most expeditious route for the audit report to be issued as soon as possible. BNP will comply with Paragraph 4.4 as clarified to read:
      An audit report shall be signed by the audit team leader or the leader's supervisor in the absence of the audit team leader. In cases where the audit report is not signed by the audit team leader due to the leader's absence, the record copy of the report must be signed by the audit team leader upon return. The report shall not require the audit team leader's review/concurrence/signature if the audit team leader is no longer employed by BNP at the time audit report is issued. The audit report shall provide:
   b. BNP will comply with subparagraph 4.4.3 clarified to read: "Supervisory level personnel with whom significant discussions were held during the course of preaudit (where conducted), audit, and post audit (where conducted) activities.
   c. Subparagraph 4.4.6 requires audit reports to include recommendations for corrective actions. BNP may choose not to comply with this requirement. Instead, BNP audit reports are required to document findings.

Table A17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.146, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Rev. 0 August 1980)
ANSI Standard N45.2.23-1978, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
BNP 1 and 2 complies with NRC Regulatory Guide 1.146, Revision 0, which endorses ANSI N45.2.23-1978, with the following exceptions:

1. Section 1.4 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 will be used; "Audit" which is included in ANSI N45.2.10 will be used as clarified in BNP's commitment to Regulatory Guide 1.74.
2. Section 2.2 titled Qualification of Auditors: Subsection 2.2.1 references an ANSI B45.2 which will be assumed to be N45.2. BNP will comply with an alternate subsection 2.2.1 which reads:
   Orientation to provide a working knowledge and understanding of the BNP Quality Assurance Program, including the Regulatory Guides and ANSI standards included in the Program, and BNP procedures for performing audits and reporting results.
3. (Deleted)
4. Section 4.1 titled Organizational Responsibility: BNP will comply with this Section with the substitution of the following sentence in place of the last sentence in the Section:

   Management or the Audit/NOS Team Leader shall, prior to commencing the audit, assign personnel who collectively have experience or training commensurate with the scope, complexity, or special nature of the activities to be audited.
5. Section 5.3 titled Updating of Lead Auditors' Records: BNP will substitute the following sentence for this Section:

   Records for each Lead Auditor shall be maintained and updated during the annual management assessment as defined in Section 3.2 (as clarified).
6. Section 5.4 titled Record Retention: BNP will substitute the following sentence for this Section:
   Qualification records shall be retained as required by the BNP Quality Assurance Program.
7. ANSI N45.2.23-1978, Section 2.3.4 titled Audit Participation: BNP will substitute the following for this Section:

   Prospective Lead Auditors shall demonstrate the ability to effectively implement the audit process and effectively lead an audit team. This process is described in written procedures which provide for evaluation and documentation of the results of this demonstration. In addition, the prospective Lead Auditor shall have participated in at least two Nuclear Oversight audits within the year preceding the individual's effective date of qualification. Upon successful demonstration of the ability to effectively implement the audit process and effectively lead audits, and having met the other provisions of Section 2.3 of ANSI/ASME N45.2.23-1978, the individual may be certified as being qualified to lead audits.

Table A17-2. Site Specific Response to Regulatory Guides and Industry Standards

Table A17-2 identifies additional Regulatory Guides addressing subjects related to implementation of the QAP but the implementation is site specific and controlled with the UFSAR in accordance with 10 CFR 50.59.

Regulatory Guide 1.8, Personnel Selection and Training
Personnel selection and training is site specific. Brunswick addresses conformance with Regulatory Guide 1.8 (SAFETY GUIDE 8, MARCH 1971) in UFSAR Chapter 1 Table 1-6.

Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
Quality group classifications and standards trace to the original design and construction of the nuclear power plant and therefore are site specific. Brunswick does not address Regulatory Guide 1.26 in UFSAR Chapter 1 Table 1-6. Quality group classifications are addressed in UFSAR Chapter 3.

Regulatory Guide 1.29, Seismic Design Classification
Seismic design classification trace to the original design and construction of the nuclear power plant and therefore is site specific. Brunswick addresses conformance with Regulatory Guide 1.29 in UFSAR Chapter 1 Table 1-6.

Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel
Nonmetallic thermal insulation for austenitic stainless steel trace to the original design and construction of the nuclear power plant and therefore is site specific. Brunswick does not address conformance with Regulatory Guide 1.36 in UFSAR Chapter 1 Table 1-6. Thermal insulation for austenitic stainless steel is addressed in UFSAR Section 5.2.

Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
Quality assurance requirements for protective coatings applied to water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Brunswick addresses conformance with Regulatory Guide 1.54 in UFSAR Chapter 1 Table 1-6.

Table A17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Brunswick does not address conformance with Regulatory Guide 1.143 in UFSAR Chapter 1 Table 1-6. Design guidance for radioactive waste management systems, structures, and components is addressed in UFSAR Chapter 11.

Regulatory Guide 1.155, Station Blackout
Addressing Station Blackout is site specific. Brunswick addresses conformance with Regulatory Guide 1.155 in UFSAR Chapter 1 Table 1-6.

Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment
Quality assurance for radiological monitoring program (normal operations) – effluent streams and the environment is site specific. Brunswick does not address conformance to Regulatory Guide 4.15 in UFSAR Chapter 1 Table 1-6. The radiological monitoring program is addressed in UFSAR Chapter 11.

A17.3.1 MANAGEMENT

A17.3.1.1 Methodology

There are no Brunswick specific amplifications for this section.

A17.3.1.2 Organization

There are no Brunswick specific amplifications for this section.

A17.3.1.3 Responsibility

There are no Brunswick specific amplifications for this section.

A17.3.1.4 Authority

The program and procedures require that the authority and duties of persons and organizations performing activities affecting quality functions be clearly established and delineated in writing and that these individuals and organizations have sufficient authority and organizational freedom to:

1. Identify quality, nuclear safety, and performance problems.
2. Order unsatisfactory work to be stopped and control further processing, delivery, or installation of nonconforming material.
3. Initiate, recommend, or provide solutions for conditions adverse to quality.
4. Verify implementation of solutions.

A17.3.1.5 Personnel Training and Qualification

There are no Brunswick specific amplifications for this section.

A17.3.1.6 Corrective Action

The program requires that an evaluation of adverse conditions such as conditions adverse to quality, nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment is conducted to determine need for corrective action.

Conditions adverse to quality are identified through inspections, assessments, tests, checks, and review of documents.

The program requires corrective action to be initiated to preclude recurrence of significant conditions adverse to quality.

Procedures require follow-up reviews, verifications, inspections, etc., to be conducted to verify proper implementation of corrective action and to close out the corrective action documentation.

The program outlines the methodology for resolution of disputes involving quality and nuclear safety issues arising from a difference of opinion between identifying personnel and other groups.

Significant conditions adverse to quality are reported to appropriate management for review and evaluation.

Periodic review and evaluation of adverse trends are performed by management.

A17.3.1.7 Regulatory Commitments

Written procedures shall be established, implemented, and maintained to ensure implementation of the Process Control Program.

A17.3.2 PERFORMANCE/VERIFICATION

A17.3.2.1 Methodology

There are no Brunswick specific amplifications for this section.

A17.3.2.2 Design Control

There are no Brunswick specific amplifications for this section.

A17.3.2.3 Design Verification

There are no Brunswick specific amplifications for this section.

A17.3.2.4 Procurement Control

Potential contractors and suppliers are evaluated prior to award of a procurement contract when needed to assure the contractor's or supplier's capability to comply with applicable technical and quality requirements.

Procurement documents, such as purchase specifications, contain or reference the following:

1. Technical, administrative, regulatory, and reporting requirements, including material and component identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions.
2. Identification of the documentation to be prepared, maintained, or submitted (as applicable) to BNP for review and approval. These documents may include, as necessary, inspection and test records, qualification records, or code required documentation.
3. Identification of those records to be retained, controlled, and maintained by the supplier, and those delivered to the purchaser prior to use or installation of the hardware.

Procurement documents require suppliers to operate in accordance with QA programs which are compatible with the applicable requirements of the QA Program and procedures where their services are utilized in support of plant activities.

A17.3.2.5 Procurement Verification

There are no Brunswick specific amplifications for this section.

A17.3.2.6 Identification and Control of Items

Procedures require that materials, parts, and components be identified and controlled to prevent the use of incorrect or defective items. These procedures also require that identification of items be maintained either on the item in a manner that does not affect the function or quality of the item, or on records traceable to the item.

Procedures implementing these requirements provide for the following:

1. Verification that items received at the plant are properly identified and can be traced to the appropriate documentation, such as drawings, specifications, purchase orders, manufacturing and inspection documents, nonconformance reports, or material test reports.
2. Verification of item identification consistent with the BNP inventory control system and traceable to documentation which identifies the proper uses or applications of the item.

A17.3.2.7 Handling, Storage, and Shipping

Provisions are established to control the shelf life and storage of chemicals, reagents, lubricants, and other consumable materials.

A17.3.2.8 Test Control

Test procedures incorporate or reference the following, as required:

1. Instructions and prerequisites for performing the test,
2. Use of proper test equipment,
3. Mandatory inspection hold points,
4. Acceptance criteria

Test results are documented, evaluated, and their acceptability determined by a qualified, responsible individual or group.

When the acceptance criteria are not met, affected areas are to be retested or evaluated, as appropriate.

A17.3.2.9 Measuring and Test Equipment Control

Portable measuring and test equipment are calibrated by standards at least four times as accurate as the portable measuring and test equipment, unless limited by the state of the art.

Special tools such as torque wrenches, calipers, and micrometers are calibrated to be at least as accurate as the application(s) for which it is used, using standards which are at least as accurate as the special tool being calibrated.

Installed measuring and test instruments are calibrated by instruments at least as accurate as the installed, unless limited by the state of the art.

Reference and transfer standards are traceable to nationally recognized standards; or where national standards do not exist, provisions are established to document the basis for the calibration.

A17.3.2.10 Inspection Test and Operating Status

These procedures include the application, removal, and verification of inspection and welding stamps, or other status indicators as appropriate.

Altering the sequence of required tests, inspections, and safety-related operations can only be accomplished by methods outlined in procedures.

A17.3.2.11 Special Process Control

There are no Brunswick specific amplifications for this section.

A17.3.2.12 Inspection

There are no Brunswick specific amplifications for this section.

A17.3.2.13 Corrective Action

The primary goal of the BNP corrective action program is to improve overall plant operations and performance by identifying and correcting root causes of equipment and human performance problems.

Procedures define requirements for a corrective action program that charges personnel working at or supporting the nuclear plants with the responsibility to identify adverse conditions (including conditions adverse to quality).

Procedures include requirements for verification of the acceptability of the rework/repair of items by re-inspection and/or testing in accordance with the original inspection or test requirements or by an accepted alternative inspection and testing method.

Conditions that require rework/repairs are identified through the use of maintenance work request forms.

A17.3.2.14 Control of Documents

Changes to documents are reviewed and approved by the same organization that performed the original review and approval or by other designated qualified responsible organizations.

A17.3.2.15 Records

The structure in which single copy records are maintained is designed to prevent destruction, deterioration or theft. This structure ensures protection against destruction by fire, flooding, theft and deterioration by the environmental conditions of temperature and humidity.

A17.3.2.16 Record Retention

A list of typical operational phase QA Records is included in 17.3.2.15.

A17.3.3 ASSESSMENT

A17.3.3.1 Methodology

There are no Brunswick specific amplifications for this section.

A17.3.3.2 Independent Review

There are no Brunswick specific amplifications for this section.

A17.3.3.3 Independent Assessment

There are no Brunswick specific amplifications for this section.

A17.3.3.3.1 Organization

There are no Brunswick specific amplifications for this section.

A17.3.3.3.2 Internal Assessment Process

There are no Brunswick specific amplifications for this section.

A17.3.3.3.3 Internal Audit Program

A17.3.3.3.3.1 Other Reviews Prescribed by the Code of Federal Regulations

There are no Brunswick specific amplifications for this section.

A17.3.3.3.3.2 Independent Audit of Fire Protection Program

There are no Brunswick specific amplifications for this section.

A17.3.3.3.4 Results

There are no Brunswick specific amplifications for this section.

A17.3.3.3.5 Supplier Oversight

There are no Brunswick specific amplifications for this section.

A17.3.3.3.6 Independent Audit of QA Functions

There are no Brunswick specific amplifications for this section.

A17.3.3.3.7 Audit Frequency Extensions

There are no Brunswick specific amplifications for this section.

A17.3.4 REVIEW AND AUDIT

A17.3.4.1 Procedures, Tests, and Experiments

1. The procedures established, implemented, and maintained for the Quality Assurance Program for effluent and environmental monitoring use the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.
2. See Section 17.3.2.14 for required reviews for changes to procedures, tests, and experiments.

A17.3.4.2 Modifications

There are no Brunswick specific amplifications for this section. See Section 17.3.2.2 for reviews required for modifications.

A17.3.4.3 Operating License/BNP Technical Specifications

1. Operating License/BNP Technical Specification changes shall be processed in accordance with 10CFR 50.90.
2. Operating License/BNP Technical Specification change requests shall be reviewed by the On-Site Review Committee in accordance with Section 17.3.3.2.
3. Changes to the 61BTH Independent Spent Fuel Storage Installation (ISFSI) BNP Technical Specifications and License are processed by Transnuclear, Inc., and will only be reviewed by the On-Site Review Committee if a plant-specific safety issue is identified.

A17.3.4.4 10CFR 50.59 Evaluations and Independent Review Control

There are no Brunswick specific amplifications for this section. See Section 17.3.4.2, 10 CFR 50.59 Reviews.

A17.3.4.5 Nuclear Reviewers

There are no Brunswick specific amplifications for this section. Technical reviewer qualifications are addressed in Section 17.3.4.1, Technical Reviews and 10 CFR 50.59 evaluator qualifications are addressed in Section 17.3.4.2, 10 CFR 50.59 Reviews.

A17.3.4.6 Plant Nuclear Safety Committee

This content is addressed in Section 17.3.3.2, Independent Review.

Attachment B, Harris Specific QAPD

Information presented in this attachment is specific to Harris and was contained in the UFSAR prior to Amendment 41.

Where a section contains no descriptive information beyond that in the generic text in the body of the document, a statement is made to that effect and no content is included. See B17.3.1.2, Organization for example.

B17. QUALITY ASSURANCE

B17.1 QA DURING DESIGN AND CONSTRUCTION

See Harris UFSAR Chapter 17 for historic information from the description of the QA Program for design and construction.

B17.2 OPERATIONAL QA

Deleted

(NOTE: In April 1995, NRC approved the reformatting of the description of the Harris QA Program to follow Standard Revision Plan Section 17.3, replacing the content of 17.2.)

B17.3 HNP QUALITY ASSURANCE PROGRAM (QAP) DESCRIPTION

INTRODUCTION

This content is not addressed in SRP Section 17.3; therefore, the Harris description of the QA Program did not include this section.

DEFINITIONS

Harris specific definitions are found in Table B17.1 addressing conformance with Regulatory Guide 1.74, Quality Assurance Terms and Definitions.

EXPLANATION OF "QUALITY ASSURANCE"

There is no Harris specific content.

QA STANDARDS AND GUIDES

Table B17-1 and B17-2 address QAP conformance to the referenced regulatory and program guidance in NUREG-0800 Section 17.3.

The content of Table B17-1 was transferred from Section 1.8 of the Harris UFSAR. Changes to the content of Table B17-1 are controlled in accordance with 10 CFR 50.54(a). Subsequent changes to the QAP are incorporated in this document as identified in Section 17.3.1.7.

Table B17-2 addresses additional Regulatory Guides that relate to implementation of the QAP but the implementation is site specific and controlled with the Harris UFSAR in accordance with 10 CFR 50.59.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards

Generic Exception:

Table B17-1 addresses Harris's Conformance of the Quality Assurance Program to certain NRC Regulatory Guides. In so doing, specific editions of industry standards are identified for compliance with exceptions and alternatives. Those identified standards include references to other industry standards for activities including, but not limited to; design, fabrication, inspection, and testing. Those included reference industry standards are considered to be guidance documents for details of how activities may be accomplished. The actual standard to be used in such cases is controlled by each station's current licensing and design bases.

Regulatory Guide 1.28, Quality Assurance Program Requirements (Design and Construction) (Rev 0)
ANSI N45.2-1971, Quality Assurance Program Requirements for Nuclear Power Plants
For those activities performed under operating license, HNP shall comply with the requirements of Regulatory Guide 1.33 as specified in Harris Nuclear Plant's (HNP)'s position on Regulatory Guide 1.33. Regulatory Guide 1.28 is not considered necessary and is not included as part of the operational QA program.
Regulatory Guide 1.30, Quality Assurance Requirements for the Installation and Testing of Instrumentation and Electric Equipment (Rev. 0)
HNP complies with the requirements of ANSI N45.2.4-1972), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations, as it is endorsed by Regulatory Guide 1.30 with the following clarifications:
1. Paragraph 2.1, planning: requirements, as determined by responsible plant management, will be incorporated into procedures. 2. Paragraphs 2.2 and 2.3; prerequisites, procedures, and instructions: these controls will be implemented as determined by responsible plant management in approved procedures. 3. Paragraph 2.4, results, will be implemented as set forth in 17.3.2.12 and by compliance with Regulatory Guide 1.33. 4. Paragraph 2.5, measuring and test equipment, will be implemented as set forth in 17.3.2.9 in lieu of the requirements set forth in this paragraph. 5. Paragraph 3, preconstruction verification: "approved instructions" are interpreted to include vendor manuals. 6. Paragraph 4, installation, will be implemented by inclusion of requirements in modification or maintenance procedures, where such procedures are used. Standard HNP practices require that appropriate care be exercised whether a procedure is required or not. 7. Paragraph 5.1, inspections, including subparagraphs 5.1.1, 5.1.2, and the first sentence in 5.1.3, will be implemented as set forth in 17.3.2.12. The remaining sentence in 5.1.3 is covered in equivalent detail by HNP's commitment to Regulatory Guide 1.33, paragraph 5.2.6; the requirements as set forth in that commitment will be implemented in lieu of the requirements stated here. 8. Paragraph 5.2, tests, including subparagraphs 5.2.1 through 5.2.3, will be implemented as set forth in 17.3.2.8. The test program will consider the elements outlined in this paragraph when developing test requirements for inclusion in maintenance and modification procedures. In some cases, testing requirements may be met by post-installation surveillance testing in lieu of a special post-installation test.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation and Testing of Instrumentation and Electric Equipment (Rev. 0)

HNP complies with the requirements of ANSI N45.2.4-1972), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations, as it is endorsed by Regulatory Guide 1.30 with the following clarifications:

9. Paragraph 6, post-construction verification, is not generally considered applicable at operating facilities because of the scope of the work and the relatively short interval between installation and operation.
10. Paragraph 6.2.1 titled equipment tests: the last paragraph of this section deals with tagging and labeling. HNP will comply with an alternate last paragraph which reads: "Each safety-related component of process instrumentation is identified with a unique number. This number is utilized in instrument maintenance records so that current calibration status, including data such as the date of the calibration and identity of person that performed the calibration, can be readily determined. Such information may also be contained on tags or labels which may be attached to installed instrumentation."
11. Paragraph 7, data analysis and evaluation, will be implemented as stated with adding the clarifying phrase "when used" at the beginning of that paragraph. The plant shall have procedures, to the extent determined by responsible plant management, for the performance of analyzing test data, but these procedures are not referred to as data processing procedures.

Regulatory Guide 1.33, Quality Assurance Program Requirements (Rev. 2) (Operation)

HNP complies with this guide, which endorses ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, with the following clarifications:

1. Paragraph 1,"Scope", recommends that this standard applies to activities other than those associated with safety related equipment, activities, and procedures. ANSI N18.7-1976 has not fully taken into account the requirements of regulations other than 10CFR 50. Conflicts may exist between ANSI N18.7-1976 and those other regulations, such as OSHA, 10CFR 19, 20, 21, 30, 40, 70, 71, 73, and ASME. Therefore, HNP shall apply ANSI N18.7-1976 only to those plant features addressed in Section 3.2 of the HNP UFSAR that are classified as safety-related and under the control of the QA program.
2. Written audit reports are not formally reviewed as part of the independent review function.
3. The CNO will assure that an independent assessment of the overall nuclear oversight program is conducted at least once every 24 months. Results of the independent assessment will be reported directly to the CNO and entered into the corrective action program for resolution. (for scheduling consistency, the exceptions included in paragraph 18 of this section will be used as clarification for scheduling this independent assessment).
4. Paragraph 5.2.6, Equipment Control: HNP will comply with the "independent verification" requirements based on the definition of this phrase as given under the commitment to Regulatory Guide 1.74.
   Since HNP sometimes uses descriptive names to designate equipment, the sixth paragraph, second sentence is replaced with: "Suitable means include identification numbers or other descriptions which are traceable to records of the status of inspections and tests.
   The first sentence in the seventh paragraph will be complied with after clarifying "operating personnel" to mean trained employees assigned to, or under the control of, Duke Energy management at an operating nuclear facility.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Rev. 2) (Operation)
HNP complies with this guide, which endorses ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, with the following clarifications:

5. Paragraph 5.2.7, Maintenance and Modification: since some emergency situations could arise which preclude preplanning of all activities, HNP will comply with an alternate to the first sentence in the second paragraph which reads:
   "Except in emergency or abnormal operating conditions where immediate actions are required to protect the health and safety of the public, to protect equipment or personnel, or to prevent the deterioration of plant conditions to a possible unsafe or unstable level, maintenance or modification of equipment shall be preplanned and performed in accordance with written procedures. Where written procedures would be required and are not used, the activities that were accomplished shall be documented after the fact and receive the same degree of review as if they had been preplanned." where procedures are not available, documented instructions may be used to perform maintenance and modification activities. "Documented instructions" are defined as any credible information (e.g., vendor manuals, vendor recommendations, engineering direction etc.) used during work planning/execution which is reviewed and approved prior to use in accordance with approved procedures.
   Paragraph 5.2.7.1, Maintenance Programs: HNP will comply with the requirements of the first sentence of the fifth paragraph. This clarification is needed since it is not always possible to promptly determine the cause of the malfunction. HNP will initiate proceedings to determine the cause, and will make such determination promptly where practical. Determination of the term "promptly" and the term "practical" will be the responsibility of plant management and shall be based on the effect of the condition on the immediate health and safety of the public.
6. Paragraph 5.2.8, Surveillance Testing and Inspection Schedule: In lieu of a "master surveillance schedule," the following requirement shall be complied with: "surveillance testing schedule(s) shall be established reflecting the status of all planned in-plant surveillance tests and inspections."
7. Paragraph 5.2.9, Plant Security and Visitor Control, requires certain procedures and controls. In order to ensure that a conflict between 10CFR 73 and Regulatory Guide 1.17 and ANSI N18.17 does not exist, HNP shall not follow paragraph 5.2.9. An NRC approved security plan shall be implemented prior to fuel loading.
8. Paragraph 5.2.11, Corrective Action, requires certain activities to be performed. In order to avoid conflict between requirements, HNP shall follow the requirements in Sections 17.3.1.6 and 17.3.2.13, in lieu of paragraph 5.2.11.
9. Paragraph 5.2.13.1, Procurement Document Control: When purchasing commercial-grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Alternate requirements described in this table for Regulatory Guide 1.123 may be implemented in lieu of imposing a quality assurance program consistent with ANSI N45.2-1971.
10. Paragraph 5.2.15, Review, Approval and Control of Procedures: The third sentence in paragraph three is interpreted to mean: "Applicable procedures shall be reviewed following an accident, an unexpected transient or a significant operator error. Applicable procedures shall also be reviewed following an equipment malfunction which results in a reportable event."

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Rev. 2) (Operation)
HNP complies with this guide, which endorses ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, with the following clarifications:

Paragraph 5.2.15 titled Review, Approval and Control of Procedures, states that, "Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary. A revision to a procedure constitutes a procedure review." In lieu of this commitment, Duke Energy addresses programmatic controls in Section 17.3.2.14 to continually identify procedure revisions which may be needed to ensure that procedures are appropriate for the circumstance and are maintained current.

11. Paragraph 5.2.16, Measuring and Test Equipment - In order to properly address this paragraph, HNP submits the following discussion of M&TE:
    IEEE Standard 498-1975 defines measuring and test equipment (M&TE) as follows:
    Devices or systems used to calibrate, measure, gauge, test, inspect, or control in order to acquire research, development, test, or operational data to determine compliance with design, specifications, or other technical requirements. M&TE does not include permanently installed operating equipment or test equipment used for preliminary checks where accuracy is not required; for example, circuit checking multimeters.
    Note: M&TE does not include rules, tape measures, levels, and other devices if normal commercial practices provide adequate accuracy.
    There is a key distinction between installed process instruments and measuring and test equipment. A piece of measuring and test equipment may be used to calibrate a number of plant instruments. Thus, a calibration error could affect a wide variety of plant equipment. Process instruments, on the other hand, perform a single function and may be used to operate equipment, verify operability of equipment, or perform a single monitoring or trip function. In the case of measuring and test equipment, the key concern when a device is out of calibration is to identify other instruments to which this accuracy has been transferred and, secondly, to prevent recurrence. In the case of process instruments, the key emphasis is to prevent recurrence of the out-of-calibration condition.
    In ANSI N18.7-1976 (and other documents), the distinction between measuring and test equipment and process instruments is not well defined.
    The requirements in the second and third paragraphs in Section 5.2.16 will be applied to measuring and test equipment and those in the first and third paragraphs applied to process instruments with the exception that process instrumentation shall be "suitably marked or tracked to indicate calibration status" versus "suitably marked to indicate calibration status." in addition, a review of out-of-calibration process instruments will be made to determine if action is required to prevent recurrence. Such action may include modification, procedural revision, or corrective maintenance. Section 17.3.2.9 provides additional requirements for control of M&TE.
12. Paragraph 5.2.17, Inspections: As a general clarification, when inspections are not contained in a separate inspection report, inspection requirements will be integrated into appropriate procedures or other documents with the procedure or document serving as the record. Records of inspections will be identifiable and retrievable.
13. Paragraph 5.2.17, second to the last sentence in the last paragraph, "Deviations, their cause, and any . . .", to be consistent with paragraph 5.2.11, the cause of the condition will be determined for only significant conditions adverse to safety.
14. Paragraph 5.3.5(4), HNP interprets the review requirements for "supporting maintenance documents" which have not been incorporated in a procedure, be performed in an equivalent manner as described in approved procedures.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Rev. 2) (Operation)
HNP complies with this guide, which endorses ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, with the following clarifications:

15. Paragraph 5.3.6, Radiation Control Procedures, Discusses certain control programs. As previously stated, paragraph 1, scope, of ANSI N18.7-1976 references those activities involved with being safety-related.
    The radiation protection program is not considered to be in this category but rather a program required to comply with 10CFR 19, 20, 30, 70, 71, and 100. Therefore, HNP shall develop its radiation protection program as stated in Section 12.5 of the HNP UFSAR.
16. Paragraph 5.3.9.3, Emergency Procedures: As directed by the NRC, HNP will follow a format for emergency procedures in accordance with 10CFR 50, Appendix E.
17. Exception to Paragraph C.3 of Regulatory Guide 1.33 and ANSI N18.7-1976 Paragraph 4.3: Independent Review Program requirements are replaced by Section 17.3.3.2, Independent Review. This exception uses NRC Safety Evaluation dated January 13, 2005 to Nuclear Management Company (ADAMS ML050210276).
18. Regulatory position C.4 modifies the audit frequencies in Section 4.5 of ANSI N18.7. Duke Energy takes exception to this regulatory position. The audits of selected aspects of operational phase activities as identified in Section 17.3.3.3.3, Internal Audit Program, are performance based scheduled based on plant performance and importance to safety but at a frequency not to exceed twenty-four months with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.
19. Paragraph C.5.d of the Regulatory Guide 1.33 will be implemented by adding the clarifying phrase "Where practicable" in front of the fourth sentence of the fifth paragraph. The Regulatory Guide's changing of the two uses of the word "should" in this sentence to "shall" unnecessarily restricts HNP's options on repair or replacement parts. It is not always practicable to test parts prior to use. Modification review in accordance with the provisions of 10CFR 50.59 will be conducted and documented.
    The words "where practical" will be determined by responsible plant management and the results documented.
20. Paragraph C.5.e of Regulatory Guide 1.33 will be implemented subject to the same clarifications made for ANSI N45.2.2.
21. Paragraph C.5.f of Regulatory Guide 1.33 will be implemented with the substitution of the word "practical" for the word "possible" in the last sentence.
22. Paragraph C.5.g of Regulatory Guide 1.33 will be implemented with the addition of the modifier "normally" after each of the verbs (should) which the regulatory guide converts to "shall". It is HNP's intent to fully comply with the requirements of this paragraph, and any conditions which do not fully comply will be documented and approved by the plant staff. In these cases, the reason for the exception shall be retained for the same period of time as the affected preoperational tests.
23. Paragraph 5.2.2, Procedure Adherence describes that for temporary changes to procedures that one of the approvers shall be the supervisor in charge of the shift and hold a senior reactor operator license. To avoid overloading the supervisor in charge of the shift with administrative tasks, any member of operation's management with a senior reactor operator license will be allowed to approve temporary changes to procedures. The change is documented and, if appropriate, reviewed and approved for incorporation in the next revision of the procedure within 14 days of implementation of the temporary change.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Rev. 2) (Operation)

HNP complies with this guide, which endorses ANSI N18.7-1976, Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants, with the following clarifications:

24. Paragraph 5.3.10 of ANSI N18.7-1976/ANS-3.2, the last sentence in the first paragraph requires "test and inspection results, shall be documented and evaluated..." also, the last sentence in the second paragraph requires "the test and inspection procedures shall require recording the date, identification of those performing the test or inspection, as-found condition, corrective actions performed, if any, and as-left condition." as an alternative to the records required for inspections outlined in paragraph 5.3.10, HNP shall provide the following as the method to document results of inspections:
    the results of inspections will be documented in appropriate records and those records shall, as a minimum, identify (A) through (H) below:
    (A) authorized individual approving results.
    (B) date of inspection.
    (C) inspector/data recorder.
    (D) item inspected.
    (E) M&TE used.
    (F) reference to information on action taken in connection with non-conformances.
    (G) results or acceptability.
    (H) type of observation.

Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0)

HNP shall comply with the requirements of ANSI N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.37-March 1973, with the following clarifications:

1. Paragraph 2.5, Test Equipment, outlines control of inspection and test equipment. HNP has addressed its position relative to measuring & test equipment (M&TE) in 17.3.2.9.
2. Paragraph 5, Installation Cleaning: The recommendation that local rusting on corrosion resistant alloys be removed by mechanical methods is interpreted to mean that local rusting may be removed mechanically, but the use of other removal means is not precluded provided other cleaning methods are not considered detrimental as determined by responsible plant management.
3. The guide and standard are applicable to those areas of the quality assurance program addressing on-site cleaning of materials and components, cleanness control, preoperation cleaning and layup of fluid systems.
4. With regard to paragraph C.3 of Regulatory Guide 1.37: Chromates or other additives, normally in the system water, will not necessarily be added to the flush water.
5. With regard to paragraph C.4 of Regulatory Guide 1.37: Expendable materials, such as inks and related products; temperature indicating sticks; tapes; gummed labels; wrapping materials; water soluble dam materials; lubricants, NDT penetrant materials and couplants, desiccants, which contact stainless steel or nickel alloy surfaces shall be of commercial quality. Levels for halogens, sulfur, chlorides, low melting point metal, etc., for use on stainless steel and nickel alloy surfaces will be as determined by responsible technical group to limit or preclude intergranular cracking and stress corrosion cracking.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2)
HNP shall comply with the requirements of ANSI N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.38 with the following clarifications:

1. Paragraph 2.1, Planning: (first sentence) the specific items to be governed by the standard shall be identified. However, the standard is part of the HNP QA program and it will, therefore, be applied to those structures, systems, and components which are included in that program.
2. Paragraph 2.3 - Results - The full requirements of this paragraph shall apply to the inspections and tests that are performed to determine the acceptability of product quality.
3. Paragraph 2.4 - those personnel that perform inspection, examination, and testing activities for verification and acceptance/rejection purposes shall be qualified in accordance with Regulatory Guide 1.58.
4. Paragraph 2.5 - Measuring and Test Equipment (2.5.2) - That equipment which measures quality of the permanent plant items shall be under the calibration and control program; whereas the equipment used to measure secondary conditions, such as warehouse temperature, humidity, etc., will be maintained in good working order and checked for proper functioning when accuracy is in doubt, but not maintained under the calibration and control program. Traceability to calibration records will be provided when it is impractical (because of size, configuration, or application) to physically mark calibration information on the item. Note: M&TE does not include rulers, tape measures, levels, and other devices if normal commercial practices provide adequate accuracy.
5. Paragraph 2.7, Classification of Items: HNP may choose not to explicitly use the four level classification system. However, the specific requirements of the standard that are appropriate to each class will be applied unless justified and documented.
6. Paragraph 2.7.1(3) requires special nuclear material (fuel) and sources to be classified as Level A. HNP shall store new/used nuclear fuel and radioactive sources in storage locations as described in the Chapters 9 and 12 of the UFSAR. Radioactive sources used by HP personnel shall be stored and controlled in accordance with HP practices and procedures.
7. Paragraph 3.2 - Levels of Packaging - Packaging for shipment off-site will be equal to or exceed the original packaging by the vendor, as required to assure the quality of the item is not degraded as a result of shipping or handling.
8. Paragraph 3.4, Methods of Preservation: (first sentence) HNP will comply with these requirements subject to the clarification that the term "deleterious corrosion" means corrosion which cannot be subsequently removed and which adversely affects form, fit, or function.
9. Paragraph 3.6 - Barrier and Wrap Materials and Desiccants - The use of clear plastic in warehouses will be minimized. The guide rule is that the clear plastic shall be used only where periodic visual inspection is necessary.
   Plastic wrap on items supplied in accordance with a vendor's approved QA/QC program will be accepted and stored without rewrapping.
10. Paragraph 3.7, Containers, Crating and Skids: In lieu of the requirements of this paragraph, HNP will use means as determined by responsible plant technical personnel needed to provide adequate protection of the items in storage.
11. Paragraph 4 - Shipping - Requirements of paragraph 4, Shipping, primarily applies to the vendor. Plant functions with regard to return shipments will meet or exceed the methods of the vendor for the item or approved alternatives.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2)
HNP shall comply with the requirements of ANSI N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.38 with the following clarifications:

12. Paragraph 5.2.1, Shipping Damage Inspection: Warehouse personnel will normally visually scrutinize incoming shipments for damage of the types listed in this paragraph; this activity is not necessarily performed prior to unloading. Since required items receive the item inspection of paragraph 5.2.2, separate documentation of the shipping damage inspection is not necessary. Release of the transport agent after unloading and the signing for the receipt of the shipment may be all of the action taken to document completion of the shipping damage inspection. Any nonconformances noted will be documented and dispositioned as required by 17.3.2.13. The person performing the visual scrutiny during unloading is not considered to be performing an inspection function as defined under Regulatory Guide 1.74; therefore, while he will be trained and qualified to perform this function, he may not necessarily be certified (N45.2.6) as an inspector.
13. Paragraph 5.2.2, Item Inspection: The need and extent for inspection of items will be determined by responsible plant technical personnel. Receiving inspections shall be performed in an area designated for receipt of material and shall normally be performed in the receiving building. The receiving building and the areas designated will provide adequate protection for the material, but may not comply with all of the specific requirements contained in Section 6 of this standard. Material that is suspected of being compromised during the receiving process shall be evaluated by responsible technical personnel, as determined by plant management.
14. Paragraph 5.2.2(1) - Identification and Marking - Item inspection will include inspection for identification and marking required by the purchase order documents. Marking that is not quality related or which provides no traceability will not be inspected.
15. Paragraph 5.3.1 - Acceptable - Item acceptance status will be indicated by application of tags, stickers, ribbons, or signs. Storage areas are not designated as accept areas except for bulk items.
16. Paragraph 6.1.1 - Scope - The levels and methods of storage for items between the time of removal from the prescribed storage until placement in the installed location may be relaxed as determined by responsible plant management for short periods of time, according to the sensitivity of the item being handled and the elements of contact anticipated during this interval. Where relaxation of storage requirements of this standard are deemed appropriate, the item, conditions, precautions and follow-up inspection for assurance that quality of the item has been maintained will be documented.
17. Paragraph 6.1.2, Levels of Storage: Subpart (2) is replaced with the following:
    (2) Level B items shall be stored within a fire-resistant, weather-tight, and well ventilated building or equivalent enclosure. This building shall be situated and constructed so that it will not normally be subject to flooding; the floor shall be paved or equal, and well drained. If any water comes in contact with stored equipment, such equipment will be labeled or tagged nonconforming, and then the nonconformance document will be processed and evaluated. Items shall be placed on pallets, shoring, or shelves to permit air circulation. The building shall be provided with heating and temperature control or its equivalent to reduce condensation and corrosion. Minimum temperature shall be 40°F and maximum temperature shall be 140°F or less if so stipulated by a manufacturer.
18. Paragraph 6.2.1, Access to Storage Areas: Items which fall within the level d classification of the standard will be stored in areas which may be posted to limit access, but other positive controls such as fencing or guards will not normally be provided.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2)
HNP shall comply with the requirements of ANSI N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.38 with the following clarifications:

19. Paragraph 6.2.4, Storage of Food and Associated Items: The sentence is replaced with the following: "The use or storage of food, drinks, and salt tablet dispensers in any storage area shall be controlled and shall be limited to designated areas where such use or storage is not deleterious to stored items."
20. Paragraph 6.2.5, Measures to Prevent Entrance of Animals: The sentence is replaced with the following: "Warehouse personnel shall be alert to detect evidence of rodents or small animals in indoor storage areas.
    Consideration will be given when setting up the system to provide reasonable assurance that rodents or other small animals will not be present. If any such evidence is detected, a survey or inspection will be utilized to determine the extent of the damage; exterminators or other appropriate measures shall be used to control these animals to minimize possible contamination and mechanical damage to stored material."
21. Paragraph 6.3.3, Storage of Hazardous Material: The sentence is replaced with the following: "Hazardous chemicals, paints, solvents, and other materials of a like nature shall be stored in approved cabinets or containers which are not in close proximity to installed safety systems required for safe shutdown."
22. Paragraph 6.4.2, Care of Items: The following alternates are provided for indicated subparts:
    (5) "Space heaters in electrical equipment shall be energized unless a documented engineering evaluation determines that such space heaters are not required."
    (6) "Large (greater than or equal to 50 hp) rotating electrical equipment shall be given insulation resistance tests on a scheduled basis unless a documented engineering evaluation determines that such tests are not required."
    (7) "Prior to being placed in storage, rotating equipment weighing over approximately 50 pounds shall be evaluated by engineering personnel to determine if shaft rotation in storage is required; the results of the evaluation shall be documented. If rotation is required, it shall be performed at specified intervals, and documented.
    The degree of turn shall be established so that the parts receive a coating of lubrication where applicable, and so that the shaft does not come to rest in the position prior to rotation. (90 deg. and 450 deg. rotations are examples.) For long shafts or heavy equipment subject to undesirable bowing, shaft orientation after rotation shall be specified and obtained."
    (8) Other maintenance requirements specified by the manufacturer's instructions shall be evaluated by responsible plant personnel to determine applicability during storage of the item.
23. Paragraph 6.5, Removal of Items from Storage: HNP does not consider the last sentence of this paragraph to be applicable to the operations phase due to the relatively short period of time between installation and use. The first sentence of the paragraph is replaced with: "HNP will develop, issue, and implement a procedure(s) which cover(s) the removal of items from storage. The procedure(s) will assure that the inspection status of all material issued is known, controlled, and appropriately dispositioned."
    When items are released and waiting at a location prior to installation, responsible plant management in accordance with plant procedures will determine and document the extent of inspection and storage requirements.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2)

HNP shall comply with the requirements of ANSI N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.38 with the following clarifications:

24. Paragraph 6.6, Storage Records: HNP will comply with the requirements of this section with the clarification that, for record purposes, personnel access to storage areas will not be recorded. Unloading or pick-up of material shall not be considered "access," nor shall inspection by NRC or other regulatory agents, nor shall tours by non-HNP employees who are accompanied by HNP employees.
25. Paragraph 7.3 - Hoisting Equipment - The load chart for each crane includes the model number for that crane. This load chart is considered to be "certification" by the manufacturer for that crane as required by paragraph 7.3.1. Likewise, forklifts are considered certified by the manufacturer's literature giving maximum capacity as required by Paragraph 7.3.2.
    Paragraph 7.3, Hoisting Equipment: Rerating of hoisting equipment will be considered only when absolutely necessary. Prior to performing any lift above the load rating, the equipment manufacturer will be contacted for his approval and direction. The manufacturer will be requested to supply a document granting approval for a limited number of lifts at the new rating and any restrictions involved, such as modifications to be made to the equipment, the number of lifts to be made at the new rating, and the test lift load. At all times, the codes governing rerating of hoisting the equipment will be complied with.
    If rerating of hoisting equipment is necessary and HNP cannot or does not contact the equipment manufacturer as described above, the test weight used in temporarily rerating hoisting equipment for special lifts will be at least equal to 110 percent of the lift weight. A dynamic load test over the full range of the lift using a weight at least equal to the lift weight will be performed.
26. Paragraph 7.4 - Inspection of Equipment and Rigging - Nondestructive examinations will be performed by QC personnel qualified in accordance with Regulatory Guide 1.58 (except as amended by safety analysis report position). Operators will be trained in the operation and maintenance inspections of their assigned equipment.
27. Appendix A.3.5.1 - Caps and Plugs; A.3.5.2, Tapes and Adhesives; and A.3.6.3, Desiccants - Plugs, caps, tapes, adhesives, desiccants, markers and other temporary items will be of commercial quality. Levels for halogens, sulfur, chlorides, low melting point metal, etc., for use on stainless steel and nickel alloy surfaces will be as determined by the responsible technical group to limit or preclude intergranular cracking and stress corrosion cracking.

Regulatory Guide 1.39, Housekeeping Requirements for Water Cooled Nuclear Power Plants (Rev. 2)

HNP complies with the requirements of ANSI N45.2.3-1973, Housekeeping, During the Construction Phase of Nuclear Power Plants, as endorsed by Regulatory Guide 1.39, September 1977, with the following clarifications for:

1. Paragraph 2.1, Planning: The zone designations provided in the standard will be used as a guide in developing plant procedures; however, plant areas will not necessarily be divided into zones I through V. Equivalent controls will be maintained as prescribed in approved procedures.
2. Paragraph 3.5, Surveillance, Inspection, and Examinations: Subparagraph (1) is not applicable during normal operations but will be implemented if large items are to be moved or handled.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel (Rev. 1)
HNP shall comply with NRC Regulatory Guide 1.58, Revision 1, which endorses ANSI N45.2.6-1978, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants, with the following clarifications:

1. With regard to paragraph 1.2 of ANSI N45.2.6-1978 titled Applicability: HNP elects not to apply the requirements of this guide to those personnel who are involved in the daily operations of surveillance, maintenance, and certain technical and support services whose qualifications are controlled by 17.3 or are controlled by other QA program commitment requirements. Only personnel in the following listed categories will be required to meet ANSI N45.2.6-1978 requirements: (1) nondestructive examination (NDE) personnel (2) QC inspection personnel, and (3) receipt inspection personnel.
2. The fourth paragraph of Paragraph 1.2 requires that the standard be imposed on personnel other than HNP employees. The applicability of the standard to suppliers and contractors will be documented and applied as specified in the procurement documents for each supplier and contractor or in interface agreements for Duke Energy non-nuclear organizations providing services identified in Section 17.3.1.2.3.
3. With regard to Paragraph 2.5 of ANSI N45.2.6-1978 titled Physical: HNP will implement the requirements of this section with the stipulation that, where no special physical characteristics are required, none will be specified. The converse is also true: if no special physical requirements are stipulated by HNP, none are considered necessary. HNP employees receive an initial physical examination to assure satisfactory physical condition; however, only the following listed personnel will receive an annual examination: (1) NDE personnel (2) QC inspection personnel, and (3) receipt inspection personnel. This annual examination shall consist of the near visual acuity using the standard Jaeger's type chart or equivalent test.
4. With regard to Paragraph 3 of ANSI N45.2.6-1978 titled Qualifications: Only personnel performing NDE (such as LP, MT, UT, and RT) are required to be grouped in levels of capability and certified for inspection, review, and evaluation of inspection data, and reporting of inspection and test results. Inspection personnel are qualified based on pre-established experience, education, on-the-job training, written examinations and proficiency tests associated with the specific activity. Proficiency tests are given to personnel performing independent QC inspections and documented acceptance criteria are developed to determine if individuals are properly trained and qualified. Certificates of qualification delineate the functions personnel are qualified to perform. Qualification records are maintained and performance evaluations conducted at least once every three years. If organizations elect to utilize qualifications by levels for non-NDE inspections, Level I inspectors receive a minimum of 4 months experience as Level I before being certified as Level II, in lieu of one year experience recommended by ANSI N45.2.6 Section 3.5.2(1). Organizations identify in their procedures if they qualify their inspectors by Level or by task qualifications. Inspectors are only assigned functions for which they have been qualified.
5. With regard to Paragraph 3.5 of ANSI N45.2.6-1978 titled Education & Experience Recommendations: HNP will certify individual inspectors through training and experience to requirements appropriate to the specific assignment; however, except for NDE, personnel are not required to be classified by levels of capability. The training experience requirements will be directed toward qualifying personnel for specific inspection and testing operations.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants (Rev. 2)

HNP shall comply with NRC Regulatory Guide 1.64, Rev. 2, which endorses ANSI standard N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants, with the following clarification:

Paragraph C.2(1): For the exceptional circumstance in which the designer's immediate supervisor is the only technically qualified individual available, this review can be conducted by the supervisor, provided that: i) the other provisions of the regulatory guide are satisfied, ii) the justification is individually documented and approved in advance by the supervisor's management, and iii) quality assurance audits cover frequency and effectiveness of the use of supervisors as design verifiers to guard against abuse.

Regulatory Guide 1.74, Quality Assurance Terms and Definitions (Rev. 0)

Regulatory Guide 1.74 endorses ANSI N45.2.10-1973, Quality Assurance Terms and Definitions. The HNP project complies with this guide as described below:

HNP complies with the requirements of this guide with the following clarifications:

1. HNP reserves the right to define additional words or phrases which are not included in this standard. Such additional definitions will be documented in appropriate procedures, manuals, etc.
2. In addition to the standard's definition of "inspection," HNP will use the following: "Inspection (when used to refer to activities that are not performed by quality organization personnel) - examining, viewing closely, scrutinizing, looking over or otherwise checking activities. Personnel performing these functions are not necessarily certified to ANSI N45.2.6."

   When HNP intends for inspection to be performed in accordance with the QA program by personnel certified as required by that program and for activities defined by "Inspection" in ANSI N45.2.10, appropriate references to the plant quality organization which will perform the activity and/or to quality procedures to be used for performing the activity will be made. If such references are not made, inspections are considered under the additional definition given above.
3. In addition to the standard's definition of "procurement documents," HNP will utilize the definitions given in ANSI N45.2.13 and in Regulatory Guide1.74. The compound definition, procurement documents-contractually binding documents that identify and define the requirements which items or services must meet in order to be considered acceptable by the purchaser. They include documents which authorize the seller to perform services or supply equipment, material or facilities on behalf of the purchaser (e.g. contracts, letters of intent, work orders, purchase orders, or proposals and their acceptance, drawings, specifications, or instructions which define requirements for purchase).
4. "Quality assurance program requirements" (not defined in ANSI N45.2.10, but used and defined differently in ANSI N45.2.13) - those individual requirements of the QA program which, when invoked in total or in part, establish the requirements to the quality assurance program for the activity being controlled. Although not specifically used in the operational QA program, ANSIN45.2 may be imposed upon HNP's suppliers.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.74, Quality Assurance Terms and Definitions (Rev. 0)
Regulatory Guide 1.74 endorses ANSI N45.2.10-1973, Quality Assurance Terms and Definitions. The HNP project complies with this guide as described below:
5. "Independent Verification" - Verification that required actions have been completed by an individual other than the person who performed the operation or activity being verified. Such verification will not require confirmation of the identical action when other indications provide assurance or indication that the prescribed activity is in fact complete. Examples include, but are not limited to: verification of a breaker opening by observing remote breaker indication lights; verification of a set point (made with a voltmeter or ammeter for example) by observing the actuation of status or indicating lights are the required panel-meter indicated value; verification that a valve has been positioned by observing the starting or stopping of flow on meter indications or by remote valve position indicating lights. 6. "Audit" (will be a modification of the word - to allow the use of subjective evidence if available - as defined in paragraph 1.4 of ANSI N45.2.12-1977 and paragraph 1.4.3 of ANSI N45.2.23-1978 as opposed to the definition given in ANSI N45.2.10-1973) - A documented activity performed in accordance with written procedures or checklists to verify, by examination and evaluation of objective evidence where available, that applicable elements of the quality assurance program have been developed, documented, and effectively implemented in accordance with specified requirements. An audit should not be confused with surveillance or inspection for the sole purpose of control or product acceptance.
Regulatory Guide 1.88, Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records (Rev. 2)
HNP shall comply with NRC Regulatory Guide 1.88, Rev. 2, which endorses ANSI N45.2.9-1974, Collection, Storage, and Maintenance of QA Records, with the following clarifications:
See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records. 1. Appendix A of ANSI N45.2.9 is not considered to be a mandatory list. This list will be used as a guideline for classifying those documents that need to be maintained as QA records. Whether a particular type of document needs to be classified as a QA record and its appropriate retention period is determined in accordance with records management procedures. 2. Paragraph 1.4, Definitions: The phrase "When the document has been completed" is clarified to mean when the document has received the final review performed by the organizational element responsible for generating or collecting the records. In the case of a record package (plant change request, equipment qualification, etc.) made up of several individual documents, the package will be considered to be the document for the purpose of determining when the document is complete. 3. Paragraph 3.2.1, Generation of Quality Assurance Records: The phrase "Completely filled out" is clarified to mean that sufficient information is recorded to fulfill the intended purpose of the record. 4. Paragraph 3.2.2, Index: The storage location will be delineated in procedures in lieu of in the index. The specific location of a record "within a storage area" is delineated by a computerized indexing system plus a storage area labeling system which provides information by record type and storage medium. 5. Paragraph 4.2, Timeliness: HNP 's contractual agreement with its contractors and suppliers will constitute fulfillment of the requirements of this paragraph.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.88, Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records (Rev. 2)

HNP shall comply with NRC Regulatory Guide 1.88, Rev. 2, which endorses ANSI N45.2.9-1974, Collection, Storage, and Maintenance of QA Records, with the following clarifications:

6. Paragraph 5.4, Preservation: The following clarification is substituted for the current subparagraph 5.4.2: "Records shall not be stored loosely. They shall be secured for storage in file cabinets or on shelving in containers." the following clarification is substituted for the current subparagraph 5.4.3: "appropriate provisions shall be made for special processed records (such as radiographs, photographs, negatives, microfilm, and magnetic media) to prevent or minimize damage from excessive light, stacking, electromagnetic fields, temperature and humidity, etc. Manufacturer's recommendations will be considered as appropriate."
7. Paragraph 5.5, Safekeeping: Routine general office and nuclear site security systems and access controls are provided. No special security systems are required to be established for record storage areas.
8. Paragraph 5.6, Facility: This paragraph provides no distinction between temporary and permanent facilities. To cover temporary storage, the following clarification is added: "complete records may be stored in one-hour fire rated file cabinets until transmitted for permanent storage. In general, records shall not be maintained in temporary storage for more than ninety days after completion.
   Any exceptions to this requirement must be justified, evaluated and approved by the supervisor document services or designee and documented. A list of exceptions shall be maintained and available for NRC review. Exceptions may include records needed on a continuing basis for an extended period of time at the location of the work group responsible for generating the records and records which are cumulative in nature and could best be turned over for storage for a designated period of time.
9. Paragraph 5.6, subparagraph 3, is clarified to require a two-hour minimum fire rating to be consistent with the 1979 version of the standard and NRC Criteria for Record Storage Facilities (Guidance - ANSI N45.2.9, Section 5.6) issued 7/1/80.
10. Paragraph 5.6, subparagraph 9, is clarified to read: "No pipes or penetrations except those providing fire protection, lighting, temperature/humidity control or communications are to be located within the facility. All such penetrations shall be sealed or dampened to comply with a minimum two-hour fire protection rating."
11. Additional clarification for QA records is provided in Section 17.3.2.15.
12. See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records.
13. See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records.

Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1)

HNP complies with the requirements and guidance of ANSI N45.2.5-1974, Supplementary Quality Assurance Requirements for Installation Inspections and Testing of Structural Steel During the Contract Phase of Nuclear Power Plants, as it is referenced in Regulatory Guide 1.94, Rev. 1, with the following clarifications:

A) Paragraph 2.1, Planning: Requirements, as determined by responsible plant management, will be incorporated into procedures.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1)
HNP complies with the requirements and guidance of ANSI N45.2.5-1974, Supplementary Quality Assurance Requirements for Installation Inspections and Testing of Structural Steel During the Contract Phase of Nuclear Power Plants, as it is referenced in Regulatory Guide 1.94, Rev. 1, with the following clarifications: B) Paragraph 2.3, Results, Will be implemented as set forth in Sections 17.3.2.12, 17.3.2.8, and 17.3.2.15 and Regulatory Guide 1.33. C) Paragraph 2.5 of ANSI N45.2.5, Measuring & Test Equipment, Requires certain controls over this type of equipment. The equipment listed shall be included in the calibration control program; however, the basis and control of measuring and test equipment is that stated in Section 17.3.2.9. D) The cement test frequency for standard physical and chemical properties is in accordance with ASTM C 183, on the basis of one test per daily production at the cement plant, reference ANSI N45.2.5, Table B. Table B also lists a test frequency for ASTM C 235 which has been discontinued by ASTM. HNP plans to discontinue testing in accordance with ASTM C 235. Acceptance of aggregates for durability/hardness will be in accordance with ASTM C 131 OR C 535, Los Angeles Abrasion Test. E) Gradation - In addition to the gradations listed in ASTM C-33, an aggregate designated 78-M (State of North Carolina designation) is used in special areas such as around major penetrations or in reinforcing steel congested areas, with the approval of the engineers. This aggregate meets all other qualifications of ASTM C-33, with the exception of gradation analyses. The results during preliminary concrete mix design have been satisfactory and in accordance with the requirements of ASME Section III, Division 2/ACI-359 code. F) Paragraph 5.4, High Strength Bolting: Bolting connection points will be visually inspected in accordance with ANSI N45.2.5-1974 except that bolt length will be checked to ensure bolts are long enough as indicated by the point of the bolts being flush with or outside the face of the nuts in accordance with ANSI N45.2.5-1978.
Regulatory Guide 1.116, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems, (Rev. 0-R)
HNP complies with the requirements of ANSI N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.116, Revision O-R, June 1976, with the following clarifications: 1. Paragraph 2.1, Planning: Requirements, as determined by responsible plant management, will be incorporated into procedures. 2. Paragraph 2.3, results, will be implemented as set forth in Section 17.3.2.12 and by compliance with RG 1.33. 3. Paragraph 2.8, Measuring and Test Equipment - HNP has addressed this requirement for the operational phase of the plant in Section 17.3.2.9. 4. Paragraph 2.9, Prerequisites, References requirements of other standards. HNP has addressed applicable standards in the appropriate sections of the HNP UFSAR in lieu of the requirements of this paragraph. The extent to which this paragraph applies will be determined by responsible plant management based on end use and complexity of the item. 5. Paragraph 3.3, Processes and Procedures: "Approved instructions" are interpreted to include vendor manuals.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.116, Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems, (Rev. 0-R)

HNP complies with the requirements of ANSI N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.116, Revision O-R, June 1976, with the following clarifications:

6. Paragraph 4.6, Care of Items: This will be done as outlined in the position on Regulatory Guide 1.38.
7. Paragraph 5, including subparagraphs 5.1 through 5.4, Installed Systems, Inspections and Tests: Responsible plant management will determine the extent to which the elements in this paragraph are applied when developing test requirements for inclusion in modification procedures. In some cases, testing requirements may be met by post-installation surveillance testing in lieu of a special post-installation test.

Regulatory Guide 1.123, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, (Rev. 1)

HNP shall comply with the requirements of ANSI N45.2.13-1976, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.123 with the following clarifications:

1. Paragraph 1.2.2, Purchaser's Responsibilities: Item C is one of the options which may be used by HNP to assure quality; however, any of the options given in 10CFR50, Appendix B, Criterion VII as implemented by 17.3 may also be used. Evaluation of supplier's QA program will be conducted as determined depending on complexity and end use of item.
2. Paragraph 3.1, Procurement Document Preparation, Review and Control Change: The changed document may not always be as reviewed by the originator; however, at least an equivalent level shall review and approve any changes.
3. Paragraphs 3.2.3, 3.2.4, and 3.2.6 - HNP does not consider that these paragraphs or vendor qualifications apply for the procurement of off-the-shelf items. Off-the-shelf items (which include original as well as spare and replacements) are Commercial Grade Items which are defined in 10CFR 21.
   Special quality verification requirements shall be determined, as necessary, by responsible technical group to assure acceptability of the item. The responsible technical organization will review purchase requisitions of items classified as "commercial grade" to assure proper application of the 10CFR 21 criteria.
   See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.
4. Paragraph 3.3 requires procurement documents to be reviewed prior to bid or award of contract. The documented review of procurement documents is provided through review of the procurement specification and purchase requisition by the responsible technical organization prior to bid or award of contract.
5. Paragraph 3.4, Procurement Document Control: HNP will meet the requirements of 17.3 in lieu of the requirements specified in this paragraph.
6. Paragraph 4.2, Selection Measures, Outlines certain methods acceptable for the selection of suppliers. HNP's history of using similar methods has proven adequate in the procurement of items; therefore, HNP wishes to replace paragraph 4.2(a), (b), and (c) with the following selection methods:

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.123, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, (Rev. 1)
HNP shall comply with the requirements of ANSI N45.2.13-1976, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.123 with the following clarifications:

1) The supplier's quality assurance capabilities as determined by a direct survey of his facilities and personnel, and the implementation of his quality assurance program.
2) Evaluating the supplier's history of providing a product which performs satisfactorily in actual use. One or more of the following information shall be evaluated:
   (i) Experience of users of identical or similar products of the same prospective supplier.
   (ii) HNP's records that have been accumulated in connection with previous procurement actions and product operating experience. Historical data should be representative of the supplier's current capability. If there has been no recent experience with the supplier, or if he is a new supplier, the prospective supplier shall be requested to submit information on a similar item or service for evidence of his current capabilities.
   (iii) Evaluating the supplier's current quality records supported by documented qualitative and quantitative information which can be objectively evaluated.
   This would include review and evaluation of the supplier's quality assurance program manual and procedures, as appropriate, to ensure that the applicable requirements of 10CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants" are met.
   (iv) Verification that the supplier holds an active certificate of authorization from the ASME to supply or manufacture materials or the item(s) described in the purchase requisition. A supplier may be considered acceptable, without a survey, to supply off-the-shelf items. An inspection shall be performed to assure that the correct item was received and no damage exists.
   Verification that the supplier is listed in the current NUPIC (Nuclear Procurement Issues Committee) database. However, the audit report which formed the basis for listing the supplier in the NUPIC database must be obtained and reviewed for applicability to the procurement. All deficiencies which could degrade the procured item must be resolved prior to the procurement. This review shall be documented and, together with the audit report, be retained.
3) See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.

7. Paragraphs 5.2 and 5.3 shall be applied to the extent determined by responsible plant management based on complexity of the item and its end use. It is not intended that these paragraphs be applied to spares or replacement parts that do not change original design intent.
8. Paragraph 6.1, General, Outlines methods for monitoring and evaluating supplier performance. HNP wishes to replace paragraph 6.1(a), (b), (c), (d), and (e) with the following methods for monitoring and evaluating supplier performance:
   A. Reviewing documents generated or processed during activities fulfilling procurement requirements.
   B. Reviewing LER'S.
   C. Periodic audits.
   D. Annual evaluations.
   E. Those controls specified 17.3.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.123, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, (Rev. 1)
HNP shall comply with the requirements of ANSI N45.2.13-1976, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.123 with the following clarifications:

9. Paragraph 6.4, Control of Changes in Items or Services: Since ANSI N45.2 does not apply to the operational phase, equivalent controls outlined in ANSI N18.7-1976 will be used in lieu of the requirements of ANSI N45.2, Section 7.
10. Paragraph 7.4, Measuring and Test Equipment, outlines certain measures to be taken. HNP for the operating phase has addressed the topic of measuring and test equipment in 17.3.2.9 in lieu of the requirements in this paragraph.
11. Paragraph 8 provides guidance for purchaser review and disposition of vendor nonconformances. HNP, as purchaser, requires as a minimum deviations to procurement documents and previously approved supplier documents that cannot be brought into conformance prior to shipment of the material to be submitted to dep for approval. Such deviations, when approved by purchaser, are required to be submitted along with shipment of the material. Additionally, paragraph 8.2, disposition: the third sentence of item b is revised to read:
    Nonconformances to the contractual procurement requirements or purchaser approved documents which consist of one or more of the following shall be submitted to the purchaser for approval of the recommended disposition prior to shipment, when the nonconformance could adversely affect the end use of a *module or shippable component relative to safety, interchangeability, operability, reliability, integrity, or maintainability:
    A. Technical or material requirement is violated;
    B. Requirement in supplier documents, which have been approved by the purchaser, is violated;
    C. Nonconformance cannot be corrected by continuation of the original manufacturing process or by rework; and/or
    D. The item does not conform to the original requirement, even though the item can be restored to a condition such that the capability of the item to function is unimpaired.
    A module is any assembly of interconnected components which constitute an identifiable device, instrument, or piece of equipment. A module can be disconnected, removed as a unit, and replaced with a spare. It has definable performance characteristics which permit it to be tested as a unit. A module could be a card or other subassembly of a larger device, provided it meets the requirements of this definition.
12. Regulatory Position C.3 indicates that purchaser should verify the implementation of the supplier's corrective action systems when such a system is required, but this verification need not be included as part of the purchaser's corrective action measures.
    HNP interprets this statement to mean that once corrective action has been verified by purchaser on nonconforming vendor items, the items can be released for use in its intended application.
    The cause and action to preclude recurrence of deficiencies is the responsibility of the vendor, and independent verification of such vendor action by purchaser or vendor notification of such action to purchaser, is not required on the basis that the vendor's QA program has been accepted by the purchaser. The QA program provides for determining cause and action to preclude recurrence on significant deficiencies, and purchaser audits are conducted to ensure vendor's compliance with his accepted QA program commitments. In addition, HNP will provide overview of those causes and corrective action activities associated with items of high volume and which are considered significant to safety in cases where vendor's recent performance has appeared marginal.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.123, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, (Rev. 1)
HNP shall comply with the requirements of ANSI N45.2.13-1976, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants, as it is endorsed by Regulatory Guide 1.123 with the following clarifications:
13. Paragraph 10.2a: HNP will comply with this paragraph to the extent that for non-code items, certificates of compliance will be traceable only to the purchase order and not to the specific item.
Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants (Rev. 0)
HNP shall comply with requirements of Regulatory Guide 1.144, January 1979, which endorses ANSI N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants, with the following clarifications.
1. C.3.(B)(2): The concepts of when audits are required, i.e., annually, triennially, will be complied with; however, such audits would only be required of the vendor if the vendor is involved with an active contract/procurement document. This concept is as discussed in paragraphs 3.5.3.1 and 3.5.3.2 of ANSI N45.2.12-1977. See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories. 2. Paragraph 2.3, Training: The training of HNP audit personnel will be accomplished as described in HNP's position on Regulatory Guide 1.146. 3. Paragraph 2.4, Maintenance of Proficiency: The maintenance of proficiency of HNP audit personnel will be accomplished as described in HNP's position on Regulatory Guide 1.146. 4. Paragraph 3.2.2 indicates that objective evidence is to be examined and evaluated. HNP believes that the use of subjective evidence is also an important element of the audit program. See paragraph 4.3.2 clarifications below. 5. Paragraph 3.3, Essential Elements of the Audit System; HNP will comply with subparagraph 3.3.5 as it was originally written (subparagraph 3.2.5) in ANSI N45.2.12, Draft 3, Revision 4: "Provisions for reporting on the effectiveness of the quality assurance program to the responsible management." For the audited organization, effectiveness is reported as required in Section 17.3.3.3 and by audit procedures. Other than audit reports, HNP may not directly report on the effectiveness of the quality assurance programs to the audited organization, when such organizations are outside of Duke Energy. Subparagraph 3.3.7 requires verification of effective corrective action on a "timely basis". Timely basis is interpreted to mean within the period of time that is accepted by the organization. Each finding requires a response and a corrective action completion date. These dates are subject to revision and must be escalated to higher authority when there is a disagreement between the audited and the auditing organization on what constitutes "timely corrective action."

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants (Rev. 0)
HNP shall comply with requirements of Regulatory Guide 1.144, January 1979, which endorses ANSI N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants, with the following clarifications.

6. Paragraph 4.3.1, Preaudit Conference: HNP will comply with the requirement of this paragraph by inserting the word "normally" at the beginning of the first sentence. This clarification is required because, in the case of certain unannounced audits or audits of a particular operation or work activity, a preaudit conference might interfere with the spontaneity of the operation or activity being audited. In other cases, persons who should be present at a preaudit conference may not always be available. Such lack of availability should not be an impediment to beginning an audit. Even in the above examples, which are not intended to be all inclusive, the material set forth in paragraph 4.3.1 will normally be covered during the course of the audit.
7. Paragraph 4.3.2, Audit/Assessment Process:
   A. Subparagraph 4.3.2.2 could be interpreted to limit auditors to the review of only objective evidence. Sometimes objective evidence may not be available; therefore, HNP will comply with an alternate sentence which reads: "When available, objective evidence shall be examined for compliance with quality assurance program requirements. If subjective evidence is used (e.g., personnel interviews, direct observations by the auditor), then the audit report or checklist must indicate how the evidence is obtained."
   B. Subparagraph 4.3.2.4 is modified as follows to take into account the fact that some nonconformances are virtually "obvious" with regards to the needed corrective action. As a result of this, HNP proposes the following alternate words: "When a nonconformance or quality assurance program deficiency is identified as a result of an audit, unless the apparent cause, extent, and corrective action is readily evident, further investigations shall be conducted by the audited organization in an effort to identify the cause and effect and to determine the extent of the corrective action required."
   C. Subparagraph 4.3.2.5 contains a statement "acknowledged by a member of the audited organization." This is clarified to mean that "A member of the audited organization has been informed to the findings. Agreement or disagreement with a finding may be expressed in the response from the audited organization."
   D. Subparagraph 4.3.2.6 is modified as follows to account for the fact that immediate notification is not always possible: "Conditions requiring immediate corrective action (i.e., those which are so severe that any delay would be undesirable) shall be reported as immediately as practical to management of the audited organization."
8. Paragraph 4.3.3, Post Audit Conference: HNP will substitute and comply with the following paragraphs: "For all external audits, a postaudit conference shall be held with management of the audited organization to present audit findings and clarify misunderstandings. Where no adverse findings exist, this conference may be waived by management of the audited organization. Such waiver shall be documented in the audit report. For all internal audits unless unusual operating or maintenance conditions preclude attendance by appropriate managers/supervisors, an audit debrief shall be held with managers/supervisors. If there are no adverse findings, management of the internal assessed organization may waive the audit debrief. Such waiver shall be documented in the audit report."
9. Paragraph 4.4, Reporting:

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.144, Auditing of Quality Assurance Programs for Nuclear Power Plants (Rev. 0)
HNP shall comply with requirements of Regulatory Guide 1.144, January 1979, which endorses ANSI N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants, with the following clarifications.

A. This paragraph requires that the audit report shall be signed by the audit team leader which is not always the most expeditious route for the audit report to be issued as soon as practical. HNP will comply with Paragraph 4.4 as clarified by the following words: "An audit report, which shall be signed by the unit team leader, or his supervisor in the absence of the audit team leader shall provide:" in cases where the audit report is not signed by the lead auditor due to his absence, the record copy of the report must be signed by the lead auditor upon his return. The report shall not require the lead auditor's review/concurrence/signature if the lead auditor is no longer employed by HNP at the time audit report is issued.
B. HNP will comply with subparagraph 4.4.3 clarified to read: "Supervisory level personnel with whom significant discussions were held during the course of preaudit (where conducted), audit, and postaudit (where conducted) activities.
C. Audit reports may not necessarily contain an evaluation statement regarding the effectiveness of the quality assurance program elements which were audited, as required by subparagraph 4.4.4, but they will provide an effectiveness summary of the audited areas."
D. Subparagraph 4.4.6 - Nuclear Oversight section management will determine the need for audit reports to include recommendations for corrective actions.
E. HNP will comply with the last paragraph of Section 4.4 of ANSI N45.2.12 concerning issuing audit reports with the following clarification: "Audit reports shall be issued within thirty working days after the last day of the audit. The last day of an audit shall be considered to be the day of the post-audit conference. If a post-audit conference is not held because it was deemed unnecessary, the last day of the audit shall be considered to be the date the post-audit conference was deemed unnecessary as documented in the audit report."

10. Paragraph 4.5.1, By Audited Organization: HNP will comply with the following clarification of this paragraph:
    "Management of the audited organization or activity shall review and investigate all adverse audit findings, as necessary, (cause, etc.) to determine and schedule appropriate corrective action including action to prevent recurrence. They shall respond, in writing, within thirty days after the date of receipt of the audit report. The response shall clearly state the corrective action taken or planned to prevent recurrence and the results of the investigation if conducted. In the event that corrective action is not completed by the time the response is submitted, the audited organization's response shall include a scheduled date for completion of planned corrective action. A follow-up response shall be provided stating the corrective action was completed.
    If corrective actions are verified as satisfactorily completed by the quality organization prior to the scheduled completion date or when completion of corrective action can be verified during a follow-up audit, no follow-up response is required. The audited organization shall take appropriate action to assure that corrective action is accomplished as scheduled."
11. Paragraph 5 - audit checklists are not considered QA records. HNP believes that actual audit reports provide sufficient detail to substantiate the results of the audit, and the checklist is maintained as an audit "tool" versus a QA record. Additionally, the audit checklist need only document objective evidence examined to support the audit findings.

Table B17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.146, Qualification of QA Program Audit Personnel for Nuclear Power Plants (Rev. 0, 8/80)
HNP shall comply with requirements of Regulatory Guide 1.146, August 1980, which endorses ANSI N45.2.23-1978, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants with the following clarifications.

1. Paragraph 2.2, Qualification of Auditors: subparagraph 2.2.1 references an "ANSI B45.2" (presumed to be N45.2); therefore, HNP will comply with an alternate subparagraph 2.2.1 which reads:
   "Orientation to provide working knowledge and understanding of the HNP QA program, including the ANSI standards and Regulatory Guides included in the program, and Duke Energy's procedures for implementing audits and reporting results."
2. Paragraph 4.1, Organization Responsibility: HNP will comply with this paragraph with the substitution of the following sentence in place of the last sentence in the paragraph.
   "The NOS manager or the audit team leader shall, prior to commencing the audit, assign personnel who collectively have experience or training commensurate with the scope, complexity, or special nature of the activities to be audited."
3. Paragraph 5.3, Updating of Lead Auditor's Records: HNP will substitute the following sentence for this paragraph:
   "Records for each lead auditor shall be maintained and updated during the period of the annual management assessment. This annual management assessment shall be as defined in the clarification for Paragraph 3.2 noted above."
4. ANSI N45.2.23, Paragraph 2.3.4 states, "The prospective lead auditor shall have participated in a minimum of five (5) quality assurance audits within a period of time not to exceed three (3) years prior to the date of qualification, one audit of which shall be a nuclear quality assurance audit within the year prior to qualification."
   HNP substitutes the following instead of the cited sentence of ANSI N45.2.23, Paragraph 2.3.4:
   "Prospective lead auditors shall demonstrate the ability to effectively implement the audit process and effectively lead an audit team. This process is described in written procedures that provide for evaluation and documentation of the results of this demonstration. In addition, the prospective lead auditor shall have participated in at least two nuclear quality assurance audits within the year preceding the individual's effective date of qualification. Upon successful demonstration of the ability to effectively implement the audit process and effectively lead audits, and having met the other provisions of Section 2.3 of ANSI N45.2.23-1978, the individual may be certified as being qualified to lead audits."

Table B17-2. Site Specific Response to Regulatory Guides and Industry Standards

Table B17-2 identifies additional Regulatory Guides addressing subjects related to implementation of the QAP but the implementation is site specific and controlled with the UFSAR in accordance with 10 CFR 50.59.

Regulatory Guide 1.8, Personnel Selection and Training
Personnel selection and training is site specific. Harris addresses conformance with Regulatory Guide 1.8 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
Quality group classifications and standards trace to the original design and construction of the nuclear power plant and therefore are site specific. Harris addresses conformance with Regulatory Guide 1.26 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.29, Seismic Design Classification
Seismic design classification trace to the original design and construction of the nuclear power plant and therefore is site specific. Harris addresses conformance with Regulatory Guide 1.29 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel
Nonmetallic thermal insulation for austenitic stainless steel trace to the original design and construction of the nuclear power plant and therefore is site specific. Harris addresses conformance with Regulatory Guide 1.36 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
Quality assurance requirements for protective coatings applied to water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Harris addresses conformance with Regulatory Guide 1.54 in UFSAR Chapter 1 Section 8.

Table B17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Harris addresses conformance with Regulatory Guide 1.143 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.155, Station Blackout
Addressing Station Blackout is site specific. Harris addresses conformance with Regulatory Guide 1.155 in UFSAR Chapter 1 Section 8.

Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment
Quality assurance for radiological monitoring program (normal operations) – effluent streams and the environment is site specific. Harris does not address conformance to Regulatory Guide 4.15 in UFSAR Chapter 1 Section 8. The radiological monitoring program is addressed in UFSAR Chapter 11.

B17.3.1 MANAGEMENT

B17.3.1.1 Methodology

There are no Harris specific amplifications for this section.

B17.3.1.2 Organization

There are no Harris specific amplifications for this section.

B17.3.1.3 Responsibility

There are no Harris specific amplifications for this section.

B17.3.1.4 Authority

The program and procedures require that the authority and duties of persons and organizations performing activities affecting quality be clearly established and delineated in writing and that these individuals and organizations have sufficient authority and organizational freedom to:

1. Identify quality, nuclear safety, and performance problems.
2. Order unsatisfactory work to be stopped and control further processing, delivery, or installation of nonconforming material.
3. Initiate, recommend, or provide solutions for conditions adverse to quality.
4. Verify implementation of solutions.

B17.3.1.5 Personnel Training and Qualification

There are no Harris specific amplifications for this section.

B17.3.1.6 Corrective Action

The program requires that an evaluation of adverse conditions such as conditions adverse to quality, nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment is conducted to determine need for corrective action. Conditions adverse to quality are identified through inspections, assessments, tests, checks, and review of documents.

The program requires corrective action to be initiated to preclude recurrence of significant conditions adverse to quality.

For significant conditions adverse to quality, procedures require follow-up reviews, verifications, inspections, etc., to be conducted to verify proper implementation of corrective action and to close out the corrective action documentation.

The program outlines the methodology for resolution of disputes involving quality and nuclear safety issues arising from a difference of opinion between identifying personnel and other groups.

Significant conditions adverse to quality are reported to appropriate management for review and evaluation.

Periodic review and evaluation of adverse trends are performed by management.

B17.3.1.7 Regulatory Commitments

There are no Harris specific amplifications for this section.

B17.3.2 PERFORMANCE/VERIFICATION

B17.3.2.1 Methodology

There are no Harris specific amplifications for this section.

B17.3.2.2 Design Control

Controls are applied to the development, content and use of computer codes to ensure (1) the codes are developed, documented, verified and certified for use per approved procedures; (2) the codes are properly controlled to preclude use of outdated or obsolete codes; (3) that proper instructions concerning the use of the codes are provided; and (4) adequate QA provisions are implemented for the procurement of computer codes.

B17.3.2.3 Design Verification

There are no Harris specific amplifications for this section.

B17.3.2.4 Procurement Control

Potential contractors and suppliers are evaluated prior to award of a procurement contract when needed to assure the contractor's or supplier's capability to comply with applicable technical and quality requirements.

Procurement documents, such as purchase specifications, contain or reference the following:

1. Technical, administrative, regulatory, and reporting requirements, including material and component identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions.
2. Identification of the documentation to be prepared, maintained, or submitted (as applicable) to HNP for review and approval. These documents may include, as necessary, inspection and test records, qualification records, or code required documentation.
3. Identification of those records to be retained, controlled, and maintained by the supplier, and those delivered to the purchaser prior to use or installation of the hardware.

Procurement documents require suppliers to operate in accordance with QA programs which are compatible with the applicable requirements of the HNP QA Program and procedures where their services are utilized in support of plant activities.

B17.3.2.5 Procurement Verification

There are no Harris specific amplifications for this section.

B17.3.2.6 Identification and Control of Items

Procedures require that materials, parts, and components be identified and controlled to prevent the use of incorrect or defective items.

These procedures also require that identification of items be maintained either on the item in a manner that does not affect the function or quality of the item, or on records traceable to the item.

Procedures implementing these requirements provide for the following:

1. Verification that items received at the plant are properly identified and can be traced to the appropriate documentation, such as drawings, specifications, purchase orders, manufacturing and inspection documents, nonconformance reports, or material test reports.
2. Verification of item identification consistent with the HNP inventory control system and traceable to documentation which identifies the proper uses or applications of the item.
3. Verification of correct identification of material, parts and components prior to fabrication, assembly installation or use, and results documented.

B17.3.2.7 Handling, Storage, and Shipping

Provisions are established to control the shelf life and storage of chemicals, reagents, lubricants, and other consumable materials.

B17.3.2.8 Test Control

Test procedures incorporate or reference the following, as required:

1. Instructions and prerequisites for performing the test.
2. Use of proper test equipment.
3. Mandatory inspection hold points.
4. Acceptance criteria.

Test results are documented, evaluated, and their acceptability determined by a qualified, responsible individual or group.

When the acceptance criteria is not met, affected areas are to be retested or evaluated, as appropriate.

B17.3.2.9 Measuring and Test Equipment Control

Portable measuring and test equipment is calibrated by standards which are at least four times as accurate as the portable measuring and test equipment, unless limited by the state of the art. In cases where the accuracy is not achievable or is limited by the state of the art, an engineering evaluation or other appropriate justification is performed and documented to justify acceptability of the M&TE in question. The evaluation is reviewed in accordance with approved procedures.

Calibration of installed plant devices shall be against M&TE having sufficient accuracy, greater than the device being calibrated, to assure that the system containing the device is within the specified system tolerance. The basis for determining the "greater than accuracy" shall be documented.

Reference and transfer standards are traceable to nationally recognized standards; or where national standards do not exist, provisions are established to document the basis for the calibration.

B17.3.2.10 Inspection, Test, and Operating Status

These procedures include the application, removal, and verification of inspection and welding stamps, or other status indicators as appropriate.

Altering the sequence of required tests, inspections, and other operations important to safety can only be accomplished by methods outlined in procedures.

B17.3.2.11 Special Process Control

There are no Harris specific amplifications for this section.

B17.3.2.12 Inspection

There are no Harris specific amplifications for this section.

B17.3.2.13 Corrective Action

The primary goal of the corrective action program is to improve overall plant operations and performance by identifying and correcting root causes of equipment and human performance problems.

Procedures define requirements for a corrective action program that charges personnel working at or supporting the nuclear plants with the responsibility to identify adverse conditions (including conditions adverse to quality).

Procedures include requirements for verification of the acceptability of the rework/repair of items by reinspection and/or testing in accordance with the original inspection or test requirements or by an accepted alternative inspection and testing method.

Conditions that require rework/repairs are identified through the use of maintenance work request forms.

B17.3.2.14 Control of Documents

Changes to documents are reviewed and approved by the same organization that performed the original review and approval or by other designated qualified responsible organizations.

B17.3.2.15 Records

The structure in which single copy records are maintained is designed to prevent destruction, deterioration, or theft. This structure ensures protection against destruction by fire, flooding, theft, and deterioration by the environmental conditions of temperature and humidity.

B17.3.3 ASSESSMENT

B17.3.3.1 Methodology

There are no Harris specific amplifications for this section.

B17.3.3.2 Independent Review

There are no Harris specific amplifications for this section.

B17.3.3.3 Independent Assessment

There are no Harris specific amplifications for this section.

B17.3.3.3.1 Organization

There are no Harris specific amplifications for this section.

B17.3.3.3.2 Internal Assessment Process

There are no Harris specific amplifications for this section.

B17.3.3.3.3. Internal Audit Program

B17.3.3.3.3.1 Other Reviews Prescribed by the Code of Federal Regulations

There are no Harris specific amplifications for this section.

B17.3.3.3.3.2 Independent Audit of Fire Protection Program

There are no Harris specific amplifications for this section.

B17.3.3.3.4 Results

There are no Harris specific amplifications for this section.

B17.3.3.3.5 Supplier Oversight

There are no Harris specific amplifications for this section.

B17.3.3.3.6 Independent Audit of QA Functions

There are no Harris specific amplifications for this section.

B17.3.3.3.7 Audit Frequency Extensions

There are no Harris specific amplifications for this section.

B17.3.4 ADMINISTRATIVE CONTROLS

This section was added to the HNP UFSAR description of the QA Program to relocate certain administrative controls from HNP Technical Specifications. These relocated administrative controls include Review and Audit, Procedure Review Requirements, and Record Retention.

Review and Audit

B17.3.4.1 10CFR50.59 and technical reviews

There are no Harris specific amplifications for this section.

B17.3.4.2 Plant Nuclear Safety Committee (PNSC)

This content is addressed in Section 17.3.3.2, Independent Review.

B17.3.4.3 HNP Independent Review Program

This content is addressed in Section 17.3.3.2, Independent Review.

B17.3.4.4 Independent Safety Engineering Group

B17.3.4.4.1 Organization

The Independent Safety Engineering Group (ISEG) functions of improving licensee safety performance and ability to respond to accidents by providing onsite technical support and continuous evaluation and feedback of lessons learned from operating experience are performed by a combination of different groups through the performance of their normal activities.

B17.3.4.4.2 Activities

Key ISEG activities are outlined below with the groups that currently perform these activities:

1. Examination of Unit Operating Characteristics:
    - HNP has an established Corrective Action Program that includes processes for the identification, classification, trending and correcting of conditions adverse to quality.
    - NOS performs independent monitoring and audit of activities as defined in Section 17.3.3.3.
    - HNP has implemented a Maintenance Rule Program that provides reasonable assurance that structures, systems, trains, and components are capable of fulfilling their intended safety significant functions.
    - Harris Engineering Section has implemented a program that provides for the systematic trending of system and component performance to determine the effectiveness of system/component maintenance
    - A corporate Probabilistic Safety Assessment Unit has been established with the mission of maintaining and updating plant specific risk models and risk based tools that are used to provide risk insights and tools to: support on-line maintenance and outage risk assessments; support the Maintenance Rule Program; evaluate proposed plant changes for risk impact; monitor the risk effectiveness of plant on-line maintenance activities; and support other regulatory activities.
2. Examination of NRC Issuances, Industry Advisories, and Licensee Event Reports and other Sources of Unit Design Information which May Indicate Areas of Improving Unit Safety:
    - Duke Energy has implemented an Operating Experience (OE) Program that provides for the receipt, processing, status reporting, screening, reviewing, evaluating, and taking preventive/corrective actions in response to OE information.
    - The Nuclear Oversight organization independently evaluates the use of OE in the conduct of audits.
    - The On-Site Review Committee reviews License Event Reports developed pursuant to 10CFR50.73 as part of the Independent Review in Section 17.3.3.2.
3. Review of Plant Operations, Modifications, Maintenance, and Surveillances to Verify Independently that these Activities are Performed Safely and Correctly and that Human Errors are Reduced as Much as Practical:
    - NOS audits in Section 17.3.3.3 and the Independent Review Program in Section 17.3.3.2accomplish this function.

B17.3.4.5 Outside agency inspection and audit program

The fire protection audit is addressed in Section 17.3.3.3.3.2, Independent Audit of Fire Protection Program.

B17.3.4.6 Procedure Review Requirements

See Section 17.3.2.14 for required reviews for changes to procedures, tests, and experiments.

B17.3.4.7 Record Retention

A list of typical operational phase QA Records is included in 17.3.2.15.

Attachment C, Robinson Specific QAPD

Information presented in this attachment is specific to Robinson and was contained in the UFSAR prior to Amendment 41.

Where a section contains no descriptive information beyond that in the generic text in the body of the document, a statement is made to that effect and no content is included. See C17.3.1.2, Organization for example.

C17. QUALITY ASSURANCE

C17.1 QA DURING DESIGN AND CONSTRUCTION

See Robinson UFSAR Chapter 17 for historic information from the description of the QA Program for design and construction.

C17.2 OPERATIONAL QA

Deleted

(NOTE: In April 1995, NRC approved the reformatting of the description of the Robinson QA Program to follow Standard Revision Plan Section 17.3, replacing the content of 17.2.)

C17.3 QUALITY ASSURANCE PROGRAM (QAP) DESCRIPTION

INTRODUCTION

This content is not addressed in SRP Section 17.3; therefore, the Robinson description of the QA Program did not include this section.

DEFINITIONS

There are no Robinson specific definitions.

EXPLANATION OF "QUALITY ASSURANCE"

There is no Robinson specific content.

QA STANDARDS AND GUIDES

Table C17-1 and C17-2 address QAP conformance to the referenced regulatory and program guidance in NUREG-0800 Section 17.3.

The content of Table C17-1 was transferred from Section 1.8 of the Robinson UFSAR. Changes to the content of Table C17-1 are controlled in accordance with 10 CFR 50.54(a). Subsequent changes to the QAP are incorporated in this document as identified in Section 17.3.1.7.

Table C17-2 addresses additional Regulatory Guides that relate to implementation of the QAP but the implementation is site specific and controlled with the Robinson UFSAR in accordance with 10 CFR 50.59.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards

Generic Exception:

Table C17-1 addresses Robinson's Conformance of the Quality Assurance Program to certain NRC Regulatory Guides. In so doing, specific editions of industry standards are identified for compliance with exceptions and alternatives. Those identified standards include references to other industry standards for activities including, but not limited to; design, fabrication, inspection, and testing. Those included reference industry standards are considered to be guidance documents for details of how activities may be accomplished. The actual standard to be used in such cases is controlled by each station's current licensing and design bases.

The content of Table C17-1 was transferred from H. B. Robinson (RNP) UFSAR Section 1.8. As identified therein, Regulatory Guides (originally called Safety Guides) have been published beginning in late 1970. Since H. B. Robinson (RNP) was licensed for operation prior to that time, they were not addressed. Applicable QA Regulatory Guides which have been addressed during the operating phase are discussed below.

Regulatory Guide 1.28, Quality Assurance Program Requirements (Design and Construction) (Rev. 0)
ANSI Standard N45.2-1971, Quality Assurance Requirements for Nuclear Power Plants
This guide and the standard it endorses have been superseded for operations activities by Regulatory Guide 1.33 and ANSI N18.7-1976 which it endorses. The Operational Quality Assurance Program complies with Regulatory Guide 1.33 and ANSI N18.7-1976 as stipulated in Appendix A to that program; therefore, Regulatory Guide 1.28 (Safety Guide 28) and ANSI N45.2-1971 which it endorses are not considered necessary and are not included as part of the program.

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment (Revision 0) (August, 1972)
ANSI standard N45.2.4-1972, (IEEE-336-1971), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations
RNP shall comply with the provisions of Regulatory Guide 1.30, August 1972 and ANSI N45.2.4-1972 with the following exceptions:
The installation, inspection, and testing of nuclear power plant instrumentation and electrical equipment at RNP will be in accordance with the applicable requirements of ANSI N45.2.4-1972 with the following exceptions:
a) Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP's commitment to Regulatory Guide 1.74. b) Section 1.5 titled Referenced Documents: RNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. c) Section 2.5 titled Measuring and Test Equipment: RNP will implement the applicable portions of this Section as follows: The status of portable items of measuring and test equipment and reference standard shall be identified by use of tags, stickers, labels, routing cards, computer programs, or other suitable means for the date recalibration is due or the frequency of recalibration. These items are in a calibration program which requires recalibration on a specified frequency or, in certain cases, prior to use.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.30, Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment (Revision 0) (August, 1972)
ANSI standard N45.2.4-1972, (IEEE-336-1971), Installation, Inspection, and Testing Requirements for Instrumentation and Electrical Equipment During the Construction of Nuclear Power Generating Stations
RNP shall comply with the provisions of Regulatory Guide 1.30, August 1972 and ANSI N45.2.4-1972 with the following exceptions:

Instrumentation and electrical equipment in the categories listed below shall be in a calibration program. This program provides, by the use of status cards, computer schedules, or tags, for the date that recalibration is due and indicates the status of calibration. The identity of person(s), performing calibration is provided on the calibration documents.

1) Instruments installed as listed in the RNP Technical Specifications
2) Installed instrumentation used to verify RNP Technical Specification parameters, and
3) Installed safety-related instruments and electrical equipment that provide an active function during operation or during shutdown; i.e., not a device being designated safety-related solely because the instrument is an integral part of a pressure retaining boundary.

d) Section 7 titled Data Analysis and Evaluation states in part, "Procedures shall be established for processing inspection and test data and their analysis and evaluation." At RNP, data processing procedures per se have not been developed; instead, test data are recorded, processed, and analyzed in accordance with procedures and instructions in appropriate functional areas; e.g., maintenance, startup.

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation) Revision 2, February 1978
ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance Requirements for the Operational Phase of Nuclear Power Plants
Comply with the provisions of Regulatory Guide 1.33, Rev. 2 February 1978, and the requirements and recommendations for administrative controls described in ANSI N18.7-1976, except as stated below:

1. Exception to Paragraph C.3 of Regulatory Guide 1.33 and ANSI N18.7-1976 Paragraph 4.3: Independent Review Program requirements are replaced by Section 17.3.3.2, Independent Review. This exception uses NRC Safety Evaluation dated January 13, 2005 to Nuclear Management Company (ADAMS ML050210276).
2. In lieu of the audit program provisions contained in Regulatory Position C.4 of Regulatory Guide 1.33, audits of facility activities will be conducted in accordance with Section 17.3.3.3.3.
3. Paragraph 4.5 - Written audit reports are not formally reviewed as part of the Independent Review function.
4. Paragraph 4.5 - The CNO will assure that an independent assessment of the overall Nuclear Oversight program is conducted at least once every 24 months. Results of the independent assessment will be reported directly to the CNO and entered into the Corrective Action Program for resolution. (For scheduling consistency, the exceptions included in paragraph 5 of this section will be used as clarification for scheduling this independent assessment).
5. Paragraph 4.5, Audit Program- ANSI N18.7-1976/ANS-3.2, Section 4.5 is implemented with the following clarification: The audits of selected aspects of operational phase activities as identified in Section 17.3.3.3.3, Internal Audit Program, are scheduled based on plant performance and importance to safety but at a frequency not to exceed twenty-four months with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation) Revision 2, February 1978
ANSI Standard N18.7-1976, Administrative Controls and Quality Assurance Requirements for the Operational Phase of Nuclear Power Plants
Comply with the provisions of Regulatory Guide 1.33, Rev. 2 February 1978, and the requirements and recommendations for administrative controls described in ANSI N18.7-1976, except as stated below:
6. Section 5.2.16 titled Measuring and Test Equipment: See Section 17.3.2.9 for clarification. 7. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Rev. 2, February 1978, shall be established, implemented, and maintained as specified in the RNP Technical Specifications. 8. Section 5.2.17 titled Inspections: The second to the last sentence in the last paragraph, "Deviations, their cause, and any," to be consistent with Paragraph 5.2.11 and 10CFR50, Appendix B, the cause of the deviation will be determined for only significant conditions adverse to safety. 9. Section 5.3.9.1 titled Emergency Procedure Format and Content: Emergency procedures shall be in the format as committed to in NUREG-0737, TMI Action Plan. 10. Section 5.2.2 titled Procedure Adherence: Temporary changes to approved procedures, tests, or experiments may be approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator License if such change does not change the intent of the original procedure, test, or experiment. Temporary changes shall be documented and approved as a permanent change or deleted within 21 days of receiving temporary approval. 11. Section 5.2.15 titled Review, Approval and Control of Procedures, states that, "Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary. A revision to a procedure constitutes a procedure review." In lieu of this commitment, Duke Energy addresses programmatic controls in Section 17.3.2.14 to continually identify procedure revisions which may be needed to ensure that procedures are appropriate for the circumstance and are maintained current. 12. Paragraph 5.2.13.1, Procurement Document Control: When purchasing commercial-grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a quality assurance program consistent with ANSI N45.2-1971. Alternate requirements described in Section 1.8 for Regulatory Guide 1.123 may be implemented in lieu of imposing a quality assurance program consistent with ANSI N45.2-1971.
Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
Those areas of the QA Program applicable to onsite cleaning of materials and components, cleanliness control, and pre-operation cleaning and layup of RNP fluid systems, will be in accordance with ANSI N45.2.1-1973, with the following exceptions:
a) At RNP a classification system similar to ANSI N45.2.1-1973 has been developed and is fully implemented for cleaning of fluid systems. b) Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.1-1973, Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
Those areas of the QA Program applicable to onsite cleaning of materials and components, cleanliness control, and pre-operation cleaning and layup of RNP fluid systems, will be in accordance with ANSI N45.2.1-1973, with the following exceptions:
c) Section 1.5 titled Referenced Documents: RNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document.
Regulatory Guide 1.38, Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.2-1972, Packing, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants
Packaging, shipping, receiving, storage, and handling of RNP items are in accordance with applicable requirements of ANSI N45.2.2-1972 with the following specific exceptions:
a) Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74. b) Section 1.5 titled Referenced Documents: RNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. c) Section 2.7 titled Classification of Items and Section 6.1.2 titled Levels of Storage: 1) Special electronic equipment and instrumentation received as assembled panels will be stored as recommended by the manufacturer and/or based on engineering evaluation to prevent damage, deterioration, or contamination, but not necessarily in a Level A storage area. 2) Chemicals used at RNP are stored at the point of use and/or in warehouse areas that satisfy the requirement of Level B storage. These storage areas have been evaluated and determined to be adequate for the limitations established by the manufacturer. 3) Special nuclear materials are stored in areas specifically designed for such storage. d) Section 7.3.4 - RNP intends to comply with the requirements of this section with the following clarification: Test loads equal to or greater than the original crane rating shall not pass over locations where special nuclear material is stored or where reactor system components or high cost equipment are located. e) Section 6.4.2 of ANSI N45.2.2 - 1972, titled Care, sub-items (5), (6), and (7) are clarified as follows: 1) Sub-item (5), space heaters in electrical equipment shall be energized, unless a documented engineering evaluation determines that such space heaters are not required. 2) Sub-item (6). large rotating electrical equipment (i.e. greater than or equal to 50 horsepower) shall be given insulation resistance tests on a scheduled basis, unless a documented engineering evaluation determines such tests are not needed. 3) Sub-item (7). prior to being placed in storage, rotating equipment weighing over approximately 50 lbs. shall be evaluated and documented by engineering personnel to determine if shaft rotation during storage is required. If rotation is required the degree of turn shall be such that the parts receive lubrication where applicable and the shaft does not come to rest in a previous position. Required rotation shall be performed at the necessary intervals and documented.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.39, Housekeeping Requirements for Water- Cooled Nuclear Power Plants (March 1973)
ANSI Standard N45.2.3-1973, Housekeeping, During the Construction Phase of Nuclear Power Plants
The applicable requirements of ANSI N45.2.3-1973 are followed at Robinson 2 within the context of the established QA Program with the following specific exception -- the zone designations of Section 2.1 of ANSI N45.2.3 and the requirements associated with each zone are considered impractical for implementation, as stated, at Robinson 2 during the operations phase. Instead, procedures or instruction for housekeeping activities, which include the applicable requirements outlined in Section 2.1 of ANSI N45.2.3 and which take into account radiation control considerations, security considerations, and cleanliness requirements are developed on a case basis for work to be performed.
Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (September, 1980)
ANSI Standard N45.2.6-1978, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
RNP shall comply with NRC Regulatory Guide 1.58, September 1980 which endorses ANSI N45.2.6-1978, with the following exceptions: 1. Section 1.2 titled Applicability: RNP elects not to apply the requirements of this guide to those personnel who are involved in the daily operations of surveillance, maintenance, and certain technical and support services whose qualifications are controlled by the RNP Technical Specifications or are controlled by other QA Program commitment requirements. Only personnel in the following listed categories will be required to meet ANSI N45.2.6-1978 requirements: a. Nondestructive examination (NDE) personnel b. QC inspection personnel c. Receipt inspection personnel 2. The fourth paragraph of Section 1.2 requires that the Standard be imposed on personnel other than RNP employees. The applicability of the Standard to suppliers and contractors will be documented and applied, as appropriate, in the procurement documents for such suppliers and contractors or in interface agreements for Duke Energy non-nuclear organizations providing services identified in Section 17.3.1.2.3. 3. Section 1.4 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74. 4. Section 2.5 titled Physical: RNP will implement the requirements of this Section with the stipulation that, where no special physical characteristics are required, none will be specified. The converse is also true: if no special physical requirements are stipulated by RNP, none are considered necessary. RNP employees receive an initial physical examination to assure satisfactory physical condition; however, only the following listed personnel will receive an annual (± 2 months) examination: a. NDE personnel b. QC inspection personnel c. Receipt inspection personnel This annual examination shall consist of the near visual acuity using the standard Jaeger's type chart or equivalent test.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (September, 1980)
ANSI Standard N45.2.6-1978, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
RNP shall comply with NRC Regulatory Guide 1.58, September 1980 which endorses ANSI N45.2.6-1978, with the following exceptions: 5. Section 3 titled Qualifications: Only personnel performing NDE (such as LP, MT, UT, & RT) are required to be grouped in levels of capability and certified as such. Personnel performing inspection will be certified for inspection, review and evaluation of inspection data, and reporting of inspection and test results. 6. Section 3.5 titled Education & Experience Recommendations: RNP will certify individual inspectors through training and experience to requirements appropriate to the specific assignment; however, except for NDE, personnel are not required to be classified by levels of capability. Inspection personnel may be qualified based on pre-established experience, education, on-the-job training, written examinations and proficiency tests associated with the specific activity. Proficiency tests are given to personnel performing independent QC inspections and documented acceptance criteria are developed to determine if individuals are properly trained and qualified. Certificates of qualification delineate the functions personnel are qualified to perform. Qualification records are maintained and performance evaluations conducted at least once every three years. If organizations elect to utilize qualifications by levels for non-NDE inspections, Level I inspectors receive a minimum of 4 months experience as Level I before being certified as Level II, in lieu of one year experience recommended by ANSI N45.2.6 Section 3.5.2(1). Organizations identify in their procedures if they qualify their inspectors by Level or by task qualifications. Inspectors are only assigned functions for which they have been qualified.
Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants (October 1973)
ANSI Standard N45.2.11-1974, Quality Assurance Requirements for the Design of Nuclear Power Plants
Those areas of the QA Program for RNP applicable to design or modification of the plant are in accordance with the applicable guidance of ANSI N45.2.11-1974, with the following exception: a) Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74.
Regulatory Guide 1.74, Quality Assurance Terms and Definitions (February, 1974)
ANSI Standard N45.2.10-1973, Quality Assurance Terms and Definitions
The quality assurance terms and definitions of ANSI N45.2.10-1973 and Regulatory Guide 1.74 are being complied with for use in describing and implementing the Robinson 2 QA Program.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.88 , Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants
ANSI Standard N45.2.9-1979 , "Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants"
As documented in RNP Letter to the NRC dated March 23, 1993, RNP is no longer committed to Regulatory Guide 1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records," August 1974.
See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records. The requirements for collection, storage, and maintenance of QA records at RNP will be in accordance with ANSI N45.2.9-1979 and Section 17.3.2.15, subject to the following: 1. Section 1.5 titled Referenced Documents: RNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. 2. Section 5.4 Item 2 "Records shall be firmly attached in binders or placed in folders or envelopes for storage in steel file cabinets or on shelving in containers." RNP complies with this requirement except for periods when records are in the receipt process. 3. Section 5.6 states: "Records shall be stored in facilities constructed and maintained in a manner which minimizes the risk of damage or destruction from the following: a. Natural disasters such as winds, floods, or fires. b. Environmental conditions such as high and low temperatures and humidity. c. Infestation of insects, mold, or rodents." Records are stored in permanent and temporary facilities as follows: 1) One hour UL-rated fireproof file cabinets are utilized for temporary storage of hardcopy records. These file cabinets are located at work locations throughout the plant and will contain the records until the records are transmitted to the appropriate Document Control Center. Records being processed in Document Control Centers will be stored in fireproof cabinets when they are not being processed and until they are sent to the vault. In addition, records that are generated and authenticated electronically are afforded protection as described in N45.2.9-(1979) prior to conversion to permanent storage media. 2) Permanent storage of QA records will be in the plant vault constructed to meet the requirements of this ANSI standard, and via electronic means which also meet applicable provisions of this standard, in addition to those delineated below. 3) Selected records may be stored off-site by a QA Records Storage supplier provided that supplier meets the applicable sections of this ANSI standard. 4. Section 6.2 states: "Storage systems shall provide for retrieval of information in accordance with planned retrieval times based upon the record type." Retrieval of records at RNP is via a random access computer system using key words and document identification numbers, or through a manual index for records completed prior to 1982. The manual system is keyed to Plant Systems. 5. Section 7.3.3 states: "Various regulatory agencies have requirements concerning records that are within the scope of this Standard. The most stringent requirements shall be used in determining the retention period." 6. RNP will continue to adhere to the recommendations of Appendix A of ANSI N45.2.9-1974, or with the most stringent requirement with respect to records retention.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.94, Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (April 1976)
ANSI Standard N45.2.5-1974, Supplementary Quality Assurance Requirements for Installation, Inspections, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
The original specification requirements, applicable guidance contained in Regulatory Guide 1.94, or acceptable alternatives based on an engineering evaluation will be utilized in the event future structural work is to be performed which falls under the established requirements of the Robinson 2 QA Program. Future field production welding acceptance criteria will be based on NCIG-01, "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants," Revision 2, dated May 7, 1985, Prepared by the Nuclear Construction Issues Group (NCIG) for structural safety-related and non-safety related pipe, conduit, cable tray, duct, and equipment supports where welding is specified to be in accordance with AWSD1.1. This will be implemented through appropriate RNP specifications.
Regulatory Guide 1.116, QA Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (June, 1976)
ANSI Standard N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants
Regulatory Guide 1.116, June, 1976, endorses ANSI N45.2.8-1975. RNP does not commit to Regulatory Guide 1.116 but does endorse parts of ANSI N45.2.8-1975 as described below.
Within the context of the established QA Program, the applicable guidance contained in ANSI N45.2.8-1975 will be utilized in relation to mechanical maintenance or modification with the following exceptions: a) Section 1.4 titled Definitions: Definitions in this standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74. b) Section 1.5 titled Referenced Documents: RNP's commitment to other documents referenced in this standard shall be as stated in our commitment to that document. c) Section 2.8 titled Measuring and Test Equipment: RNP will implement the applicable portions of this section as follows: The status of portable items of measuring and test equipment and reference standards shall be identified by use of tags, stickers, labels, routing cards, computer programs, or other suitable means for the date recalibration is due or the frequency of recalibration. These items are in a calibration program which requires recalibration on a specified frequency or, in certain cases, prior to use. Instrumentation and electrical equipment in the categories listed below shall be in a calibration program. This program provides, by the use of status cards, computer schedules, or tags, for the date that recalibration is due and indicates the status of calibration. The identity of person(s) performing the calibration is provided on the calibration documents. 1) Instruments installed as listed in the RNP Technical Specifications, 2) Installed instrumentation used to verify RNP Technical Specification parameters, and 3) Installed safety-related instruments and electrical equipment that provide an active function during operation or during shutdown; i.e., not a device being designated safety-related solely because the instrument is an integral part of a pressure retaining boundary.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.116, QA Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (June, 1976)
ANSI Standard N45.2.8-1975, Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants
Regulatory Guide 1.116, June, 1976, endorses ANSI N45.2.8-1975. RNP does not commit to Regulatory Guide 1.116 but does endorse parts of ANSI N45.2.8-1975 as described below.
Within the context of the established QA Program, the applicable guidance contained in ANSI N45.2.8-1975 will be utilized in relation to mechanical maintenance or modification with the following exceptions:
d) Section 6 titled Data Analysis and Evaluation states in part, "Procedures shall be established for processing inspection and test data and their analysis and evaluation." At H. B. Robinson 2 data processing procedures per se have not been developed; instead, test data are recorded, processed, and analyzed in accordance with procedures and instructions in appropriate functional areas; e.g., maintenance, startup.
Regulatory Guide 1.123, Quality Assurance Requirements for Control or Procurement of Items and Services for Nuclear Power Plants (July, 1977)
ANSI Standard N45.2.13, Quality Assurance Requirements for (Draft 2, Rev. 4, April, 1974) Control or Procurement of Items and Services for Nuclear Power Plants
RNP does not commit to Regulatory Guide 1.123; however, the applicable guidance contained in ANSI N45.2.13-1974, Draft 2, Rev. 4, and ANSI N18.7-1976 will be utilized in relation to procurement of items and services performed under the established requirements of the RNP QA Program.
See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.
Regulatory Guide 1.144, Auditing of Quality Assurance (January 1979)
ANSI Standard N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
RNP will follow the requirements and recommendations of Regulatory Guide 1.144 and ANSI N45.2.12 with the following clarifications:
1. RNP will follow the requirements and recommendations of Regulatory Guide 1.144 paragraphs C.1, C.2, C.3.a.2, C.3.b, and C.4. Our position on paragraph C.3.a.1 is as follows: Audits of operational phase activities, as outlined in Section 17.3.3.3.3, shall be performed at the frequencies specified therein. See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories. 2. RNP will comply with the last paragraph of Section 4.4 of ANSI N45.2.12 concerning issuing audit reports with the following clarification: "Audit reports shall be issued within thirty working days after the last day of the audit. The last day of an audit shall be considered to be the day of the post-audit conference. If a post-audit conference is not held because it was deemed unnecessary, the last day of the audit shall be considered to be the date the post-audit conference was deemed unnecessary as documented in the audit report."

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.144, Auditing of Quality Assurance (January 1979)
ANSI Standard N45.2.12-1977, Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
RNP will follow the requirements and recommendations of Regulatory Guide 1.144 and ANSI N45.2.12 with the following clarifications:

3. ANSI N45.2.12 Paragraph 4.3. 1, Preaudit Conference: RNP will comply with the requirement of this paragraph by inserting the word "Normally" at the beginning of the first sentence. This clarification is required because, in the case of certain unannounced audits or audits of a particular operation or work activity, a preaudit conference might interfere with the spontaneity of the operation or activity being audited. In other cases, persons who should be present at a preaudit conference may not always be available. Such lack of availability should not be an impediment to beginning an audit. Even in the above examples, which are not intended to be all inclusive, the material set forth in Paragraph 4.3.1 will normally be covered during the course of the audit.
4. ANSI N45.2.12 Paragraph 4.3.3, Post Audit Conference: RNP will substitute and comply with the following paragraphs: "For all external audits, a post audit conference shall be held with management of the audited organization to present audit findings and clarify misunderstandings.
   Where no adverse findings exist, this conference may be waived by management of the audited organization. Such waiver shall be documented in the audit report. For all internal audits, unless unusual operating or maintenance conditions preclude attendance by appropriate managers/supervisors, an audit exit shall be held with managers/supervisors. If there are no adverse findings, management of the audited organization may waive the audit exit. Such waiver shall be documented in the audit report."
5. ANSI N45.2.12 Paragraph 4.4, Reporting:
   a. This paragraph requires that the audit report be signed by the audit team leader which is not always the most expeditious route for the audit report to be issued as soon as practical. RNP will comply with Paragraph 4.4 as clarified to read:
      "An audit report shall be signed by the audit team leader or the leader's supervisor in the absence of the audit team leader. In cases where the audit report is not signed by the audit team leader due to the leader's absence, the record copy of the report must be signed by the audit team leader upon return . The report shall not require the audit team leader's review/concurrence/signature if the audit team leader is no longer employed by Duke Energy at the time audit report is issued. The audit report shall provide:"
   b. RNP will comply with subparagraph 4.4.3 clarified to read: "Supervisory level personnel with whom significant discussions were held during the course of preaudit (where conducted) , audit, and post audit (where conducted) activities.
   c. Subparagraph 4.4.6 requires audit reports to include recommendations for corrective actions. RNP may choose not to comply with this requirement. Instead, audit reports are required to document findings.

Table C17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.146, Qualification of QA Program Audit Personnel for Nuclear Power Plants (Revision 0) (August, 1980)
ANSI Standard N45.2.23-1978, Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
RNP shall comply with NRC Regulatory Guide 1.146, Revision 0, which endorses ANSI N45.2.23-1978, with the following exceptions:

1. Section 1.4 titled Definitions: Definitions in this Standard which are not included in ANSI N45.2.10 will be used; definitions which are included in ANSI N45.2.10 will be used as clarified in RNP commitment to Regulatory Guide 1.74.
2. Section 2.2 titled Qualification of Auditors: Subsection 2.2.1 references an ANSI B45.2, which will be assumed to be N45.2. RNP will comply with an alternate subsection 2.2.1 which reads:
   Orientation to provide a working knowledge and understanding of the QA program, including the Regulatory Guides and ANSI standards included in the program, and Duke Energy procedures for performing audits and reporting results.
3. Section 4.1 titled Organizational Responsibility: RNP will comply with this Section with the substitution of the following sentence in place of the last sentence in the Section.
   NOS Management or the Audit Team Leader shall, prior to commencing the audit, assign personnel who collectively have experience or training commensurate with the scope, complexity, or special nature of the activities to be audited.
4. Section 5.3 titled Updating of Lead Auditors' Records: RNP will substitute the following sentence for this Section:
   Records for each Lead Auditor shall be maintained and updated during the annual management assessment as defined in Section 3.2 (as clarified).
5. Section 5.4 titled Record Retention: RNP will substitute the following sentence for this section.
   Qualification records shall be retained as required by the QA Program.
6. Section 2.3.4 titled For Audits: RNP will substitute the following instead of the cited sentence. Prospective Lead Auditors shall demonstrate the ability to effectively implement the audit process and effectively lead an audit team. This process is described in written procedures, which provide for evaluation and documentation of the results of this demonstration. In addition, the prospective Lead Auditor shall have participated in at least two Nuclear Oversight audits within a one-year period preceding the individual's effective date of qualification. Upon successful demonstration of the ability to effectively implement the audit process and effectively lead audits, and having met other provisions of Section 2.3 of ANSI/ASME N45.2.23-1978, the individual may be certified to lead audits.

Table C17-2. Site Specific Response to Regulatory Guides and Industry Standards

Table C17-2 identifies additional Regulatory Guides addressing subjects related to implementation of the QAP but the implementation is site specific and controlled with the UFSAR in accordance with 10 CFR 50.59.

Regulatory Guide 1.8, Personnel Selection and Training
Personnel selection and training is site specific. Robinson addresses conformance with Regulatory Guide 1.8 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
Quality group classifications and standards trace to the original design and construction of the nuclear power plant and therefore are site specific. Robinson does not address Regulatory Guide 1.26 in UFSAR Chapter 1 Section 8. Quality group classifications are addressed in UFSAR Chapter 3.

Regulatory Guide 1.29, Seismic Design Classification
Seismic design classification trace to the original design and construction of the nuclear power plant and therefore is site specific. Robinson addresses conformance with Regulatory Guide 1.29 in UFSAR Chapter 1 Section 8.

Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel
Nonmetallic thermal insulation for austenitic stainless steel trace to the original design and construction of the nuclear power plant and therefore is site specific. Robinson does not address conformance with Regulatory Guide 1.36 in UFSAR Chapter 1 Section 8. See UFSAR Chapters 5 and 6 for insulation of austenitic stainless steel.

Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
Quality assurance requirements for protective coatings applied to water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Robinson addresses conformance with Regulatory Guide 1.54 in UFSAR Chapter 1 Section 8.

Table C17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Robinson does not address conformance with Regulatory Guide 1.143 in UFSAR Chapter 1 Section 8. Design guidance for radioactive waste management systems, structures, and components is addressed in UFSAR Chapter 11.

Regulatory Guide 1.155, Station Blackout
Addressing Station Blackout is site specific. Robinson addresses conformance with Regulatory Guide 1.155 in UFSAR Chapter 1 Section 8.

Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment
Quality assurance for radiological monitoring program (normal operations) – effluent streams and the environment is site specific. Robinson addresses Regulatory Guide 4.15 in UFSAR Chapter 1 Section 8.

C17.3.1 MANAGEMENT

C17.3.1.1 Methodology

There are no Robinson specific amplifications for this section.

C17.3.1.2 Organization

There are no Robinson specific amplifications for this section.

C17.3.1.3 Responsibility

There are no Robinson specific amplifications for this section.

C17.3.1.4 Authority

The program and procedures require that the authority and duties of persons and organizations performing activities affecting quality functions be clearly established and delineated in writing and that these individuals and organizations have sufficient authority and organizational freedom to:

1. Identify quality, nuclear safety, and performance problems.
2. Order unsatisfactory work to be stopped and control further processing, delivery, or installation of nonconforming material.
3. Initiate, recommend, or provide solutions for conditions adverse to quality.
4. Verify implementation of solutions.

C17.3.1.5 Personnel Training and Qualification

There are no Robinson specific amplifications for this section.

C17.3.1.6 Corrective Action

The program requires that an evaluation of adverse conditions such as conditions adverse to quality, nonconformances, failures, malfunctions, deficiencies, deviations, and defective material and equipment is conducted to determine need for corrective action.

Conditions adverse to quality are identified through inspections, assessments, tests, checks, and review of documents.

The program requires corrective action to be initiated to preclude recurrence of significant conditions adverse to quality.

Procedures require follow-up reviews, verifications, inspections, etc., to be conducted to verify proper implementation of corrective action and to close out the corrective action documentation.

The program outlines the methodology for resolution of disputes involving quality and nuclear safety issues arising from a difference of opinion between identifying personnel and other groups.

Significant conditions adverse to quality are reported to appropriate management for review and evaluation.

Periodic review and evaluation of adverse trends are performed by management.

C17.3.1.7 Regulatory Commitments

There are no Robinson specific amplifications for this section.

C17.3.2 PERFORMANCE/VERIFICATION

C17.3.2.1 Methodology

There are no Robinson specific amplifications for this section.

C17.3.2.2 Design Control

There are no Robinson specific amplifications for this section.

C17.3.2.3 Design Verification

There are no Robinson specific amplifications for this section.

C17.3.2.4 Procurement Control

Potential contractors and suppliers are evaluated prior to award of a procurement contract when needed to assure the contractor's or supplier's capability to comply with applicable technical and quality requirements.

Procurement documents, such as purchase specifications, contain or reference the following:

1. Technical, administrative, regulatory, and reporting requirements, including material and component identification requirements, drawings, specifications, codes and industrial standards, test and inspection requirements, and special process instructions.
2. Identification of the documentation to be prepared, maintained, or submitted (as applicable) to RNP for review and approval. These documents may include, as necessary, inspection and test records, qualification records, or code required documentation
3. Identification of those records to be retained, controlled, and maintained by the supplier, and those delivered to the purchaser prior to use or installation of the hardware.

Procurement documents require suppliers to operate in accordance with QA programs which are compatible with the applicable requirements of RNP's QA Program and procedures where their services are utilized in support of plant activities.

C17.3.2.5 Procurement Verification

There are no Robinson specific amplifications for this section.

C17.3.2.6 Identification and Control of Items

Procedures require that materials, parts, and components be identified and controlled to prevent the use of incorrect or defective items. These procedures also require that identification of items be maintained either on the item in a manner that does not affect the function or quality of the item, or on records traceable to the item.

Procedures implementing these requirements provide for the following:

1. Verification that items received at the plant are properly identified and can be traced to the appropriate documentation, such as drawings, specifications, purchase orders, manufacturing and inspection documents, nonconformance reports, or material test reports.
2. Verification of item identification consistent with the RNP inventory control system and traceable to documentation which identifies the proper uses or applications of the item.

C17.3.2.7 Handling, Storage, and Shipping

Provisions are established to control the shelf life and storage of chemicals, reagents, lubricants, and other consumable materials.

C17.3.2.8 Test Control

Test procedures incorporate or reference the following, as required:

1. Instructions and prerequisites for performing the test,
2. Use of proper test equipment,
3. Mandatory inspection hold points,
4. Acceptance criteria

Test results are documented, evaluated, and their acceptability determined by a qualified, responsible individual or group.

When the acceptance criteria is not met, affected areas are to be retested or evaluated, as appropriate.

C17.3.2.9 Measuring and Test Equipment Control

Portable measuring and test equipment are calibrated by standards at least four times as accurate as the portable measuring and test equipment, unless limited by the state of the art.

Special tools such as torque wrenches, calipers, and micrometers are calibrated to be at least as accurate as the application(s) for which it is used, using standards which are at least as accurate as the special tool being calibrated.

Installed measuring and test instruments are calibrated by instruments at least as accurate as the installed, unless limited by the state of the art.

Reference and transfer standards are traceable to nationally recognized standards; or where national standards do not exist, provisions are established to document the basis for the calibration.

C17.3.2.10 Inspection, Test, and Operating Status

These procedures include the application, removal, and verification of inspection and welding stamps, or other status indicators as appropriate.

Altering the sequence of required tests, inspections, and safety-related operations can only be accomplished by methods outlined in procedures.

C17.3.2.11 Special Process Control

There are no Robinson specific amplifications for this section.

C17.3.2.12 Inspection

There are no Robinson specific amplifications for this section.

C17.3.2.13 Corrective Action

The primary goal of the RNP corrective action program is to improve overall plant operations and performance by identifying and correcting root causes of equipment and human performance problems.

Procedures define requirements for a corrective action program that charges personnel working at or supporting the nuclear plants with the responsibility to identify adverse conditions (including conditions adverse to quality).

Procedures include requirements for verification of the acceptability of the rework/repair of items by reinspection and/or testing in accordance with the original inspection or test requirements or by an accepted alternative inspection and testing method.

Conditions that require rework/repairs are identified through the use of maintenance work request forms.

C17.3.2.14 Control of Documents

Changes to documents are reviewed and approved by the same organization that performed the original review and approval or by other designated qualified responsible organizations.

C17.3.2.15 Records

The structures in which certain records are maintained are designed to prevent destruction, deterioration, or theft. These structures ensure protection against destruction by fire, flooding, theft, and deterioration by the environmental conditions of temperature and humidity.

C17.3.3 ASSESSMENT

C17.3.3.1 Methodology

There are no Robinson specific amplifications for this section.

C17.3.3.2 Independent Review

There are no Robinson specific amplifications for this section.

C17.3.3.3 Independent Assessment

There are no Robinson specific amplifications for this section.

C17.3.3.3.1 Organization

There are no Robinson specific amplifications for this section.

C17.3.3.3.2 Internal Assessment process

There are no Robinson specific amplifications for this section.

C17.3.3.3.3 Internal Audit Program

C17.3.3.3.3.1 Other Reviews Prescribed by the Code of Federal Regulations

There are no Robinson specific amplifications for this section.

C17.3.3.3.3.2 Independent Audit of Fire Protection Program

There are no Robinson specific amplifications for this section.

C17.3.3.3.4 Results

There are no Robinson specific amplifications for this section.

C17.3.3.3.5 Supplier Oversight

There are no Robinson specific amplifications for this section.

C17.3.3.3.6 Independent Audit of QA Functions

There are no Robinson specific amplifications for this section.

C17.3.3.3.7 Audit Frequency Extensions

There are no Robinson specific amplifications for this section.

C17.3.4 REVIEW AND AUDIT

C17.3.4.1 Procedures, Tests, and Experiments

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.1 follows:

1. The procedures established, implemented, and maintained for the Quality Assurance Program for effluent and environmental monitoring use guidance from Regulatory Guide 4.15. RNP is not committed to specific guidance within Regulatory Guide 4.15 or to a specific revision to the Regulatory Guide.
2. 10 CFR 50.59 reviews are addressed in Section 17.3.4.2.

C17.3.4.2 Modifications

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.2 modifications are addressed in Section 17.3.2.2, Design Control.

C17.3.4.3 RNP Technical Specifications and License Changes

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.3 follows:

Each proposed RNP Technical Specification or Operating License change for the 10CFR 50 license and 7P-ISFSI license is reviewed per Section 17.3.3.2 and submitted to the NRC for approval. The 24P ISFSI RNP Technical Specifications and License are processed by Transnuclear, Inc., and will only be reviewed by the On-Site Review Committee if a plant specific safety issue is identified.

C17.3.4.4 Review of RNP Technical Specifications Violations

Addressed in Section 17.3.4.6.

C17.3.4.5 10CFR 50.59 Review Qualification

Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.5 is addressed in Section 17.3.4.2.

C17.3.4.6 Plant Nuclear Safety Committee (PNSC)

Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.6 is addressed in Section 17.3.3.2, Independent Review.

C17.3.4.7 Independent Review Program

Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.7, Nuclear Oversight Section Independent Review Program, has been replaced by Section 17.3.3.2, Independent Review.

C17.3.4.8. (Deleted)

There was no content in Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.8.

C17.3.4.9. Outside Agency Inspection and Audit Program

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 1.9 is reflected in Section 17.3.3.3.3.2, Independent Audit of Fire Protection Program.

C17.3.4.10. Reportable Event Action

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 2.0 is addressed in Section 17.3.4.6, Reportable Event Action.

C17.3.4.11. Safety Limit Violation

Content from Robinson UFSAR Section 17.3 Appendix A, QA Program Relocated Technical Specifications Requirements, Section 3.0 is addressed in 17.3.4.6.

C17.3.4.12. Record Retention

A list of typical operational phase QA Records is included in 17.3.2.15.

Attachment D, Catawba, McGuire, and Oconee Specific QAPD

The term 'Duke Energy Carolinas' as used in this document means Catawba, McGuire, and Oconee Nuclear Plants. If content is specific to a single nuclear plant, that nuclear plant will be identified by name. See Table D17-2 addressing Regulatory Guide 1.8 for example.

Information presented in this attachment was contained in the Duke Energy Carolinas Topical Report Quality Assurance Program prior to Amendment 41.

Where a section contains no descriptive information beyond that in the generic text in the body of the document, a statement is made to that effect and no content is included. See D17.3.1.2, Organization for example.

D17. QUALITY ASSURANCE

D17.1 QA DURING DESIGN AND CONSTRUCTION

Deleted

D17.2 OPERATIONAL QA

Deleted

(NOTE: In August 1992, Amendment 15 of the Duke Energy Carolinas Topical Report reformatted the description of the QA Program to follow Standard Revision Plan Section 17.3, replacing the content of 17.1 and 17.2.)

D17.3 QUALITY ASSURANCE PROGRAM (QAP) DESCRIPTION

INTRODUCTION

As discussed herein, the Quality Assurance Program (QAP) includes the description contained in this document and the procedures providing implementation of the requirements of this document, including the requirements of industry standards to the degree identified in Table 17-1. This Topical Report describes the QAP for those systems, components, items, and services which have been determined to be nuclear safety related. The QAP provides a method of applying graded controls to certain non-nuclear safety related systems, components, items, and services (such as fire protection and radioactive waste structures, systems, and components) through implementing documents.

Duke Energy Carolinas may use QA Conditions as a method for identifying applicability of the QAP, where implementing documents define a Quality Assurance (QA) "Condition" for each level of QA required. These will be designated as "QA Condition _____". The quality of systems, components, items, and services within the scope of QA Conditions is assured through implementing documents commensurate with the system's, component's, item's, or service's importance to safety.

In this approach, QA Condition 1 identifies those systems and their attendant components, items, and services which have been determined to be nuclear safety related. These systems are detailed in the Safety Analysis Report applicable to each nuclear station. The Topical Report applies in its entirety to systems, components, items, and services identified as QA Condition 1.

QA Condition 5 covers those systems, components, items, and services which are important to the mitigation of design basis and other selected events as defined in applicable procedures and directives. QA Condition 5 only applies to Oconee Nuclear Station.

QA Conditions 2, 3, 4, and others are defined in implementing documents. These address SSCs and related functions important to the management and containment of liquid, gaseous, and solid radioactive waste, important to fire protection, seismic interaction, etc.

QA Condition 3 includes those fire protection features (systems, components, items, and services) which are credited in addressing 10 CFR 50.48.

Quality assurance program requirements for Oconee, McGuire, and Catawba dry cask storage activities are performed in accordance with applicable 10CFR72.212 reports for each site which invoke the NRC approved 10CFR50 Appendix B QAP as described in this Topical Report.

DEFINITIONS

There are no Duke Energy Carolinas specific definitions.

EXPLANATION OF "QUALITY ASSURANCE"

There is no Duke Energy Carolinas specific content.

QA STANDARDS AND GUIDES

Table D17-1 and D17-2 address Catawba, McGuire, and Oconee conformance to the referenced regulatory and program guidance in NUREG-0800 Section 17.3.

Changes to the content of Table D17-1 are controlled in accordance with 10 CFR 50.54(a). Subsequent changes to the QAP are incorporated in this document as identified in Section 17.3.1.7.

Table D17-2 addresses additional Regulatory Guides that relate to implementation of the QAP but the implementation is site specific and controlled with each site's UFSAR.

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards

Generic Exception:

Table D17-1 addresses Duke Energy Carolinas Conformance of the Quality Assurance Program to certain NRC Regulatory Guides. In so doing, specific editions of industry standards are identified for compliance with exceptions and alternatives. Those identified standards include references to other industry standards for activities including, but not limited to; design, fabrication, inspection, and testing. Those included reference industry standards are considered to be guidance documents for details of how activities may be accomplished. The actual standard to be used in such cases is controlled by each station's current licensing and design bases.

Regulatory Guide 1.28, Rev (2), Feb. 1979 – Quality Assurance Program Requirements (Design and Construction)
Duke Energy Carolinas conforms to Regulatory Guide 1.28 Rev (2) and ANSI N45.2-1977 with the clarifications and exceptions noted below.
Exception to ANSI N45.2 Section 5. Duke Energy Carolinas procurement documents shall require suppliers to provide a quality assurance program consistent with the pertinent requirements of 10 CFR Part 50 Appendix B instead of ANSI N45.2-1977. Alternate requirements for purchase of Commercial Grade Items are described in this table addressing compliance for Regulatory Guide 1.123.

Regulatory Guide 1.30, Rev 0, Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment
Duke Energy Carolinas conforms to Regulatory Guide 1.30 Rev 0 and ANSI N45.2.4-1972 with the following Clarifications and Exceptions:
Conforms with no exceptions.

Regulatory Guide 1.33, Rev 2, Quality Assurance Program Requirements (Operation)
Duke Energy Carolinas conforms to Regulatory Guide 1.33 Rev 2 and ANSI N18.7-1976/ANS-3.2 with the following Clarifications and Exceptions:
Regulatory position C.4 modifies the audit frequencies in Section 4.5 of ANSI N18.7. Duke Energy Carolinas takes exception to this regulatory position. The audits of selected aspects of operational phase activities as identified in Section 17.3.3.3.3, Internal Audit Program, are performance based. The schedule is based on plant performance and importance to safety but at a frequency not to exceed twenty-four months with extensions as allowed in Section 17.3.3.3.7, Audit Frequency Extensions.

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.33, Rev 2, Quality Assurance Program Requirements (Operation)
Duke Energy Carolinas conforms to Regulatory Guide 1.33 Rev 2 and ANSI N18.7-1976/ANS-3.2 with the following Clarifications and Exceptions:
Exception to ANSI N18.7-1976, Section 5.2.15, Review, Approval and Control of Procedures, which states in part that, "Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary. A revision to a procedure constitutes a procedure review." In lieu of this paragraph, Duke Energy addresses programmatic controls in Section 17.3.2.14 to continually identify procedure revisions which may be needed to ensure that procedures are appropriate for the circumstance and are maintained current. When purchasing commercial-grade calibration services from certain accredited calibration laboratories, the procurement documents are not required to impose a QAP consistent with ANSI N45.2-1977. Alternate requirements described in the QA Topical Report for Regulatory Guide 1.123 may be implemented in lieu of imposing a QAP consistent with ANSI N45.2-1977. Exception to Paragraph C.3 of Regulatory Guide 1.33 and ANSI N18.7-1976 Paragraph 4.3: Independent Review Program requirements are replaced by Section 17.3.3.2, Independent Review. This exception uses NRC Safety Evaluation dated January 13, 2005 to Nuclear Management Company (ADAMS ML050210276). Section 5.2.2 titled Procedure Adherence first paragraph addresses temporary change to procedures, which is clarified as follows: Temporary changes to procedures, tests, or experiments may be made provided; a) such change does not change the intent of the original procedure, test, or experiment; b) the change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected; and c) the change is documented and approved as a permanent change or deleted within 14 days of implementation.
Regulatory Guide 1.37, Rev 0, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.37 Rev 0 and ANSI N45.2.1-1973 with the following clarifications and exceptions:
Conforms with no exceptions.
Regulatory Guide 1.38, Rev 2, May 1977 – Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.38 Rev 2 and ANSI N45.2.2-1972 with the following Clarifications and Exceptions:
Container markings shall be marked on at least one side (A.3.9(1)) and shall be applied with waterproof ink or paint in characters of a legible size, and caps and plugs for pipe and fittings are required unless specified by Engineering, and off-site inspection, examination, and testing is monitored by personnel qualified to ANSI N45.2.12 in lieu of ANSI N45.2.6.

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.39, Rev (2), Sept. 1977 – Housekeeping Requirements for Water-Cooled Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.39 Rev 2 and ANSI N45.2.3-1973 with the following clarifications and exceptions:
Personnel accountability for personnel entering housekeeping zones I, II, and III without materials shall be maintained by housekeeping logs or alternate methods such as radiation work permits, confined space permits, work requests or other accepted methods capable of assuring personnel accountability.
Regulatory Guide 1.58, Rev (1), Sept. 1980 – Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel
Duke Energy Carolinas conforms Regulatory Guide 1.58 Rev 1 and ANSI N45.2.6-1978 with the following Clarifications and Exceptions:
Duke Energy Carolinas nondestructive examination (NDE) personnel will meet the qualification requirements of SNT TC-1A and ANSI/SNT-CP-189 as governed by the applicable ASME Section XI requirement or other code requirement consistent with the conditions identified in 10 CFR 50.55a. Operational/functional testing personnel will meet the requirements of ANSI N18.1 or ANS 3.1 rather than ANSI N45.2.6. This reflects Regulatory Position C.1. With regard to Paragraph 3 of ANSI N45.2.6-1978 titled Qualifications: Only personnel performing NDE (such as LP, MT, UT, and RT) are required to be grouped in levels of capability and certified for inspection, review, and evaluation of inspection data, and reporting of inspection and test results. In lieu of qualification by Levels, inspection personnel may be qualified based on pre-established experience, education, on-the-job training, written examinations and proficiency tests associated with the specific activity. Proficiency tests are given to personnel performing independent QC inspections and documented acceptance criteria are developed to determine if individuals are properly trained and qualified. Certificates of qualification delineate the functions personnel are qualified to perform. Qualification records are maintained and performance evaluations conducted at least once every three years. If organizations elect to utilize qualifications by levels, Level I inspectors receive a minimum of 4 months experience as Level I before being certified as Level II, in lieu of one year experience recommended by ANSI N45.2.6 Section 3.5.2(1). Organizations identify in their procedures if they qualify their inspectors by Level or by task qualifications. Inspectors are only assigned functions for which they have been qualified.

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.64, Rev (2), June 1976 – Quality Assurance Requirements for the Design of Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.64, Rev. 2 and ANSI Standard N45.2.11-1974 with the following Clarifications and Exceptions:
The use of the originator's immediate supervisor for design verification shall be restricted to special situations where the immediate supervisor is the only individual capable of performing the verification. Advance justification for such use shall be documented and signed by the supervisor's management. And the frequency and effectiveness of the supervisor's use as design verifier are independently verified to guard against abuse. The supervisor will not be the design verifier on work for which he is the actual performer / originator.
Regulatory Guide 1.74, Rev (0), Feb. 1974 – Quality Assurance Terms and Definitions
Duke Energy Carolinas conforms to Regulatory Guide 1.74, Rev 0 and ANSI N45.2.10-1973 with the following Clarifications and Exceptions:
The quality assurance terms and definitions contained in ANSI N45.2.10-1973 are generally used in describing and implementing the quality assurance program described in this QAPD except where terms are explicitly defined in this document.
Regulatory Guide 1.88, Rev (2), Oct. 1976 - Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
Duke Energy Carolinas conforms to Regulatory Guide 1.88, Rev. 2 and ANSI N45.2.9-1974 with the following Clarifications and Exceptions:
The records storage facilities have a minimum 3-hour rating. A qualified Fire Protection Engineer (meeting Professional Member grade qualifications of the SFPE) will evaluate record storage areas (including satellite files) to assure records are adequately protected from damage. The Duke Energy Carolinas program for storage of records on microfilm, dual storage or in electronic format meets the preservation requirement for the retention of QA Records. See standard exception in Table 17-1 Regulatory Guide 1.88 for the appropriate controls on quality in the management of electronic records.
Regulatory Guide 1.94, Rev (1), Apr. 1976 – Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.94. Rev. 1 and ANSI N45.2.5-1974 with the following Clarifications and Exceptions:
The length of bolts shall be flush with the outside face of the nut. Paragraph 5.5 requires inspection of structural steel welding to be performed in accordance with the provisions of Section 6 of the AWS D1.1. Visual Weld Acceptance Criteria (VWAC) for Structural Welding at Nuclear Power Plants, NCIG-01, Revision 2, prepared by the Nuclear Construction Issues Group (NCIG) and accepted by the NRC in their letter to the NCIG dated June 26, 1985 may be used as an alternative to AWSD1.1 for non ASME Code structural weld inspections. (July 31, 2000 J M Farley SER)

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.116, Rev (0-R), June 1976, (Reissued May 1977) – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
Duke Energy Carolinas conforms to Regulatory Guide 1.116 Rev (0-R) and ANSI N45.2.8-1975 with the following Clarifications and Exceptions:
Conforms
Regulatory Guide 1.123, Rev (1), July 1977 – Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.123 and ANSI N45.2.13-1976 with the following clarifications and exceptions:
Section 3.2, "Content of the Procurement Documents," Subsection 3.2.3, "QAP Requirement," Duke Energy Carolinas takes the following exception: See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.
Regulatory Guide 1.144, Rev (1), Sept. 1980 - Auditing of Quality Assurance Programs for Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.144, Rev 1 and ANSI N45.2.12-1977 with the following clarifications or exceptions:
Section 4.4.6. In lieu of making recommendations for correcting program deficiencies we will identify the deficiencies to the audited organization. For external audits, the results of the audit will be provided to the audited organization in lieu of the audit report. Also, the re-evaluation may be extended to 15 months and the triennial period as specified in Regulatory Position c.3.b.(2) may be extended as described in Section 17.3.3.3.7, Audit Frequency Extensions. Additionally, the Duke Energy Carolinas QAP meets regulatory position C.3.b of this regulatory guide, as clarified by NRC Information Notice 86-21, Supplement 2. Internal Technical Audits shall require a response describing corrective action and implementation schedule as requested by the audit report but not to exceed sixty days of receipt of the audit report. See standard exceptions in Table 17-1 for Regulatory Guide 1.123 for the procurement of Commercial Grade Items and services including, purchasing commercial-grade calibration services from calibration laboratories.

Table D17-1. Conformance with QA Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.146, Rev (0), Aug. 1980 – Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
Duke Energy Carolinas conforms to Regulatory Guide 1.146 Rev 0 and ANSI N45.2.23-1978 with the following clarifications and Exceptions:
In lieu of prospective lead auditors participating in a minimum of five QA audits within a period of three years prior to date of certification, prospective lead auditors shall demonstrate their ability to effectively lead an audit team and shall have participated in at least one nuclear QA audit within one year preceding the individual's effective date of qualification. Upon successful demonstration of the ability to lead audits, and having met the other provisions of ANSI N45.2.23-1978, the individual may be certified as being qualified to lead audits. This process is described in approved procedures which require documentation of the evaluation and demonstration of results.
Regulatory Guide 1.152 Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants
Conformance to Regulatory Guide 1.152 was not addressed during the licensing of the operating Duke Energy Carolinas Nuclear plants.
Regulatory Guide 7.10, Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material
Duke Energy Carolinas does not conform to Regulatory Guide 7.10. This QAPD is used to satisfy applicable Quality Assurance requirements for packaging and transportation of radioactive material.
Generic Letter 89-02, Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products
See Table 17-1.

Table D17-2. Site Specific Response to Regulatory Guides and Industry Standards

Table D17-2 identifies additional Regulatory Guides addressing subjects related to implementation of the QAP but the implementation is site specific and controlled with each site's UFSAR in accordance with 10 CFR 50.59.

Regulatory Guide 1.8, Personnel Selection and Training
Personnel selection and training is site specific. Catawba addresses conformance with Regulatory Guide 1.8 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.8 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide 1.8. Personnel selection and training is addressed in UFSAR Chapter 13.

Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
Quality group classifications and standards trace to the original design and construction of the nuclear power plant and therefore are site specific. Catawba addresses conformance with Regulatory Guide 1.26 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.26 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide 1.26. Quality group classifications and standards are addressed in UFSAR Section 3.2.2.

Regulatory Guide 1.29, Seismic Design Classification
Seismic design classification trace to the original design and construction of the nuclear power plant and therefore is site specific. Catawba addresses conformance with Regulatory Guide 1.29 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.29 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide 1.29. Seismic design classifications are addressed in UFSAR Section 3.2.1.

Table D17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel
Nonmetallic thermal insulation for austenitic stainless steel trace to the original design and construction of the nuclear power plant and therefore is site specific. Catawba addresses conformance with Regulatory Guide 1.36 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.36 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide 1.36. Thermal insulation for austenitic stainless steel is addressed in UFSAR Section 5.4.

Regulatory Guide 1.54, Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
Quality assurance requirements for protective coatings applied to water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Catawba addresses conformance with Regulatory Guide 1.54 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.54 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide 1.54. Protective coatings are addressed in UFSAR Section 6.2.1.6.

Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants
Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants trace to the original design and construction of the nuclear power plant and therefore is site specific. Catawba addresses conformance with Regulatory Guide 1.143 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.143 in UFSAR Chapter 1 Table 1-4. Oconee does not address conformance with Regulatory Guide1.143. Design guidance for radioactive waste management systems, structures, and components is addressed in UFSAR Chapter 11.

Table D17-2. Site Specific Response to Regulatory Guides and Industry Standards (Continued)

Regulatory Guide 1.155, Station Blackout
Addressing Station Blackout is site specific. Catawba addresses conformance with Regulatory Guide 1.155 in UFSAR Chapter 1 Section 7. McGuire addresses conformance with Regulatory Guide 1.155 in UFSAR Chapter 1 Table 1-4. Oconee address conformance with Regulatory Guide 1.155 in UFSAR Chapter 8.

Regulatory Guide 4.15, Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment
Quality assurance for radiological monitoring program (normal operations) – effluent streams and the environment is site specific. Catawba addresses conformance with Regulatory Guide 4.15 in UFSAR Chapter 1 Section 7. McGuire does not address conformance to Regulatory Guide 4.15 in UFSAR Chapter 1 Table 1-4. The radiological monitoring program is addressed in UFSAR Chapter 11. Oconee does not address conformance with Regulatory Guide 4.15. The radiological monitoring program is addressed in UFSAR Chapter 11.

D17.3.1 MANAGEMENT

D17.3.1.1 Methodology

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.1.2 Organization

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.1.3 Responsibility

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.1.4 Authority

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.1.5 Personnel Training and Qualification

The following provide Duke Energy Carolinas specific amplifications for this section.

A training program is established for each nuclear station and support organization to develop and maintain an organization qualified to be responsible for operation, engineering, testing, inspection, maintenance, engineering changes and other technical aspects of the nuclear station involved. The program is formulated to provide the required training based on individual employee experience and intended position. The program is in compliance with NRC licensing requirements, where applicable. The training program is such that trained and qualified operating, maintenance, work control, engineering, inspection, testing, technical support and supervisory personnel are available in necessary numbers at the times required. In all cases, the objectives of the training program shall be to assure safe and reliable operation of the station.

A continuing effort is used after a station goes into commercial operation for training of replacement personnel and for periodic retraining, reexamining, and/or recertifying as required to assure that personnel remain proficient. Personnel receive orientation training in basic QA policies and practices.

Personnel receive additional training, as appropriate, which addresses specific topics such as NRC regulations and guides, QA procedures, auditing and applicable codes and standards. Special training of personnel in QA related matters, particularly new or revised requirements, is conducted as necessary. Training and qualification records are maintained for each employee. Documentation of training includes the objectives, content of the program, attendees, and date of attendance.

D17.3.1.6 Corrective Action

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.1.7 Regulatory Commitments

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.2 PERFORMANCE/VERIFICATION

D17.3.2.1 Methodology

The following provide Duke Energy Carolinas specific amplifications for this section.

The program receives on-going review and is revised as necessary to assure its continued effectiveness.

D17.3.2.2 Design Control

Each design document is checked by another individual qualified in the same discipline and is reviewed for concept and conformity with applicable codes, standards, and other design inputs (as specified within the design documentation package). The document is approved by the individual having overall responsibility for the design function. A review of each specification is made to assure incorporation of necessary QA information. The entire review process is documented.

Computer programs are controlled in accordance with appropriate department procedures, whereby programs are certified to demonstrate their applicability and validity.

D17.3.2.3 Design Verification

Analytical models, theories, examples, tables, codes, computer programs, etc., used as bases for design must be referenced in the design document and their application verified in the design verification. Model tests, when required, to prove the adequacy of concept or design are reviewed and approved by the responsible engineer. The tests used for design verification must meet all the requirements of the designing activity. Computer programs are controlled in accordance with the applicable software QA document whereby programs are certified to demonstrate their applicability and validity.

Following completion of design and evaluation of an engineering change, the responsible individual/organization summarizes the engineering change design and identifies the design documents and information required for engineering change implementation. This information is provided for design verification. This addresses such items as:

a) A description of the engineering change.
b) References utilized in the evaluation and design of the engineering change, and necessary for the implementation of the engineering change.
c) Special installation instructions.
d) Operational, test, maintenance and inspection requirements.
e) Materials, parts and components required in order to implement the engineering change.
f) Drawings revised and/or requiring revision.
g) UFSAR revision(s) and/or Technical Specifications amendment(s) necessary.
h) Whether or not the engineering change requires a license amendment.

D17.3.2.4 Procurement Control

Procedures identify the responsibility within Nuclear Generation for the technical qualification of suppliers and control of the initial procurement of nuclear safety related items and services. Procurement requirements/specifications are prepared, checked, and approved by appropriate

personnel and forwarded to Nuclear Supply Chain for procurement actions from qualified suppliers.

Technical qualifications are determined by engineering personnel. Commercial qualification is determined by Supply Chain following evaluation of bids from qualified suppliers. Bid evaluation includes evaluation of the technical, quality and commercial qualifications of the prospective suppliers.

NOS performs qualification of supplier QA programs. NOS may place a supplier on the Qualified Suppliers List following review, approval and acceptance of an audit performed by another licensed nuclear utility or joint utility audit team. Review of such third party audits shall ensure that items to be procured are within the audit scope and any unique plant quality and technical requirements are adequately addressed by such audits. When basic components and services are procured from a supplier whose quality performance has not been verified by audit, additional assurance of product quality shall be obtained by supplier surveillance, inspection or test.

Materials, parts and components shall be procured to specified technical and quality requirements at least equivalent to those applicable to the original equipment or those specified by a properly reviewed and approved revision. As required by the applicable purchase documents, suppliers furnish documentation which identifies the material and equipment purchased and the specific procurement requirements met by the items. Also, as required by the applicable purchase documents, suppliers will provide documentation which identifies any procurement requirements which have not been complied with, together with a description of any deviations and repair records.

Procurement of materials, parts, components and services associated with nuclear safety related structures, systems, and components is controlled during the operational life of the station so as to assure the suitability for their intended service and that the safety and reliability of the station are not compromised.

Procurement information for nuclear safety related materials, parts and components is reviewed to assure that QA, technical and regulatory requirements including supplier documentation requirements are adequately incorporated into the purchase document(s). Significant changes to the content of such purchasing information are reviewed and approved in a manner consistent with the original.

Critical characteristics for the dedication of Commercial Grade Items are determined by Procurement Engineering or Supply Chain technical sponsors and approved by the responsible engineering personnel based on the manufacturer's published specifications and the intended safety function for the items. Critical characteristics used for acceptance and dedication of commercial grade items are selected to provide reasonable assurance that the items will meet their catalog or manufacturer specifications and will perform the necessary safety functions in the intended applications. Verification of critical characteristic acceptability will be by manufacturer/supplier survey, source verification, receipt tests or inspections, or post installation testing. Historical data, when documented, will represent industry wide experience.

If verification of a critical characteristic is to be by supplier survey, NOS is responsible for verifying the acceptability of the supplier control of the identified critical characteristic.

D17.3.2.5 Procurement Verification

NOS Vendor Quality performs a documented on-going evaluation of each qualified supplier in order to maintain the supplier on the qualified suppliers list. The evaluation is performed to a depth consistent with the item's or service's importance to safety, complexity, and the quantity

and frequency of procurement. As applicable, this evaluation takes into account (1) review of supplier-furnished documents such as certificates of conformance, nonconformance notices, and corrective actions, (2) results of previous source verifications, audits, and receiving inspections, (3) operating experience of identical or similar products furnished by the same supplier, and (4) results of audits from other sources (e.g., customer, ASME, or NRC audits). The results of the evaluations are reviewed and appropriate corrective action initiated. Adverse findings resulting from these evaluations are periodically reviewed in order to determine if, as a whole, they result in a significant condition adverse to quality and to provide input to support supplier audit activities conducted by the licensee or a third party auditing entity.

Suppliers of nuclear safety related items or services are re-evaluated by means of an audit at least triennially, if initial qualification was by audit or pre-award survey. The triennial audit schedule may be extended as identified in Section 17.3.3.3.7, Audit Frequency Extensions.

NOS is responsible for oversight when procurement documents require characteristics or processes to be witnessed, inspected or verified at the supplier shop. NOS surveillance activities assure that the supplier complies with all quality requirements outlined in the procurement document(s). The surveillance representative has the authority and responsibility to stop work when the required quality standards are not met.

D17.3.2.6 Identification and Control of Items

Specific identification requirements are as follows:

a) Materials, parts, components, assemblies, and subassemblies shall be identified either on the item or records traceable to the item to show that only correct items are received, issued and installed.
b) Some components, such as pressure vessels are identifiable by nameplates as required by applicable codes, or Duke Energy Carolinas specifications. Materials, parts, and components are traceable from such identification to a specific purchase order to manufacturer's records and to QA records and documentation.
c) When required by procurement documents, materials are identified by heat, batch or lot numbers which are traceable to the original material at receipt. Upon receipt, a unique tracking number is assigned to provide traceability. When several parts are assembled, a list of parts and corresponding numbers is included in the documentation.
d) When required by specifications or codes and standards, identification of material or equipment with the corresponding mill test reports, certifications and other required documentation is maintained throughout the life of the material or equipment by a unique tracking number.
e) Sufficient precautions will be taken to preclude identifying materials in a manner that will affect the function or quality of the item being identified.

Control of material, parts and components is governed by approved procedures. Specific control requirements include:

1) Nonconforming or rejected materials, parts, or components are identified to assure that they will not be inadvertently used.
2) The verification of correct identification of material, parts, and components is required prior to release for assembling, shipping and installation.
3) Upon receipt, procedures require that materials, parts or components undergo a receipt inspection to assure they are properly identified and that the supporting documentation is available as required by the procurement

requirements/specifications. Items having limited shelf or service life are identified and controlled.

4) Each organization which performs an operation that results in a change in the material, part or component is required to make corresponding revisions and/or additions to the documentation record as applicable.

When a designated item is subdivided, each subdivision is identified in accordance with the above requirements. Where physical identification of an item is impractical or insufficient, physical separation, administrative controls or other appropriate means are utilized.

D17.3.2.7 Handling, Storage, and Shipping

Conforming nuclear safety related materials, parts and components are stored in controlled, segregated areas designated for the storage of such items. Inspections and examinations are performed on a periodic basis to assure that recommended shelf life of chemicals, reagents, and other consumable materials is not exceeded. Hazardous items are stored in suitable environments with controls to prevent contamination of nuclear safety related structures, systems, or components.

D17.3.2.8 Test Control

Test controls include requirements on the review and approval of test procedures, and on the review and approval of changes to such procedures, as discussed in Section 17.3.2.14, "Document Control." Also, specific criteria are established with regard to procedure content. Examples of items which must be considered in the preparation and review of procedures include:

a) References to material necessary in the preparation and performance of the procedure, including applicable design documents.
b) Tests which are required to be completed prior to, or concurrently with, the specified testing.
c) Special test equipment required to perform the specified testing.
d) Limits and precautions associated with the testing.
e) Station, unit and/or system status or conditions necessary to perform the specified testing.
f) Criteria for evaluating the acceptability of the results of the specified testing, compatible with any applicable design specifications.

Test procedures contain the following information or require this information be documented:

1) Requirements and acceptance limits contained in applicable design and vendor documents.
2) Instructions for performing the test.
3) Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of the item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.
4) Mandatory inspection hold points.
5) Acceptance and rejection criteria.
6) Methods of documenting or recording test data and results.
7) Provisions to assure test prerequisites have been met.

Requirements are also established for verification of test completion and for determining acceptability of tests results. Test results are reviewed and accepted by the testing organization

and the organization responsible for the item being tested. In the event that test results do not meet test acceptance criteria, a review of the test, test procedure and/or test results is conducted to determine the cause, required corrective action, and retest as necessary.

In addition to the above periodic testing, after maintenance to, or modification of, nuclear safety related structures, systems and components, other post maintenance testing, post modification testing, or functional verifications are performed and documented as required to verify satisfactory performance of the affected items. Post maintenance/modification functional verifications are not subject to the requirements of periodic testing described above because they are acceptable good industrial practices that are simple and straightforward. Included in these tests are such items as diesel generators, reactor control rod systems, and leak testing of appropriate pressure isolation valves.

D17.3.2.9 Measuring and Test Equipment Control

Site specific content is retained for item c) as follows:

c) The tag or records for devices that have been acceptably calibrated include the date of calibration, the date the next calibration is due, an indication that the device is within calibration specifications and the identification of the individual who was responsible for performing the calibration.

Installed instrumentation is subject to the requirements of the Technical Specification and is not subject to the tagging requirements discussed in 17.3.2.9 c) and d). The NOS-Audit section verifies implementation of the calibration program through periodic audits.

The basis for this exception on the installed Technical Specification required equipment is the Preventive Maintenance Periodic Testing (PMPT) program. This is a computerized scheduling program that automatically schedules PMPT using model work orders. When devices have been acceptably calibrated, the clock starts for the next calibration due date. The indication that the device is within calibration specifications and identification of the individual who was responsible for performing the calibration is documented within the calibration procedure for the device. If the device fails to meet calibration specifications, it will be repaired, replaced and/or engineering involvement will be requested to further evaluate. The PMPT program along with the calibration procedures address all the requirements in Section 17.3.2.9 items c and d. Therefore, there is no need to place tags on the devices to identify the calibration status.

D17.3.2.10 Inspection, Test, and Operating Status

Inspections and tests required by the written approved procedures which address work activities are infrequently temporarily deferred. When such a deferral does occur, a discrepancy is considered to exist and documentation of the acceptable completion of the affected work activity is not performed until the discrepancy is resolved.

Proposed tests and experiments which affect station nuclear safety and are not addressed in the Updated Final Safety Analysis Report or Technical Specifications shall be prepared and approved in a manner identical to that used for station procedures as described in Section 17.3.2.14, "Document Control." These proposed tests and experiments shall be reviewed by a knowledgeable individual/organization other than the individual/organization which prepared the proposed tests and experiments.

D17.3.2.11 Special Process Control

The QAP contains or references procedures for the control of special processes such as welding, heat treating, NDE, coatings, crimping and cleaning. These procedures shall provide for documented evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel.

D17.3.2.12 Inspection

Independent inspections, examinations, measurements, observations, or tests of materials, products or activities are conducted, where necessary, to assure quality. If inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel is provided. Both inspection and process monitoring are provided when control is inadequate without both.

In addition to the content identified in 17.3.2.12, inspection procedures, instructions, and checklists contain the following information or require this information on inspection reports:

a) Measuring and test equipment information
b) Identification of required procedures, drawings, specifications, etc.

The personnel performing these inspections are examined and certified in their particular category. Current qualification and certification files are maintained for each inspector. NDE inspectors are certified in accordance with required codes and standards (See Table 17-1 Regulatory Guide 1.58). Written procedures require the test and certification of inspectors in other categories such as Mechanical, Electrical, and Structural as described in the appropriate QA manual. For cases where inspectors will perform limited functions within a category, they are tested and certified to those limitations. These inspectors are only allowed to perform inspections specifically defined in this limited certification.

For inspections of concrete containments, personnel fulfilling the role of Responsible Engineer, shall be a Registered Professional Engineer experienced in evaluating the in-service condition of structural concrete and knowledgeable of the design and construction codes and other criteria used in the design and construction of the concrete containment structure. The Responsible Engineer may also perform inspections as discussed in this section.

The inspection criteria for performing inspections are established from codes, specifications, and standards applicable to the activity. Examples of activities subject to inspection include:

a) Activities specified by the ASME Code Section XI
b) Special processes
c) Modifications
d) Maintenance
e) Material Receipt

After inspection data is collected and reviewed by the inspector, the reports are technically reviewed by personnel designated to perform that function.

D17.3.2.13 Corrective Action

Procedures require that conditions adverse to quality be corrected. In the case of significant conditions adverse to quality, the procedures assure that the cause of the condition is determined and action be taken to preclude repetition. Performance and verification personnel are to:

a) Identify conditions that are adverse to quality.
b) Suggest, recommend, or provide solutions to the problems as appropriate.
c) Verify resolution of the issue.

Additionally, performance and verification personnel are to ensure that reworked, repaired, and replacement items are to be inspected and tested in accordance with the original inspection and test requirements or specified alternatives.

Discrepancies revealed during the performance of station operation, maintenance, inspection and testing activities must be resolved prior to verification of the completion of the activity being performed. In the event of a significant malfunction of nuclear safety related structures, systems, and components, the cause of the failure is evaluated and appropriate corrective action taken. Items of the same type are evaluated to determine whether or not they can be expected to continue to function in an appropriate manner. This evaluation is documented in accordance with applicable procedures.

Nuclear safety related materials, parts and components which are determined to be nonconforming are identified, segregated or otherwise controlled (e.g. by a conditional release) in such a manner as to preclude their inadvertent substitution for and use as conforming materials, parts and components. The determination of an item's nonconformance is documented and is retained on file by Nuclear Generation and, as appropriate, by tags attached to the item. Nuclear Generation personnel are notified of any nonconformances identified in accordance with approved procedures.

Nuclear Generation maintains a listing of the status of all nonconformance documents. These reports, when complete, identify the nonconforming material, part or component; applicable inspection requirements; and the resolution, and approval thereof, of the nonconformance. Provisions are established for identifying those personnel with the responsibility and authority for approving the resolution of nonconformances. Until a determination of conformance is made, a nuclear safety related material, part or component cannot be placed in service. Tags which are placed on items to identify nonconformances are removed upon resolution.

Significant trends will be/are reported to appropriate levels of management.

D17.3.2.14 Document Control

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.2.15 Records

To the maximum extent practicable, records are stored such that they are protected from possible destruction by causes such as fire, flooding, theft, insects and rodents and from possible deterioration due to a combination of extreme variations in temperature and humidity conditions.

Record storage areas shall be evaluated by a Fire Protection Engineer (meeting Professional Member grade qualifications of the SFPE) to assure the records are adequately protected from damage. The evaluation shall include the following considerations as a minimum:

a) Structural collapse.
b) Unprotected steel (suspended floor slab or roof).
c) Fire frequency of similar occupancies.
d) Quantities of combustible materials.
e) Ceiling height/Room configuration which would contribute to heat dissipation.
f) Fire detection.
g) Fixed fire suppression systems.
h) On-site firefighting organizations including available equipment.

This evaluation shall be documented for each record storage area.

D17.3.3 SELF ASSESSMENT

D17.3.3.1 Methodology

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.2 Independent Review

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3 Independent Assessment

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.1 Organization

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.2 Internal Assessment Process

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.3 NOS Audit Program

The following audit topic is added to the list of audit topics in 17.3.3.3.3 for Catawba, McGuire, and Oconee:

- The performance of effluent and environmental monitoring activities.

D17.3.3.3.3.1 Other Reviews Prescribed by the Code of Federal Regulations

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.3.2 Independent Audit of Fire Protection Program

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.4 Results

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.5 Supplier Oversight

Supplier oversight assures that supplier QA programs provide for surveillance, evaluation, and approval of sub-supplier supplying items and services. This assurance is accomplished through one or more of the following: 1) reviewing supplier audits of sub-supplier as part of the pre-bid audit, 2) making supplier control of sub-supplier work a criterion for supplier approval or

disapproval, 3) making supplier surveillance of sub-supplier a requirement of the purchase requisition.

Supplier oversight performs source verification and audits on suppliers' QA programs including the activities of their suppliers and sub-suppliers, to assure that operations are in compliance with specified QA requirements. In the case of an audit of a supplier, any deficiencies noted by the auditor are clearly outlined in writing and given to the supplier's QA organization, which takes appropriate steps to resolve the deficiencies.

A re-audit is performed, if appropriate, to verify the implementation of the corrective action.

D17.3.3.3.6 Independent Audit of QA Functions

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.3.3.7 Audit Frequency Extensions

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4 ADMINISTRATIVE CONTROLS RELOCATED FROM TECHNICAL SPECIFICATIONS

Consistent with NRC Administrative Letter 95-06, certain administrative controls from the original station Technical Specifications have been relocated to the Quality Assurance Program. These relocated administrative controls include technical review, 10 CFR 50.59 review, record retention, and audit requirements. This section identifies those requirements or provides references to the sections of this document where the administrative controls have been integrated with QAP controls.

D17.3.4.1 Technical Reviews

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.2 10 CFR 50.59 Reviews

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.3 Record Retention

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.4 Audit Types and Frequencies

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.5 On-Site Review Committee

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.6 Reportable Event Action

There are no Duke Energy Carolinas specific amplifications for this section.

D17.3.4.7 Independent Safety Engineering Group Functions

Technical Specifications for Catawba and McGuire included requirements for Independent Safety Engineering Group functions of improving licensee safety performance and ability to respond to accidents by providing onsite technical support and continuous evaluation and feedback of lessons learned from operating experience. Those requirements were transferred to the this document at Amendment 23. At Amendment 36, the specific requirements for Independent Safety Engineering Group were eliminated based on duplication of functions performed by a combination of different groups through the performance of their normal activities.