Duke Energy
Enclosure 4
Levy Nuclear Plant Units 1 and 2

COLA Revisions

(70 pages including cover page)

Associated LNP COL Application Revisions:

The following are the revisions to the LNP Units 1 and 2 COLA based on the changes presented in the previous enclosures. These revisions will be made in a future update of the LNP COLA.

1. COLA Part 2, FSAR Chapter 1, Table 1.8-201, Summary of FSAR Departures from the DCD, will be revised to add the following departure:

Departure Number	Departure Description Summary	FSAR Section or Subsection
LNP DEP 6.4-1	The main control room habitability system design and operator dose evaluation has been revised. Shielding was added to control room VES filter, VBS signals were added, VES actuation setpoints were adjusted to meet design requirements and allowable secondary iodine activity level was lowered. The following are the departures from the DCD: Tier 1 Subsection 2.7.1, Tier 2 Table 1.6-1, Subsection 1.9.4.2.3, Appendix 1A, Subsection 3.1.2, Subsection 6.4, Subsection 6.4.2.6, Subsection 6.4.3.2, Subsection 6.4.4, Table 6.4-2, Subsection 7.3.1.2.17, Subsection 9.2.6.1.1, Subsection 9.4.1.1.1, Subsection 9.4.1.1.2, Subsection 9.4.1.2.1.1, Subsection 9.4.1.2.3.1, Figure 9.4.1-1 (Sheet 5 of 7), Table 11.1-4, Table 11.1-5, Table 11.1-6, Subsection 11.5.1.1, Subsection 11.5.2.3.1, Subsection 12.2.1.3.1, Subsection 12.2.1.3.2, Subsection 12.3.2.2.7, Table 12.2-28, Table 12.2-29, Figure 12.3-1 (Sheet 6 of 16), Table 14.3-7 (Sheet 2 of 3), Subsection 15.0.11.1, Subsection 15.0.11.6 (new), Table 15.0-2 (Sheet 4 of 5), Subsection 15.1.5.4.1, Subsection 15.1.5.4.6, Table 15.1.5-1, Subsection 15.3.3.3.1, Table 15.3-3 (Sheet 1 of 2), Subsection 15.4.8.1.1.3, Subsection 15.4.8.1.2, Subsection 15.4.8.2, Subsection 15.4.8.2.1, Subsection 15.4.8.2.1.1, Subsection 15.4.8.2.1.2, Subsection 15.4.8.2.1.3, Subsection 15.4.8.2.1.4, Subsection	Chapter 1 (Table of Contents) Table 1.6-202 1.9.4.2.3 Appendix 1AA Chapter 3 (Table of Contents) 3.1.2 Chapter 6 (Table of Contents, List of Tables) 6.4 6.4.2.6 6.4.3.2 6.4.4 Table 6.4-202 Chapter 7 (Table of Contents) 7.3.1.2.17 Chapter 9 (Table of Contents, List of Figures) 9.2.6.1.1 9.4.1.1.1 9.4.1.1.2 9.4.1.2.1.1 9.4.1.2.3.1 Figure 9.4-201 Chapter 11 (Table of Contents, List of Tables) Table 11.1-201 Table 11.1-202 Table 11.1-203 11.5.1.1 11.5.2.3.1 Chapter 12 (Table of Contents, List of Tables, List of Figures)

15.4.8.2.1.5, Subsection 15.4.8.2.1.7, Subsection 15.4.8.2.1.8, Subsection 15.4.8.2.1.9, Subsection 15.4.8.3, Subsection 15.4.8.3.1, Subsection 15.4.8.3.5, Subsection 15.4.8.3.6, Subsection 15.4.10, Table 15.4-1 (Sheet 2 of 3), Table 15.4-3 (deleted), Table 15.4-4 (Sheets 1 and 2 of 2), Figure 15.4.8-1, Figure 15.4.8-2, Figure 15.4.8-3, Figure 15.4.8-4, Subsection 15.6.5.3.5, Subsection 15.6.5.3.8.1, Subsection 15.6.5.3.8.2, Table 15.6.5-2 (Sheets 1-3 of 3), Table 15.6.5-3, Subsection 15.6.2.6, Subsection 15.6.3.3.1, Subsection 15.6.3.3.6, Table 15.6.2-1, Table 15.6.3-3, Subsection 15.7.4.5, Table 15.7-1, Subsection 15A.3.1.2, Subsection 15B.1, Chapter 16 LCO 3.7.4, SR 3.7.4.1, Bases 3.4.10, Bases 3.7.4, Bases 3.7.6.	12.2.1.3.1 12.2.1.3.2 Table 12.2-201 Table 12.2-202 12.3.2.2.7 Figure 12.3-201 Chapter 14 (Table of Contents, List of Tables) Table 14.3-203 Chapter 15 (Table of Contents, List of Tables, List of Figures) 15.0.11.1 15.0.11.6 Table 15.0-201 15.1.5.4.1 15.1.5.4.6 Table 15.1-201 15.3.3.3.1 Table 15.3-201 15.4.8.1.1.3 15.4.8.1.2 15.4.8.2 15.4.8.2.1 15.4.8.2.1.1 15.4.8.2.1.2 15.4.8.2.1.3 15.4.8.2.1.4 15.4.8.2.1.5 15.4.8.2.1.7 15.4.8.2.1.8 15.4.8.2.1.9 15.4.8.3 15.4.8.3.1 15.4.8.3.5 15.4.8.3.6 15.4.10 Table 15.4-201 Table 15.4-202 Figure 15.4-201 Figure 15.4-202 Figure 15.4-203 Figure 15.4-204 15.6.2.6 15.6.3.3.1 15.6.3.3.6 15.6.5.3.5 15.6.5.3.8.1

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15B.1
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16 SR 3.7.4.1
16 Bases 3.4.10
16 Bases 3.7.4
16 Bases 3.7.6

COLA Part 2, FSAR Sections 1.6, 1.9, Appendix 1AA, 3.1, 6.4, 7.3, 9.2, 9.4, 11.1, 11.5, 12.2, 12.3, 14.3, 15.0, 15.1, 15.3, 15.4, 15.6, 15.7, 15A, 15B and 16 will be revised to add departures from the DCD, with a LMA of LNP DEP 6.4-1 as presented below.

2. COLA Part 2, FSAR Chapter 1, will be revised to add a departure from DCD Table 1.6-1, Material Referenced (Sheet 15 of 21), as new FSAR Table 1.6-202, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 1. Table 1.6-202 is shown below:

Table 1.6-202

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
15.4	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-15806-P-A (P) WCAP-15807-NP-A	Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-16045-P-A (P) WCAP-16045-NP-A	Qualification of the Two-Dimensional Transport Code PARAGON, August 2004
	WCAP-10965-P-A, Addendum 1 (P) WCAP-10966-A Addendum 1	ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery, April 1989
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A, Revision 1 with Errata (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), July 2000
	WCAP-16045-P-A Addendum 1-A (P) WCAP-16045-NP-A	Qualification of the NEXUS Nuclear Data Methodology, August 2007

(P) Denotes Document is Proprietary

Addendum 1-A	
WCAP-10965-P-A Addendum 2-A (P)	Qualification of the New Pin Power Recovery Methodology, September 2010
WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April 1999

3. COLA Part 2, FSAR Chapter 1, will be revised to add the following to Subsection 1.9.4.2.3, with a LMA of LNP DEP 6.4-1:

   Revise the second sentence in the first paragraph of the AP1000 Response for Issue 83 in DCD Subsection 1.9.4.2.3 as follows:

   > If ac power is unavailable for more than 10 minutes or if "High-2" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES).

4. COLA Part 2, FSAR Appendix 1AA Regulatory Guide 1.52 Rev 3, 6/01 will be revised as follows, with a LMA of LNP DEP 6.4-1, to read:

   Conformance with the design and operational aspects is as stated in the DCD, with the exception of Criteria Section C.4.9 and Table 1. Conformance with Section C.4.9 and Table 1 is documented below.

C.4.9	Conforms
Table 1	Conforms

5. COLA Part 2, FSAR Chapter 3, will be revised to update Subsection 3.1, to read:

   3.1 Conformance With Nuclear Regulatory Commission General Design Criteria

   This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

6. COLA Part 2, FSAR Chapter 3, will be revised to add new Subsection 3.1.2, with a LMA of LNP DEP 6.4-1, to read:

   3.1.2 Protection by Multiple Fission Product Barriers

   Revise the first sentence of the third paragraph of AP1000 Compliance section of Criterion 19 - Control Room of DCD Subsection 3.1.2 to read as follows:

   If ac power is unavailable for more than 10 minutes or if "High-2" particulate, low pressurizer pressure is detected, or "High-2" iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES).

7. COLA Part 2, FSAR Chapter 6, will be revised to update Section 6.4, with a LMA of LNP DEP 6.4-1, to add the following before Subsection 6.4.3:

   6.4 Habitability Systems

   Revise the first sentence of the third paragraph of DCD Section 6.4 to read as follows:

   If ac power is unavailable for more than 10 minutes or if "High-2" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES).

8. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.4.2.6, with a LMA of LNP DEP 6.4-1, to read:

   6.4.2.6 Shielding Design

   Revise DCD Subsection 6.4.2.6 to read as follows:

   The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the main control room. Main control room shielding design bases are discussed in Section 12.3. In addition to shielding provided by building structural features, consideration is given to shielding provided by the VES filter shielding. Descriptions of the design basis LOCA source terms, main control room shielding parameters, and evaluation of doses to main control room personnel are presented in Section 15.6.

   The main control room and its location in the plant are shown in Figure 12.3-1.

9. COLA Part 2, FSAR Chapter 6, will be revised to add new Subsection 6.4.3.2, with a LMA of LNP DEP 6.4-1, to read:

   6.4.3.2 Emergency Mode

   Revise the first bullet of the first paragraph of DCD Subsection 6.4.3.2 to read as follows:

   - "High-2" particulate or iodine radioactivity in the main control room supply air duct

   Revise the first sentence of the second paragraph of DCD Subsection 6.4.3.2 to read as follows:

   The nuclear island nonradioactive ventilation system is isolated from the main control room pressure boundary by automatic closure of the isolation devices located in the nuclear island nonradioactive ventilation system ductwork if radiation levels in the main control room supply air duct exceed the "High-2" setpoint or if ac power is lost for more than 10 minutes.

10. COLA Part 2, FSAR Chapter 6, will be revised to update Subsection 6.4.4, with a LMA of LNP DEP 6.4-1, to read:

6.4.4 System Safety Evaluation

Revise the third paragraph of DCD Subsection 6.4.4 to read as follows:

Doses were determined for the following design basis:

	VES Operating	VBS Operating
Large Break LOCA	4.33 rem TEDE	4.84 rem TEDE
Fuel Handling Accident	1.5 rem TEDE	1.6 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	3.4 rem TEDE	4.0 rem TEDE
(Accident-initiated iodine spike)	1.0 rem TEDE	1.4 rem TEDE
Steam Line Break		
(Pre-existing iodine spike)	1.1 rem TEDE	0.9 rem TEDE
(Accident-initiated iodine spike)	1.3 rem TEDE	2.9 rem TEDE
Rod Ejection Accident	3.6 rem TEDE	2.5 rem TEDE
Locked Rotor Accident		
(Accident without feedwater available)	0.4 rem TEDE	0.7 rem TEDE
(Accident with feedwater available)	0.2 rem TEDE	0.9 rem TEDE
Small Line Break Outside Containment	0.4 rem TEDE	0.4 rem TEDE

Revise the first bullet of the thirteenth paragraph of DCD Subsection 6.4.4 to read as follows:

- "High-2" particulate or iodine radioactivity in MCR air supply duct

Revise the last sentence of the sixteenth paragraph of DCD Subsection 6.4.4 to read as follows:

The following cases are evaluated since they involve releases that extend beyond 24 hours after the initiation of the event:

Large Break LOCA	4.4 rem TEDE
Steam Line Break	
(Pre-existing iodine spike)	1.2 rem TEDE
(Accident-initiated iodine spike)	2.0 rem TEDE

11. COLA Part 2, FSAR Chapter 6, will be revised to add a departure from DCD Table 6.4-2, Main Control Room Habitability Indications and Alarms, as new FSAR Table 6.4-202, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 6. Table 6.4-202 is shown below:

Table 6.4-202 **MAIN CONTROL ROOM HABITABILITY INDICATIONS AND ALARMS**
VES emergency air storage tank pressure (indication and low and low-low alarms)
VES MCR pressure boundary differential pressure (indication and high and low alarms)
VES air delivery line flowrate (indication and high and low alarms)
VES passive filtration flow rate (indication and high and low alarms)
VBS main control room supply air radiation level (High-1 and-High-2 alarms)
VBS outside air intake smoke level (high alarm)
VBS isolation valve position
VBS MCR pressure boundary differential pressure

12. COLA Part 2, FSAR Chapter 7, will be revised to update Subsection 7.3, to read:

    7.3 Engineered Safety Features

    This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

13. COLA Part 2, FSAR Chapter 7, will be revised to add new Subsection 7.3.1.2.17, with a LMA of LNP DEP 6.4-1, to read:

    7.3.1.2.17 Main Control Room Isolation and Air Supply Initiation

    Revise the first sentence of the second paragraph of DCD Subsection 7.3.1.2.17 to read as follows:

    Condition 1 is the occurrence one of two main control room air supply radioactivity monitors detecting the iodine or particulate radioactivity level above the High-2 setpoint.

14. COLA Part 2, FSAR Chapter 9, will be revised to add new Subsection 9.2.6.1.1, with a LMA of LNP DEP 6.4-1, to read:

    9.2.6.1.1 Safety Design Basis

    Revise the first sentence of the first paragraph of DCD Subsection 9.2.6.1.1 to read as follows:

    The sanitary drainage system isolates the SDS vent penetration in the main control room boundary on High-2 particulate or iodine concentrations in the main control room air supply or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4.

15. COLA Part 2, FSAR Chapter 9, will be revised to add new Subsection 9.4.1.1.1, with a LMA of LNP DEP 6.4-1, to read:

9.4.1.1.1 Safety Design Basis

Revise the second bullet in the first paragraph of DCD Subsection 9.4.1.1.1 to read as follows:

- Isolates the HVAC penetrations in the main control room boundary on High-2 particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system as described in Section 6.4

16. COLA Part 2, FSAR Chapter 9, will be revised to add new Subsection 9.4.1.1.2, with a LMA of LNP DEP 6.4-1, to read:

9.4.1.1.2 Power Generation Design Basis

Revise the third bullet in the first paragraph of DCD Subsection 9.4.1.1.2 to read as follows:

- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides filtered outdoor air to pressurize the main control room and CSA areas to a positive pressure of at least 1/8 inch wg when a High-1 radioactivity concentration (gaseous, particulate, or iodine) is detected in the main control room supply air duct.

17. COLA Part 2, FSAR Chapter 9, will be revised to add new Subsection 9.4.1.2.1.1, with a LMA of LNP DEP 6.4-1, to read:

9.4.1.2.1.1 Main Control Room/Control Support Area HVAC Subsystem

Revise the second to last sentence of the second paragraph of DCD Subsection 9.4.1.2.1.1 to read as follows:

These monitors initiate operation of the nonsafety-related supplemental air filtration units on High-1 radioactivity concentrations (gaseous, particulate, or iodine) and isolate the main control room from the nuclear island nonradioactive ventilation system on High-2 particulate or iodine radioactivity concentrations.

18. COLA Part 2, FSAR Chapter 9, will be revised to add new Subsection 9.4.1.2.3.1, with a LMA of LNP DEP 6.4-1, to read:

9.4.1.2.3.1 Main Control Room/Control Support Area HVAC Subsystem

Revise the second and third sentences of the first paragraph of the Abnormal Plant Operation section of DCD Subsection 9.4.1.2.3.1 to read as follows:

The first is "High-1" radioactivity based upon radioactivity instrumentation (gaseous, particulate, or iodine). The second is "High-2" radioactivity based upon either particulate or iodine radioactivity instruments.

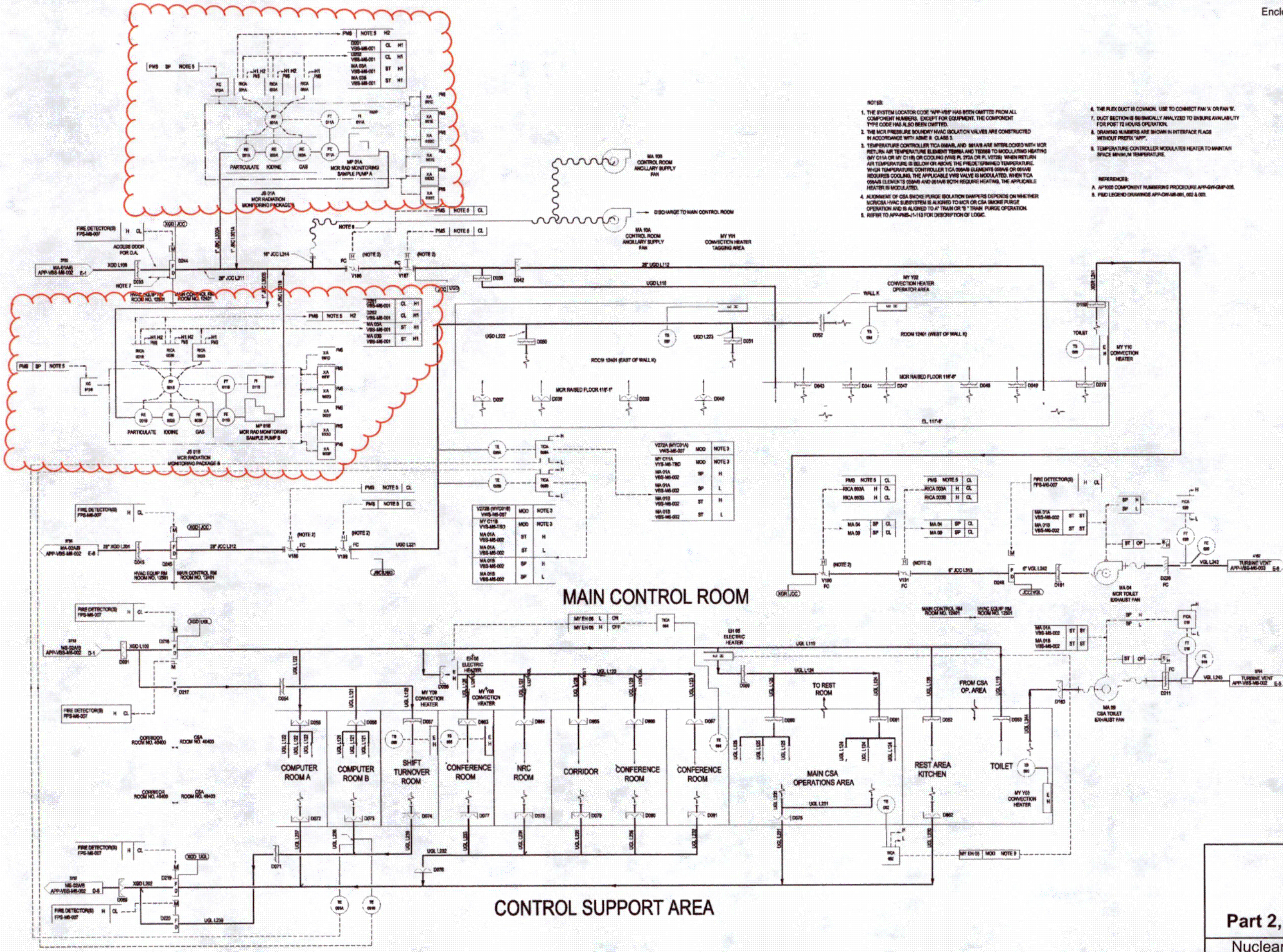
Revise the first sentence of the second paragraph of the Abnormal Plant Operation section of DCD Subsection 9.4.1.2.3.1 to read as follows:

If "High-1" ~~gaseous~~ radioactivity is detected in the main control room supply air duct and the main control room/control support area HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the main control room and CSA areas to at least 1/8 inch wg with respect to the surrounding areas and the outside environment using filtered makeup air.

Revise the first sentence of the third paragraph of the Abnormal Plant Operation section of DCD Subsection 9.4.1.2.3.1 to read as follows:

If ac power is unavailable for more than 10 minutes or if "High-2" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding GDC-19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room from the normal main control room/control support area HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves.

19. COLA Part 2 FSAR Chapter 9 will be revised to add a departure from DCD Figure 9.4.1-1 (Sheet 5 of 7), Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram, as new FSAR Figure 9.4-201, with a LMA of LNP DEP 6.4-1. This figure will also be added to the list of figures from Chapter 9. Figure 9.4-201 is shown below:



Duke Energy Florida
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Nuclear Island Non-Radioactive Ventilation System Piping and Instrumentation Diagram
FIGURE 9.4-201

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20. COLA Part 2, FSAR Chapter 11, will be revised to update Subsection 11.1, to read:

11.1 Source Terms

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

21. COLA Part 2, FSAR Chapter 11, will be revised to add a departure from DCD Table 11.1-4, Parameters Used To Calculate Secondary Coolant Activity, as new FSAR Table 11.1-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 11. Table 11.1-201 is shown below:

Table 11.1-201

PARAMETERS USED TO CALCULATE SECONDARY COOLANT ACTIVITY

Total secondary side water mass (lb/steam generator)	1.68×10^5
Steam generator steam fraction	0.058
Total steam flow rate (lb/hr)	1.5×10^7
Moisture carryover (percent)	0.1
Total makeup water feed rate (lb/hr)	700
Total blowdown rate (gpm)	186
Total primary-to-secondary leak rate (gpd)	300
Iodine partition factor (mass basis)	100

22. COLA Part 2, FSAR Chapter 11, will be revised to add a departure from DCD Table 11.1-5, Design Basis Steam Generator Secondary Side Liquid Activity, as new FSAR Table 11.1-202, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 11. Table 11.1-202 is shown below:

Table 11.1-202

DESIGN BASIS STEAM GENERATOR SECONDARY SIDE LIQUID ACTIVITY

Nuclide	**Activity (μCi/g)**	**Nuclide**	**Activity (μCi/g)**
Br-83	1.4×10^{-5}	Y-93	8.2×10^{-8}
Br-84	2.4×10^{-6}	Zr-95	1.5×10^{-7}
Br-85	3.1×10^{-8}	Nb-95	1.5×10^{-7}
I-129	1.3×10^{-11}	Mo-99	1.9×10^{-4}
I-130	7.9×10^{-6}	Tc-99m	1.7×10^{-4}
I-131	6.3×10^{-4}	Ru-103	1.2×10^{-7}

I-132	4.2×10^{-4}	Ru-106	4.1×10^{-8}
I-133	1.0×10^{-3}	Rh-103m	1.2×10^{-7}
I-134	4.9×10^{-5}	Rh-106	4.1×10^{-8}
I-135	5.0×10^{-4}	Ag-110m	3.0×10^{-6}
Rb-86	1.4×10^{-5}	Te-125m	1.5×10^{-7}
Rb-88	1.4×10^{-4}	Te-127m	7.0×10^{-7}
Rb-89	5.6×10^{-6}	Te-127	2.2×10^{-6}
Cs-134	1.1×10^{-3}	Te-129m	2.4×10^{-6}
Cs-136	1.7×10^{-3}	Te-129	2.1×10^{-6}
Cs-137	8.2×10^{-4}	Te-131m	5.6×10^{-6}
Cs-138	5.9×10^{-5}	Te-131	1.6×10^{-6}
H-3	3.8×10^{-1}	Te-132	7.0×10^{-5}
Cr-51	1.3×10^{-6}	Te-134	2.0×10^{-6}
Mn-54	6.6×10^{-7}	Ba-137m	7.7×10^{-4}
Mn-56	7.8×10^{-5}	Ba-140	9.4×10^{-7}
Fe-55	5.0×10^{-7}	La-140	3.3×10^{-7}
Fe-59	1.3×10^{-7}	Ce-141	1.4×10^{-7}
Co-58	1.9×10^{-6}	Ce-143	1.2×10^{-7}
Co-60	2.2×10^{-7}	Ce-144	1.1×10^{-7}
Sr-89	1.8×10^{-6}	Pr-143	1.4×10^{-7}
Sr-90	8.0×10^{-8}	Pr-144	1.1×10^{-7}
Sr-91	1.9×10^{-6}		
Sr-92	2.4×10^{-7}		
Y-90	1.4×10^{-8}		
Y-91m	1.0×10^{-6}		
Y-91	1.3×10^{-7}		
Y-92	2.8×10^{-7}		

23. COLA Part 2, FSAR Chapter 11, will be revised to add a departure from DCD Table 11.1-6, Design Basis Steam Generator Secondary Side Steam Activity, as new FSAR Table 11.1-203, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 11. Table 11.1-203 is shown below:

Table 11.1-203

DESIGN BASIS STEAM GENERATOR SECONDARY SIDE STEAM ACTIVITY

Nuclide	**Activity (μCi/g)**
Kr-83m	1.1×10^{-6}
Kr-85m	4.3×10^{-6}

Kr-85	1.5×10^{-5}
Kr-87	2.4×10^{-6}
Kr-88	7.7×10^{-6}
Kr-89	1.8×10^{-7}
Xe-131m	6.9×10^{-6}
Xe-133m	8.7×10^{-6}
Xe-133	6.4×10^{-4}
Xe-135m	5.5×10^{-6}
Xe-135	1.9×10^{-5}
Xe-137	3.4×10^{-7}
Xe-138	1.3×10^{-6}
I-129	1.5×10^{-13}
I-130	8.7×10^{-8}
I-131	6.9×10^{-6}
I-132	4.7×10^{-6}
I-133	1.1×10^{-5}
I-134	5.4×10^{-7}
I-135	5.5×10^{-6}
H-3	3.8×10^{-1}

24. COLA Part 2, FSAR Chapter 11, will be revised to add new Subsection 11.5.1.1, with a LMA of LNP DEP 6.4-1, to read:

11.5.1.1 Safety Design Basis

Revise the third and fourth bullets in the third paragraph of DCD Subsection 11.5.1.1 to read as follows:

- Initiate main control room supplemental filtration in the event of abnormally high particulate, iodine, or gaseous radioactivity in the main control room supply air (High-1)

- Initiate main control room ventilation isolation and actuate the main control room emergency habitability system in the event of abnormally high particulate or iodine radioactivity in the main control room supply air (High-2)

25. COLA Part 2, FSAR Chapter 11, will be revised to add new Subsection 11.5.2.3.1, with a LMA of LNP DEP 6.4-1, to read:

11.5.2.3.1 Fluid Process Monitors

Revise the second to last sentence of the first paragraph of the Main Control Room Supply Air Duct Radiation Monitors section of DCD Subsection 11.5.2.3.1 to read as follows:

When predetermined setpoints are exceeded, the monitors provide signals to initiate the supplemental air filtration system on a High-1 gaseous, particulate, or iodine concentration, and to isolate the main control room air intake and exhaust ducts and activate the main control room emergency habitability system on High-2 particulate or iodine concentrations.

26. COLA Part 2, FSAR Chapter 12, will be revised to add new Subsection 12.2.1.3, with a LMA of LNP DEP 6.4-1, to read:

12.2.1.3 Sources for the Core Melt Accident

Revise the last paragraph of DCD Subsection 12.2.1.3 to read as follows:

12.2.1.3.1 Containment

If there is core degradation, core cooling would be provided by the passive core cooling system which is totally inside the containment such that no high activity sump solution would be recirculated outside the containment. The shielding provided for the containment addresses this post-LOCA source term. The source strengths as a function of time are provided in Table 12.2-20 and the integrated source strengths are provided in Table 12.2-21.

12.2.1.3.2 Main Control Room HVAC Filters

During operation of the nuclear island nonradioactive ventilation system (VBS) supplemental filtration or the main control room emergency habitability system (VES), filters in the control room HVAC work to remove particulate and iodine from the air. As radioactivity accumulates within the filters, this becomes a potential source of dose. These source strengths as a function of time are provided in Table 12.2-28 and the integrated source strengths are provided in Table 12.2-29.

27. COLA Part 2, FSAR Chapter 12, will be revised to add a departure from DCD Subsection 12.2 to add new DCD Table 12.2-28, Core Melt Accident Source Strengths From MCR HVAC Filters as a Function of Time (Sheets 1 and 2), as new FSAR Table 12.2-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 12. Table 12.2-201 is shown below:

TABLE 12.2-201 (SHEET 1 OF 2)

CORE MELT ACCIDENT SOURCE STRENGTHS FROM MCR HVAC FILTERS AS A FUNCTION OF TIME

VES Filter (1) Source Strengths after a Loss of Coolant Accident

Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01-0.02	1.19E+06	3.11E+06	1.81E+06	1.97E+05
0.02-0.03	1.47E+06	5.26E+06	3.89E+06	2.65E+05
0.03-0.06	2.87E+06	8.30E+06	5.46E+06	6.74E+05
0.06-0.1	3.03E+06	8.13E+06	5.22E+06	5.41E+05
0.1-0.2	5.76E+06	1.41E+07	8.76E+06	9.02E+05
0.2-0.4	6.14E+07	2.61E+08	2.46E+08	1.87E+07
0.4-0.6	1.86E+08	6.02E+08	3.60E+08	1.83E+07
0.6-0.7	1.47E+08	2.33E+08	1.47E+08	1.03E+08
0.7-0.8	1.09E+08	1.80E+08	1.05E+08	7.30E+07
0.8-1.0	1.85E+08	1.67E+08	6.99E+07	7.13E+06
1.0-1.5	3.36E+08	6.99E+08	1.85E+08	1.22E+07
1.5-2.0	1.21E+08	2.55E+08	4.97E+07	2.69E+04
2.0-3.0	3.13E+07	3.87E+07	7.28E+06	9.07E+03
3.0-4.0	3.68E+05	5.98E+03	5.56E+02	1.41E+02
4.0-5.0	1.42E+04	3.16E+01	8.55E-04	7.80E-04
5.0-6.0	3.31E-05	3.12E-04	3.35E-04	3.21E-04
6.0-7.0	1.32E-05	1.24E-04	1.33E-04	1.28E-04
7.0-8.0	5.11E-06	4.82E-05	5.17E-05	4.96E-05
8.0-10.0	2.68E-06	2.53E-05	2.71E-05	2.60E-05
10.0-14.0	1.69E-07	1.60E-06	1.71E-06	1.64E-06
Total	1.19E+09	2.47E+09	1.19E+09	2.35E+08

Notes: 1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.

TABLE 12.2-201 (SHEET 2 OF 2)

CORE MELT ACCIDENT SOURCE STRENGTHS FROM MCR HVAC FILTERS AS A FUNCTION OF TIME

VBS Filter (2) Source Strengths after a Loss of Coolant Accident

Energy Group (Mev/gamma)	Source Strength (Mev/sec)			
	2 hours	8 hours	24 hours	30 days
0.01-0.02	6.86E+08	1.00E+09	5.75E+08	6.21E+07
0.02-0.03	9.55E+08	1.76E+09	1.27E+09	8.46E+07
0.03-0.06	1.71E+09	2.71E+09	1.75E+09	2.10E+08
0.06-0.1	1.72E+09	2.60E+09	1.63E+09	1.70E+08
0.1-0.2	3.49E+09	4.61E+09	2.81E+09	2.91E+08
0.2-0.4	3.54E+10	8.45E+10	7.59E+10	5.76E+09
0.4-0.6	1.03E+11	1.91E+11	1.10E+11	5.61E+09
0.6-0.7	7.99E+10	7.20E+10	4.39E+10	3.04E+10
0.7-0.8	5.97E+10	5.62E+10	3.17E+10	2.16E+10
0.8-1.0	1.03E+11	5.23E+10	2.13E+10	2.11E+09
1.0-1.5	1.86E+11	2.20E+11	5.64E+10	3.62E+09
1.5-2.0	6.71E+10	8.03E+10	1.53E+10	8.78E+06
2.0-3.0	1.66E+10	1.22E+10	2.24E+09	3.09E+06
3.0-4.0	1.82E+08	1.93E+06	1.89E+05	4.81E+04
4.0-5.0	6.86E+06	7.65E+03	2.91E-01	2.65E-01
5.0-6.0	3.74E-02	1.12E-01	1.14E-01	1.09E-01
6.0-7.0	1.49E-02	4.47E-02	4.54E-02	4.35E-02
7.0-8.0	5.78E-03	1.74E-02	1.76E-02	1.69E-02
8.0-10.0	3.03E-03	9.11E-03	9.24E-03	8.86E-03
10.0-14.0	1.92E-04	5.75E-04	5.84E-04	5.60E-04
Total	6.59E+11	7.82E+11	3.65E+11	7.00E+10

Notes: 2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

28. COLA Part 2, FSAR Chapter 12, will be revised to add a departure from DCD Subsection 12.2 to add new DCD Table 12.2-29, Core Melt Accident Integrated Source Strengths From MCR HVAC Filters, as new FSAR Table 12.2-202, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 12. Table 12.2-202 is shown below:

TABLE 12.2-202

CORE MELT ACCIDENT INTEGRATED SOURCE STRENGTHS FROM MCR HVAC FILTERS

Energy Group (Mev/gamma)	**30-Day Integrated Source Strength (Mev)**	
	VES (1)	**VBS (2)**
0.01-0.02	1.75E+08	5.65E+10
0.02-0.03	3.81E+08	1.26E+11
0.03-0.06	5.89E+08	1.90E+11
0.06-0.1	5.77E+08	1.84E+11
0.1-0.2	9.03E+08	2.95E+11
0.2-0.4	3.34E+10	1.05E+13
0.4-0.6	2.36E+10	7.44E+12
0.6-0.7	3.81E+10	1.15E+13
0.7-0.8	2.63E+10	7.92E+12
0.8-1.0	7.57E+09	2.39E+12
1.0-1.5	1.77E+10	5.67E+12
1.5-2.0	4.03E+09	1.34E+12
2.0-3.0	6.47E+08	2.18E+11
3.0-4.0	1.20E+06	4.46E+08
4.0-5.0	4.17E+04	1.52E+07
5.0-6.0	1.03E-01	3.51E+01
6.0-7.0	4.08E-02	1.40E+01
7.0-8.0	1.59E-02	5.42E+00
8.0-10.0	8.32E-03	2.84E+00
10.0-14.0	5.25E-04	1.80E-01
Total	1.54E+11	4.79E+13

Notes:1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.
2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

29. COLA Part 2, FSAR Chapter 12, will be revised to add new Subsection 12.3.2.2.7, with a LMA of LNP DEP 6.4-1, to read:

12.3.2.2.7 Control Room Shielding Design

Revise DCD Subsection 12.3.2.2.7 to read as follows:

The design basis loss-of-coolant accident dictates the shielding requirements for the control room. The rod ejection accident dictates the shielding requirements for the main control room emergency habitability (VES) filter in the operator break room. Consideration is given to shielding provided by the shield building structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criterion 19.

Shielding of the VES filtration unit is accomplished by safety-related metal shielding. This shielding is composed of either tungsten that is 0.25 inches thick or stainless steel shown to provide an equivalent amount of shielding. The length and width of the shielding are designed to match the length and width of the filtration unit being shielded.

30. COLA Part 2, FSAR Chapter 12, will be revised to add a departure from DCD Figure 12.3-1, Radiation Zones, Normal Operation/Shutdown, Nuclear Island, Elevation 100′-0″ & 107′-2″ (Sheet 6 of 16), with a LMA of LNP DEP 6.4-1:

Note number 9 of DCD Figure 12.3-1 will be revised as follows as new FSAR Figure 12.3-201:

9. Blowdown Piping May Reach Zone III Levels With Concurrent Fuel Cladding Defects of 0.25% and Steam Generator Tube Leakage of 300 gpd.

31. COLA Part 2, FSAR Chapter 14, will be revised to add a departure from DCD Table 14.3-7, Radiological Analysis (Sheet 2 of 3), as new FSAR Table 14.3-203, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 14. Table 14.3-203 is shown below:

Table 14.3-203

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Section 8.3.1.1.6	Electrical penetrations through the containment can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration.	

Section	9.4.1.1.1	The VBS isolates the HVAC ductwork that penetrates the main control room boundary on High-2 particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system.	
Section	12.3.2.2.1	During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor components.	
Section	12.3.2.2.2	The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps and steam generators. Extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels.	
Section	12.3.2.2.3	Shielding is provided for the liquid radwaste, gaseous radwaste and spent resin handling systems consistent with the maximum postulated activity. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with radiation zoning requirements of adjacent areas.	

32. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.11.1, with a LMA of LNP DEP 6.4-1, to read:

15.0.11.1 FACTRAN Computer Code

Revise the first bullet of DCD Subsection 15.0.11.1 to read as follows:

- A sufficiently large number of radial space increments to handle fast transients ~~such as rod ejection accidents~~

33. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.0.11.6, with a LMA of LNP DEP 6.4-1, to read:

15.0.11.6 ANC Computer Code

Add new DCD Subsection 15.0.11.6 to read as follows:

The ANC computer code is used to solve the two-group neutron diffusion equation in three spatial dimensions. ANC can also solve the three-dimensional kinetics equations for six delayed neutron groups.

34. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.0-2, Summary of Initial Conditions and Computer Codes Used (Sheet 4 of 5), as new FSAR Table 15.0-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.0-201 is shown below:

Table 15.0-201

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.4	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	NA	0 and 3415
	Inadvertent loading and operation of a fuel assembly in an improper position	ANC	NA	–	NA	3415
	Spectrum of RCCA ejection accidents	ANC, VIPRE	Refer to subsection 15.4.8	Refer to subsection 15.4.8	Refer to subsection 15.4.8	Refer to subsection 15.4.8
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	0.0	–	Upper curve of Figure 15.04-1	3483.3 (a)
	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.0	–	Upper curve of Figure 15.04-1	3483.3 (a)

35. COLA Part 2, FSAR Chapter 15, will be revised to update Subsection 15.1, to read:

15.1 Increase in Heat Removal From the Primary System

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

36. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.1.5.4.1, with a LMA of LNP DEP 6.4-1, to read:

15.1.5.4.1 Source Term

Revise the fourth paragraph of DCD Subsection 15.1.5.4.1 to read as follows:

The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The reactor coolant alkali metal concentrations are based on those associated with the design basis fuel defect level.

Revise the last paragraph of DCD Subsection 15.1.5.4.1 to read as follows:

The secondary coolant is assumed to have an iodine source term of 0.01 μCi/g dose equivalent I-131. This is 1 percent of the maximum primary coolant activity at equilibrium operating conditions. The secondary coolant alkali metal concentration is also assumed to be 1 percent of the primary concentration.

37. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.1.5.4.6, with a LMA of LNP DEP 6.4-1, to read:

15.1.5.4.6 Doses

Revise the text of DCD Subsection 15.1.5.4.6 to read as follows:

Using the assumptions from Table 15.1.5-1, the calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.6 rem at the site boundary for the limiting 2-hour interval (4.8 to 6.8 hours) and 1.1 rem at the low population zone outer boundary. These doses are small fractions of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being 10 percent or less. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.5 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.4 rem at the low population zone outer boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

At the time the main steam line break occurs, the potential exists for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. The 30-day contribution to the dose at the site boundary and the low population zone boundary is less than 0.01 rem TEDE. When this is added to the dose calculated for the main steam line break, the resulting total dose remains less than the values reported above.

38. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.1.5-1, Parameters Used In Evaluating The Radiological Consequences Of A Main Steam Line Break,

as new FSAR Table 15.1-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.1-201 is shown below:

Table 15.1-201

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE BREAK

Reactor coolant iodine activity	
– Accident-initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Appendix 15A). Duration of spike is 5 hours.
– Preaccident spike	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Duration of accident (hr)	72
Atmospheric dispersion (χ/Q) factors	See Table 15A-5 in Appendix 15A
Steam generator in faulted loop	
– Initial water mass (lb)	3.32 E+05
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321E+05 3.66 E+03
Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	52.25[a]
– Iodine partition coefficient	1.0
– Steam released (lb) 0 - 2 hr 2 - 72 hr	 3.321E+05 3.66 E+03
Nuclide data	See Table 15A-4

Note: a. Equivalent to 150 gpd cooled liquid at 62.4 lb/ft^3.

39. COLA Part 2, FSAR Chapter 15, will be revised to update Subsection 15.3, to read:

15.3 Decrease in Reactor Coolant System Flow Rate

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

40. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.3.3.3.1, with a LMA of LNP DEP 6.4-1, to read:

15.3.3.3.1 Source Term

Revise the last paragraph of DCD Subsection 15.3.3.3.1 to read as follows:

The initial secondary coolant activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

41. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.3-3, Parameters Used In Evaluating The Radiological Consequences Of A Locked Rotor Accident (Sheet 1 of 2), as new FSAR Table 15.3-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.3-201 is shown below:

Table 15.3-201

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.10
Core activity	See Table 15A-3
Radial peaking factor (for determination of activity in failed fuel rods)	1.75
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.04 E+05

Table 15.3-201

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Condenser	Not available
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	104.5(b)
Partition coefficient in steam generators iodine alkali metals	 0.01 0.0035
Accident scenario in which startup feedwater is not available Duration of accident (hr) Steam released (lb) 0-1.5 hours(c) Leak flashing fraction(d) 0-60 minutes > 60 minutes	 1.5 hr 6.48 E+05 0.04 0

42. COLA Part 2, FSAR Chapter 15, will be revised to update Subsection 15.4, to read:

15.4 Reactivity and Power Distribution Anomalies

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

43. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.1.1.3, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.1.1.3 Reactor Protection

Revise DCD Subsection 15.4.8.1.1.3 to read as follows:

The reactor protection in the event of a rod ejection accident is described in WCAP-15806 P-A (Reference 4). The protection for this accident is provided by the high neutron flux trip (high and low setting) and the high rate of neutron flux increase trip. These protection functions are described in Section 7.2.

44. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.1.2, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.1.2 Limiting Criteria

Revise DCD Subsection 15.4.8.1.2 to read as follows:

This event is a Condition IV incident (ANSI N18.2). See subsection 15.0.1 for a discussion of ANS classification. Because of the extremely low probability of an RCCA ejection accident, some fuel damage is considered an acceptable consequence.

NUREG-0800 Standard Review Plan (SRP) 4.2, Revision 3 (Reference 24), interim criteria applicable to new plant design certification are applied to provide confidence that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are the following:

- The pellet clad mechanical interaction (PCMI) failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B 1 of SRP 4.2, Revision 3, Appendix B.
- The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure.
- For intermediate (greater than 5 percent rated thermal power) and full power conditions, fuel cladding is presumed to fail if local heat flux exceeds thermal design limits (e.g., DNBR).
- For core coolability, it is conservatively assumed that the average fuel pellet enthalpy at the hot spot remains below 200 cal/g (360 Btu/lb) for irradiated fuel. This bounds non-irradiated fuel, which has a slightly higher enthalpy limit.
- For core coolability, the peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel to coolant interaction and (2) fuel rod burst that must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.
- Peak reactor coolant system pressure is less than that which could cause stresses to exceed the "Service Limit C" as defined in the ASME code.

45. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2 Analysis of Effects and Consequences

Revise DCD Subsection 15.4.8.2 to read as follows:

Method of Analysis

The calculation of the RCCA ejection transients is performed in two stages: first, an average core calculation and then, a hot rod calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time, including the various total core feedback effects (Doppler reactivity and moderator reactivity). Enthalpy, fuel temperature, and DNB transients are then determined by performing a conservative fuel rod transient heat transfer calculation.

A discussion of the method of analysis appears in WCAP-15806-P-A (Reference 4).

Average Core Analysis

The three-dimensional nodal code ANC (References 14, 15, 16, 17, 21, 22, and 27) is used for the average core transient analysis. This code solves the two-group neutron diffusion theory kinetic equation in three spatial dimensions (rectangular coordinates) for six delayed neutron groups. The core moderator and fuel temperature feedbacks are based on the NRC approved Westinghouse version of the VIPRE 01 code and methods (References 18 and 19).

Hot Rod Analysis

The hot fuel rod models are based on the Westinghouse VIPRE models described in WCAP-15806-P-A (Reference 4). The hot rod model represents the hottest fuel rod from any channel in the core. VIPRE performs the hot rod transients for fuel enthalpy, temperature, and DNBR using as input the time dependent nuclear core power and power distribution from the core average analysis. A description of the VIPRE code is provided in Reference 18.

System Overpressure Analysis

If the fuel coolability limits are not exceeded, the fuel dispersal into the coolant or a sudden pressure increase from thermal to kinetic energy conversion is not needed to be considered in the overpressure analysis. Therefore, the overpressure condition may be calculated on the basis of conventional fuel rod to coolant heat transfer and the prompt heat generation in the coolant. The system overpressure analysis is conducted by first performing the core power response analysis to obtain the nuclear power transient (versus time) data. The nuclear power data is then used as input to a plant transient computer code to calculate the peak reactor coolant system pressure.

This code calculates the pressure transient, taking into account fluid transport in the reactor coolant system and heat transfer to the steam generators. For conservatism, no credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

46. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1, with a LMA of LNP DEP 6.4-1, to read:

    15.4.8.2.1 Calculation of Basic Parameters

    Revise DCD Subsection 15.4.8.2.1 to read as follows:

    Input parameters for the analysis are conservatively selected as described in Reference 4. DCD Table 15.4-3 is deleted and not used.

47. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.1, with a LMA of LNP DEP 6.4-1, to read:

    15.4.8.2.1.1 Ejected Rod Worths and Hot Channel Factors

    Revise DCD Subsection 15.4.8.2.1.1 to read as follows:

    The values for ejected rod worths and hot channel factors are calculated using three dimensional static methods. Standard nuclear design codes are used in the analysis. The calculation is performed for the maximum allowed bank insertion at a given power level, as determined by the rod insertion limits. Adverse xenon distributions are considered in the calculation.

Appropriate safety analysis allowances are added to the ejected rod worth and hot channel factors to account for calculational uncertainties, including an allowance for nuclear peaking due to densification as discussed in Reference 4.

48. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.2, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2.1.2 Reactivity Feedback Weighting Factors

Revise DCD Subsection 15.4.8.2.1.2 to read as follows:

15.4.8.2.1.2 Not Used

49. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.3, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2.1.3 Moderator and Doppler Coefficients

Revise DCD Subsection 15.4.8.2.1.3 to read as follows:

The critical boron concentration is adjusted in the nuclear code to obtain a moderator temperature coefficient that is conservative compared to actual design conditions for the plant consistent with Reference 4. The fuel temperature feedback in the neutronics code is reduced consistent with Reference 4 requirements.

50. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.4, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2.1.4 Delayed Neutron Fraction, β_{eff}

Revise DCD Subsection 15.4.8.2.1.4 to read as follows:

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values no less than 0.50 percent at end of cycle. The accident is sensitive to $\boldsymbol{\beta}$eff if the ejected rod worth is equal to or greater than $\boldsymbol{\beta}$eff. To allow for future cycles, a pessimistic estimate of $\boldsymbol{\beta}$eff of 0.44 percent is used in the analysis.

51. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.5, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2.1.5 Trip Reactivity Insertion

Revise the first paragraph of DCD Subsection 15.4.8.2.1.5 to read as follows:

The trip reactivity insertion accounts for the effect of the ejected rod and one adjacent stuck rod. The trip reactivity is simulated by dropping a limited set of rods of the required worth into the core. The start of rod motion occurs 0.9 second after the high neutron flux trip setpoint is reached. This delay is assumed to consist of 0.583 second for the instrument channel to produce a signal, 0.167 second for the trip breakers to open, and 0.15 second for the coil to release the rods. A curve of trip rod insertion versus time is used, which assumes that insertion to the dashpot does not occur until 2.47 seconds after the start of fall. The choice of such a conservative insertion rate means that there is over 1 second after the trip setpoint is reached before significant shutdown reactivity is inserted into the core. This conservatism is important for the hot full power accidents.

52. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.7, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.2.1.7 Results

Revise DCD Subsection 15.4.8.2.1.7 to read as follows:

For all cases, the core is preconditioned by assuming a fuel cycle depletion with control rod insertion that is conservative relative to expected baseload operation. All cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event and xenon is skewed to yield a conservative initial axial power shape. The limiting RCCA ejection cases for a typical cycle are summarized following the criteria outlined in subsection 15.4.8.1.2.

- PCMI and high cladding temperature (hot zero power)

  The resulting maximum fuel average enthalpy rise and maximum fuel average enthalpy are less than the criteria given in subsection 15.4.8.1.2.

- High cladding temperature (≥ 5% rated thermal power)

  The fraction of the core calculated to have a DNBR less than the safety analysis limit is less than the amount of failed fuel assumed in the dose analysis described in subsection 15.4.8.3.

- Core coolability

  The resulting maximum fuel average enthalpy is less than the criterion given in subsection 15.4.8.1.2. Fuel melting is not predicted to occur at the hot spot.

There are no fuel failures due to the fuel enthalpy deposition, i.e., both fuel and cladding enthalpy limits were met. Additionally, the coolability criteria for peak fuel enthalpy and the fuel melting criteria were met. Therefore, the fuel dispersal into the coolant, a sudden pressure increase from thermal to kinetic energy conversion, gross lattice distortion, or severe shock waves are precluded.

The nuclear power and fuel transients for the limiting cases are presented in Figures 15.4.8-1 through 15.4.8-3.

The calculated sequence of events for the limiting cases is presented in Table 15.4-1. Reactor trip occurs early in the transients, after which the nuclear power excursion is terminated.

The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. The effects and consequences of loss-of-coolant accidents

(LOCAs) are discussed in subsection 15.6.5. Following the RCCA ejection, the plant response is the same as a LOCA.

The consequential loss of offsite power described in subsection 15.0.14 is not limiting for the enthalpy and temperature transients resulting from an RCCA ejection accident. Due to the delay from reactor trip until turbine trip and the rapid power reduction produced by the reactor trip, the peak fuel and cladding temperatures occur before the reactor coolant pumps begin to coast down.

53. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.8, with a LMA of LNP DEP 6.4-1, to read:

    15.4.8.2.1.8 Fission Product Release

    Revise the first paragraph of DCD Subsection 15.4.8.2.1.8 to read as follows:

    It is assumed that fission products are released from the gaps of all rods entering DNB. In the cases considered, less than 10 percent of the rods are assumed to enter DNB based on a detailed three-dimensional kinetics and hot rod analysis. The maximum fuel average enthalpy rise of rods predicted to enter DNB will be less than 60 cal/g. Fuel melting does not occur at the hot spot.

54. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.2.1.9, with a LMA of LNP DEP 6.4-1, to read:

    15.4.8.2.1.9 Peak Reactor Coolant System Pressure

    Revise DCD Subsection 15.4.8.2.1.9 to read as follows:

    Calculations of the peak reactor coolant system pressure demonstrate that the peak pressure does not exceed that which would cause the stress to exceed the Service Level C Limit as described in the ASME Code, Section III. Therefore, the accident for this plant does not result in an excessive pressure rise or further damage to the reactor coolant system.

55. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.3, with a LMA of LNP DEP 6.4-1, to read:

    15.4.8.3 Radiological Consequences

    Revise the first two paragraphs of DCD Subsection 15.4.8.3 to read as follows:

    The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with a limited number of fuel rods containing cladding defects and that leaking steam generator tubes result in a buildup of activity in the secondary coolant. See subsection 15.4.8.3.1 and Table 15.4-4.

    As a result of the accident, 10 percent of the fuel rods are assumed to be damaged (see subsection 15.4.8.2.1.8) such that the activity contained in the fuel cladding gap is released to the reactor coolant. No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

56. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.3.1, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.3.1 Source Term

Revise DCD Subsection 15.4.8.3.1 to read as follows:

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals, and noble gases. The reactor coolant iodine source term assumes a pre existing iodine spike. The reactor coolant noble gas concentrations are assumed to be those associated with equilibrium operating limits for primary coolant noble gas activity. The initial reactor coolant alkali metal concentrations are assumed to be those associated with the design fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

Based on NUREG 1465 (Reference 12), the fission product gap fraction is 3 percent of fuel inventory. For this analysis, the gap fractions are modified following the guidance of Draft Guide 1199 (Reference 25), which incorporates the effects of enthalpy rise in the fuel following the reactivity insertion, consistent with Appendix B of SRP 4.2, Revision 3 (Reference 24). Draft Guide 1199 included expanded guidance for determining nuclide gap fractions available for release following a rod ejection. Reference 26 was issued as a clarification to the gap fraction guidance in Draft Guide 1199. An enthalpy rise of 60 cal/gm is used to calculate the gap fractions (see subsection 15.4.8.2.1.8). Also, to address the fact that the failed fuel rods may have been operating at power levels above the core average, the source term is increased by the lead rod radial peaking factor. No fuel melt is calculated to occur as a result of the rod ejection (see subsection 15.4.8.2.1.8).

57. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.3.5, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.3.5 Identification of Conservatisms

Revise second bullet of DCD Subsection 15.4.8.3.5 to read as follows:

- The reactor coolant activities are based on conservative assumptions (refer to Table 15.4-4); whereas, the activities based on the expected fuel defect level are far less (see Section 11.1).

58. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.8.3.6, with a LMA of LNP DEP 6.4-1, to read:

15.4.8.3.6 Doses

Revise the first paragraph of DCD Subsection 15.4.8.3.6 to read as follows:

Using the assumptions from Table 15.4-4, the calculated total effective dose equivalent (TEDE) doses are determined to be 4.0 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 5.9 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “well within” is taken as being 25 percent or less.

59. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.4.10, with a LMA of LNP DEP 6.4-1, to read:

15.4.10 References

Revise DCD Subsection 15.4.10 References 4, 7, 8, 10 and 13, and add new References 14 through 27 as follows:

4. Beard, C. L. et al., "Westinghouse Control Rod Ejection Accident Analysis Methodology Using Multi-Dimensional Kinetics," WCAP-15806-P-A (Proprietary) and WCAP-15807-NP-A (Nonproprietary), November 2003.
7. Liu, Y. S., et al., "ANC – A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A (Proprietary) and WCAP-10966-A (Nonproprietary), September 1986.
8. Not Used.
10. American National Standards Institute N18.2, "Nuclear Safety Criteria for the Design of Stationary PWR Plants," 1973.
13. Not Used.
14. Nguyen, T. Q., et al., "Qualifications of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A (Proprietary) and WCAP-11597-A (Nonproprietary), June 1988.
15. Ouisloumen, M., et al., "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP 16045 P A (Proprietary) and WCAP-16045-NP-A (Nonproprietary), August 2004.
16 Liu, Y. S., "ANC – A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery," WCAP-10965-P- A, Addendum 1 (Proprietary) and WCAP-10966- A Addendum 1 (Nonproprietary), April 1989.
17. Letter from Liparulo, N.J. (Westinghouse) to Jones, R. C., (NRC), "Notification to the NRC Regarding Improvements to the Nodal Expansion Method Used in the Westinghouse Advanced Nodal Code (ANC)," NTD-NRC-95-4533, August 22, 1995.
18. Sung, Y. X., Schueren, P., and Meliksetian, A., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Nonproprietary), October 1999.
19. Stewart, C. W., et al., "VIPRE-01: A Thermal/Hydraulic Code for Reactor Cores," Volumes 1, 2, 3 (Revision 3, August 1989), and Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute, Palo Alto, California.
20. Foster, J. P. and Sidener, S., "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 with Errata (Proprietary) and WCAP-15064-NP-A (Nonproprietary), July 2000.
21. Zhang, B., et al., "Qualification of the NEXUS Nuclear Data Methodology," WCAP-16045-P- A, Addendum 1-A (Proprietary) and WCAP-16045-NP-A, Addendum 1-A (Nonproprietary), August 2007.
22. Zhang, B., et al., "Qualification of the New Pin Power Recovery Methodology," WCAP-10965-P-A, Addendum 2-A (Proprietary), September 2010.
23. Smith, L. D , et al., "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," WCAP-15025-P-A (Proprietary) and WCAP-15026-NP-A (Nonproprietary), April 1999.
24. NUREG-0800, Standard Review Plan, Section 4.2, Revision 3, "Fuel System Design," Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," March 2003.

25. Draft Regulatory Guide DG-1199, "Proposed Revision 1 of Regulatory Guide 1.183; Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," October 2009. NRC ADAMS Accession Number: ML090960464.
26. NRC Memorandum from Anthony Mendiola to Travis Tate, "Technical Basis for Revised Regulatory Guide 1.183 (DG-1199) Fission Product Fuel-to-Cladding Gap Inventory," July 2011. NRC ADAMS Accession Number: ML111890397.
27. Letter from Liparulo, N. J. (Westinghouse) to Jones, R. C. (NRC), "Process Improvement to the Westinghouse Neutronics Code System," NSD-NRC-96-4679, March 29, 1996.

60. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.4-1, Time Sequence of Events for Incidents Which Result in Reactivity and Power Distribution Anomalies (Sheets 2 of 3), as new FSAR Table 15.4-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.4-201 is shown below:

61. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.4-4, Parameters Used In Evaluating the Radiological Consequences of a Rod Ejection Accident (Sheets 1 and 2), as new FSAR Table 15.4-202 (Sheets 1 and 2), with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.4-202 is shown below:

62. COLA Part 2 FSAR Chapter 15 will be revised to add a departure from DCD Figure 15.4.8-1, Nuclear Power Transient Versus Time at Beginning of Life, Full Power, as new FSAR Figure 15.4-201, Nuclear Power Transient Versus Time for the PCMI Rod Ejection Accident, with a LMA of LNP DEP 6.4-1. This figure will also be added to the list of figures from Chapter 15. Figure 15.4-201 is shown below:

63. COLA Part 2 FSAR Chapter 15 will be revised to add a departure from DCD Figure 15.4.8-2, Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at Beginning of Life, Full Power, as new FSAR Figure 15.4-202, Nuclear Power Transient Versus Time for the High Cladding Temperature Rod Ejection Accident, with a LMA of LNP DEP 6.4-1. This figure will also be added to the list of figures from Chapter 15. Figure 15.4-202 is shown below:

64. COLA Part 2 FSAR Chapter 15 will be revised to add a departure from DCD Figure 15.4.8-3, Nuclear Power Transient Versus Time at End of Life, Zero Power, as new FSAR Figure 15.4-203, Nuclear Power Transient Versus Time for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident, with a LMA of LNP DEP 6.4-1. This figure will also be added to the list of figures from Chapter 15. Figure 15.4-203 is shown below:

65. COLA Part 2 FSAR Chapter 15 will be revised to add a departure from DCD Figure 15.4.8-4, Hot Spot Fuel, Average Fuel, and Outer Cladding Temperature Versus Time at End of Life, Zero Power, as new FSAR Figure 15.4-204 (Not Used), with a LMA of LNP DEP 6.4-1. This figure will also be added to the list of figures from Chapter 15. Figure 15.4-204 is shown below:

Table 15.4-201

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds)
Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant		
1. Dilution during startup	Power range – low setpoint reactor trip due to dilution	0.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	215.0
2. Dilution during full-power Operation		
a. Automatic reactor control	Operator receives low-low rod insertion limit alarm due to dilution	0.0
	Shutdown margin lost	19,680
b. Manual reactor control	Initiate dilution	0.0
	Reactor trip on overtemperature ΔT due to dilution	180.0
	Dilution automatically terminated by demineralized water transfer and storage system isolation	395.0
RCCA ejection accident		
1. PCMI limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.14
	Reactor trip setpoint reached	<0.30
	Peak cladding temperature occurs	0.36
	Peak enthalpy deposition occurs	0.44
	Rods begin to fall into core	1.20
2. Peak cladding temperature limiting event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.08
	Minimum DNBR occurs	0.11
	Peak cladding temperature occurs	0.11

	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
3. Peak enthalpy/peak fuel centerline temperature event	Initiation of rod ejection	0.00
	Peak nuclear power occurs	0.06
	Reactor trip setpoint reached	<0.30
	Rods begin to fall into core	1.20
	Peak fuel center temperature occurs	2.50
	Peak cladding temperature occurs	2.80

Table 15.4-202 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/g (2.22E+06 Bq/g) of dose equivalent I-131 (see Appendix 15A)[(a)]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 μCi/g (1.036E+07 Bq/g) dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	1% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in damaged fuel)	1.75
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fuel enthalpy increase (cal/g)	60
– Fission product gap fractions	
Iodine 131	0.1238
Iodine 132	0.1338
Krypton 85	0.5120
Other noble gases	0.1238
Other halogens	0.0938
Alkali metals	0.6860
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3
Nuclide data	See Table 15A-4
Reactor coolant mass (lb)	3.7 E+05 (1.68E+05 kg)

Note:

a. The assumption of a pre-existing iodine spike is a conservative assumption for the initial reactor coolant activity. However, compared to the activity assumed to be released from damaged fuel, it is not significant.

Table 15.4-202 (Sheet 2 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Condenser	Not available
Duration of accident (days)	30
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Secondary system release path	
– Primary to secondary leak rate (lb/hr)	104.5[(a)] (47.4 kg/hr)
– Leak flashing fraction	0.04[(b)]
– Secondary coolant mass (lb)	6.06 E+05 (2.75E+05 kg)
– Duration of steam release from secondary system (sec)	1800
– Steam released from secondary system (lb)	1.08 E+05 (4.90E+04 kg)
– Partition coefficient in steam generators	
• Iodine	0.01
• Alkali metals	0.0035
Containment leakage release path	
– Containment leak rate (% per day)	
• 0-24 hr	0.10
• >24 hr	0.05
– Airborne activity removal coefficients (hr^{-1})	
• Elemental iodine	1.9[(c)]
• Organic iodine	0
• Particulate iodine or alkali metals	0.1
– Decontamination factor limit for elemental iodine removal	200
– Time to reach the decontamination factor limit for elemental iodine (hr)	2.78

Notes:

a. Equivalent to 300 gpd (1.14 m^3/day) cooled liquid at 62.4 lb/ft^3 (999.6 kg/m^3).
b. No credit for iodine partitioning is taken for flashed leakage.
c. From Appendix 15B.

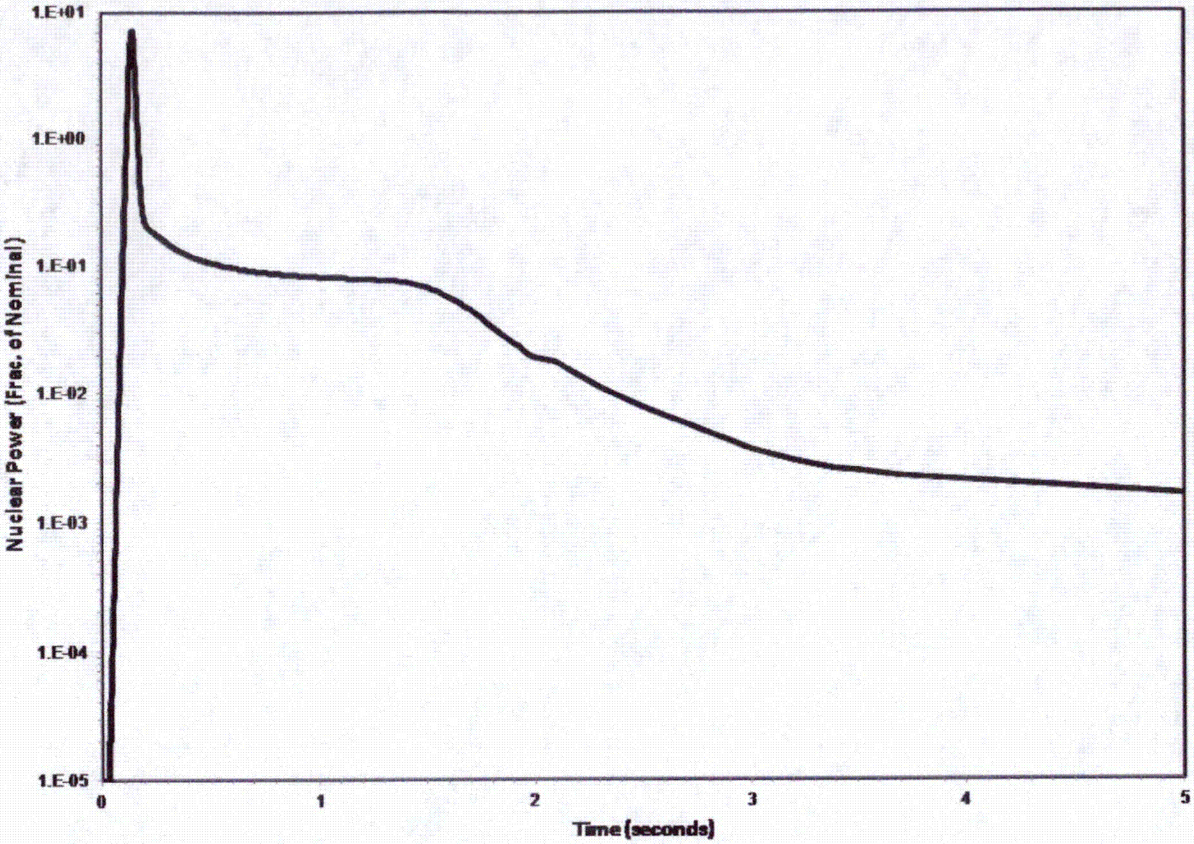

Figure 15.4-201

Nuclear Power Transient Versus Time for the PCMI Rod Ejection Accident

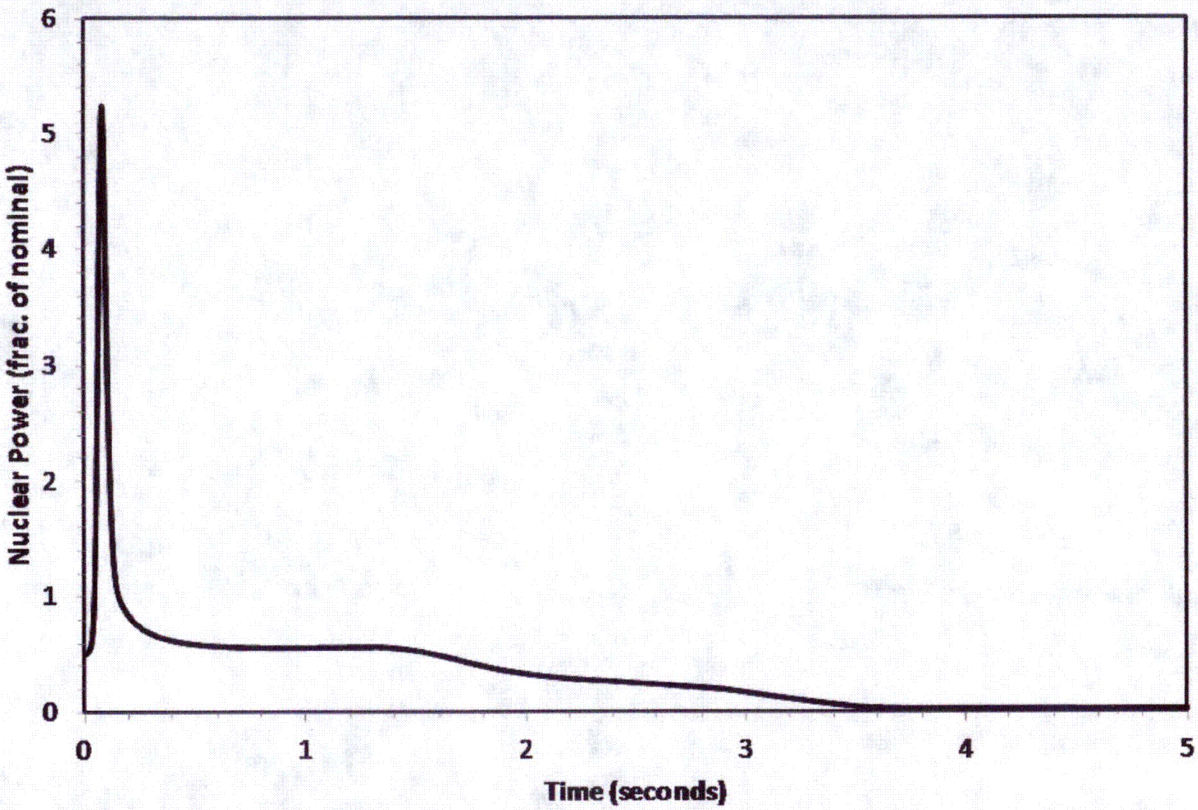

Figure 15.4 -202

Nuclear Power Transient Versus Time for the High Cladding Temperature Rod Ejection Accident

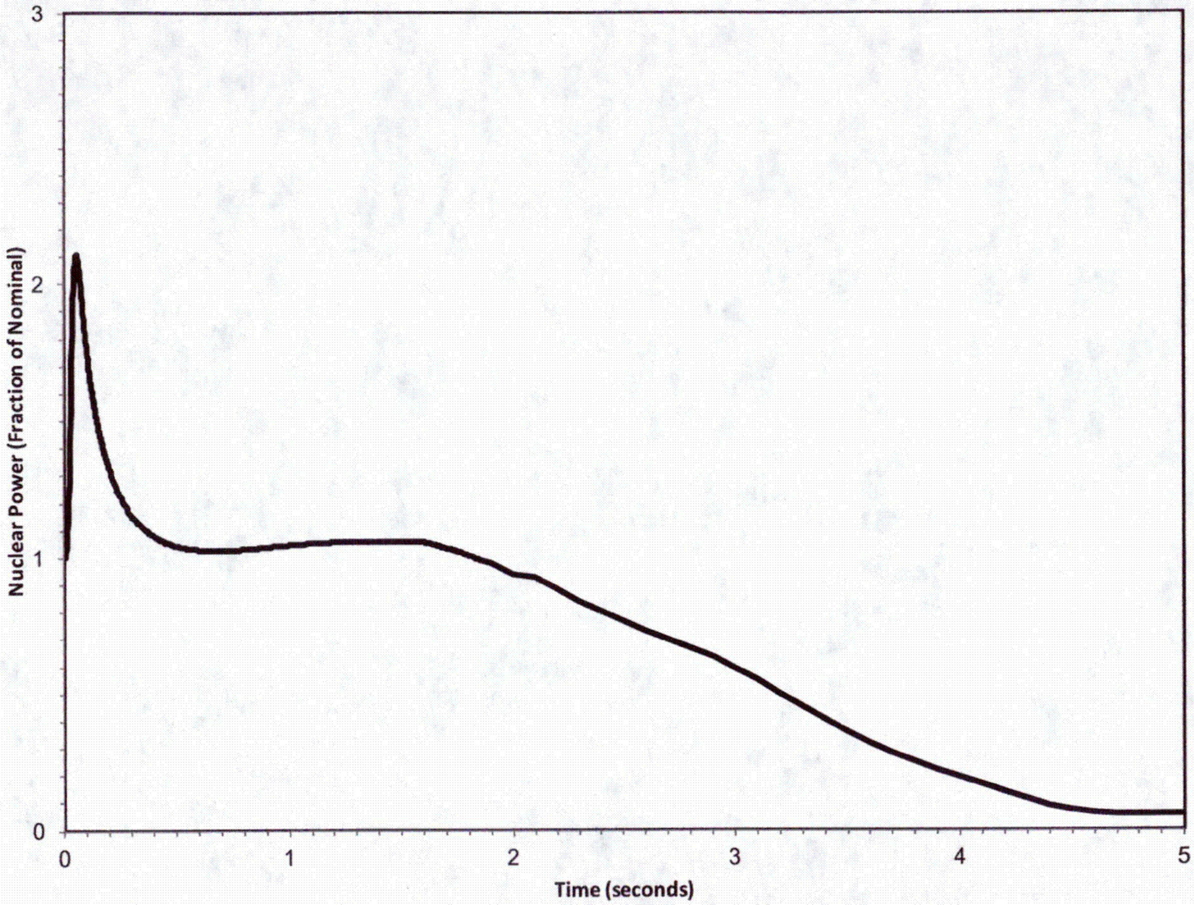

Figure 15.4-203

Nuclear Power Transient Versus Time
for the Peak Enthalpy and Fuel Centerline Temperature Rod Ejection Accident

Figure 15.4-204 Not used

66. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.2.6, with a LMA of LNP DEP 6.4-1, to read:

    15.6.2.6 Doses

    Revise the first paragraph of DCD Subsection 15.6.2.6 to read as follows:

    Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be 1.3 rem at the exclusion area boundary and 0.6 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase "a small fraction" is taken as being ten percent or less.

67. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.3.3.1, with a LMA of LNP DEP 6.4-1, to read:

    15.6.3.3.1 Source Term

    Revise the last paragraph of DCD Subsection 15.6.3.3.1 to read as follows:

    The secondary coolant iodine and alkali metal activity is assumed to be 1 percent of the maximum equilibrium primary coolant activity.

68. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.3.3.6, with a LMA of LNP DEP 6.4-1, to read:

    15.6.3.3.6 Doses

    Revise the first two paragraphs of DCD Subsection 15.6.3.3.6 to read as follows:

    Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be 0.7 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and 0.5 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

    For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be 1.4 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and 0.7 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

69. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.6.2-1, Parameters Used In Evaluating the Radiological Consequences Of A Small Line Break Outside Containment, as new FSAR Table 15.6-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.6-201 is shown below:

Table 15.6-201 **PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL LINE BREAK OUTSIDE CONTAINMENT**	
Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[(a)]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[(b)]
Fraction of reactor coolant flashing	0.47
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:

a. Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.

b. At density of 62.4 lb/ft^3.

70. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.6.3-3, Parameters Used In Evaluating the Radiological Consequences Of A Steam Generator Tube Rupture, as new FSAR Table 15.6-202, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.6-202 is shown below:

Table 15.6-202 **PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE**	
Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is 8.0 hours.

– Preaccident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	1% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.7 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	15.94
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop – Initial secondary coolant mass (lb) – Primary-to-secondary break flow – Integrated flashed break flow (lb) – Steam released (lb) – Iodine partition coefficient – Alkali metals partition coefficient	 1.16 E+05 See Figure 15.6.3-5 See Figure 15.6.3-10 See Table 15.6.3-2 1.0 E-02[(a)] 3.5 E-03[(a)]
Steam generator in intact loop – Initial secondary coolant mass (lb) – Primary-to-secondary leak rate (lb/hr) – Steam released (lb) – Iodine partition coefficient – Alkali metals partition coefficient	 2.30 E+04 52.16[(b)] See Table 15.6.3-2 1.0 E-02[(a)] 3.5 E-03[(a)]

Notes:

a. Iodine partition coefficient does not apply to flashed break flow.

b. Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

71. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.5.3.5, with a LMA of LNP DEP 6.4-1, to read:

15.6.5.3.5. Main Control Room Dose Model

Revise the first sentence of the second paragraph of DCD Subsection 15.6.5.3.5 to read as follows:

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when High-2 iodine or particulate activity is detected.

Revise the second sentence of the fourth paragraph of DCD Subsection 15.6.5.3.5 to read as follows:

With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

72. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.5.3.8.1, with a LMA of LNP DEP 6.4-1, to read:

    15.6.5.3.8.1 Offsite Doses

    Revise the first sentence of the second paragraph of DCD Subsection 15.6.5.3.8.1 to read as follows:

    The reported exclusion area boundary doses are for the time period of 1.3 to 3.3 hours.

73. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.6.5.3.8.2, with a LMA of LNP DEP 6.4-1, to read:

    15.6.5.3.8.2. Doses to Operators in the Main Control Room

    Revise the second and third sentence of the first paragraph of DCD Subsection 15.6.5.3.8.2 to read as follows:

    Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings, shine from radioactivity accumulated on the VES or VBS filters, and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of these dose paths is within the dose criteria of 5 rem TEDE as defined in GDC-19.

74. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.6.5-2, Assumptions And Parameters Used In Calculating Radiological Consequences Of A Loss-Of-Coolant Accident (Sheets 1 through 3), as new FSAR Table 15.6-203 (sheets 1 through 3), with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.6-203 is shown below:

Table 15.6-203 (Sheet 1 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Primary coolant source data	
– Noble gas concentration	280 μCi/g dose equivalent Xe-133
– Iodine concentration	1.0 μCi/g dose equivalent I-131
– Primary coolant mass (lb)	4.39 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	16000
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	1.0
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Elemental iodine deposition removal coefficient (hr^{-1})	1.9
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	3.89 E+04
– Volume of HVAC, including main control room and control support area (ft^3)	1.2 E+05
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	1650
• Filter efficiency	Not applicable
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6

Table 15.6-203 (Sheet 2 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Main control room model (cont.)	
– Occupancy	
• 0 - 24 hr	1.0
• 24 - 96 hr	0.6
• 96 - 720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited (VES Credited)	
– Main control room activity level at which the emergency habitability system actuation is actuated (Ci/m^3 of dose equivalent I-131)	2.0 E-07
– Response time to actuate VES based on radiation monitor response time and VBS isolation (sec)	200
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Unfiltered inleakage via ingress/egress (scfm)	5
• Unfiltered inleakage from other sources (scfm)	10
• Recirculation flow through filters (scfm)	600
• Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1900
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168

Table 15.6-203 (Sheet 3 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Control room with credit for continued operation of HVAC (VBS Supplemental Filtration Mode Credited)	
– Time to switch from normal operation to the supplemental air filtration mode (sec)	265
– Unfiltered air inleakage (cfm)	25
– Filtered air intake flow (cfm)	860
– Filtered air recirculation flow (cfm)	2740
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99
Miscellaneous assumptions and parameters	
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0 - 8 hr	3.5 E-04
8 - 24 hr	1.8 E-04
24 - 720 hr	2.3 E-04

75. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.6.5-3, Radiological Consequences Of A Loss-Of-Coolant Accident With Core Melt, as new FSAR Table 15.6-204, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.6-204 is shown below:

Table 15.6-204

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (1.3 - 3.3 hr)(1)	23.5
Low population zone boundary dose (0 - 30 days)	22.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	3.70
– Direct radiation from adjacent structures, including sky shine	0.30
– Filters shine	0.32
– Spent fuel pooling boiling	0.01
– Total	4.33
Main control room dose (normal HVAC operating in the supplemental filtration mode)	
– Airborne activity entering the main control room	4.50
– Direct radiation from adjacent structures, including sky shine	0.30
– Filters shine	0.03
– Spent fuel pooling boiling	0.01
– Total	4.84

Note:

1. This is the 2-hour period having the highest dose.

76. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15.7.4.5, with a LMA of LNP DEP 6.4-1, to read:

15.7.4.5 Offsite Doses

Revise the first sentence of the first paragraph of DCD Subsection 15.7.4.5 to read as follows:

Using the assumptions from Table 15.7-1, the calculated doses from the initial releases are determined to be 2.8 rem TEDE at the site boundary and 1.2 rem TEDE at the low population zone outer boundary.

77. COLA Part 2, FSAR Chapter 15, will be revised to add a departure from DCD Table 15.7-1, Assumptions Used to Determine Fuel Handling Accident Radiological Consequences, as new FSAR Table 15.7-201, with a LMA of LNP DEP 6.4-1. This table will also be added to the list of tables from Chapter 15. Table 15.7-201 is shown below:

Table 15.7-201

ASSUMPTIONS USED TO DETERMINE FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

Source term assumptions	
– Core power (MWt)	3434[(1)]
– Decay time (hr)	48
Core source term after 48 hours decay (Ci)	
I-130	1.28 E+05
I-131	8.18 E+07
I-132	9.10 E+07
I-133	4.06 E+07
I-135	1.17 E+06
Kr-85m	1.52 E+04
Kr-85	1.07 E+06
Kr-88	5.45 E+02
Xe-131m	1.02 E+06
Xe-133m	4.47 E+06
Xe-133	1.70 E+08
Xe-135m	1.91 E+05
Xe-135	1.04 E+07
Number of fuel assemblies in core	157
Amount of fuel damage	One assembly
Maximum rod radial peaking factor	1.75
Percentage of fission products in gap	
I-131	8
Other iodines	5
Kr-85	10
Other noble gases	5
Pool decontamination factor for iodine	200
Activity release period (hr)	2
Atmospheric dispersion factors	See Table 15A-5 in Appendix 15A
Breathing rates (m^3/sec)	3.5 E-4
Nuclide data	See Appendix 15A

Note:
1. The main feedwater flow measurement supports a 1-percent power uncertainty.

78. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15A.3.1.2, with a LMA of LNP DEP 6.4-1, to read:

15A.3.1.2 Secondary Coolant Source Term

Revise the first sentence of the first paragraph of DCD Subsection 15A.3.1.2 to read as follows:

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 1 percent of the primary coolant equilibrium source term.

79. COLA Part 2, FSAR Chapter 15, will be revised to update Subsection 15B, to read:

Appendix 15B Removal of Airborne Activity from the Containment Atmosphere following a LOCA

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

80. COLA Part 2, FSAR Chapter 15, will be revised to add new Subsection 15B.1, with a LMA of LNP DEP 6.4-1, to read:

15B.1 Elemental Iodine Removal

Revise the second full paragraph of DCD Subsection 15B.1 to read as follows:

The available deposition surface is 251,000 ft^2, and the containment building net free volume is 2.06×10^6 ft^3. From these inputs, the elemental iodine removal coefficient is 1.9 hr^{-1}.

81. COLA Part 4, Technical Specifications Section 3.7.4 will be revised as follows:

3.7.4 Secondary Specific Activity

LCO 3.7.4 The specific activity of the secondary coolant shall be ≤ 0.01 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. <u>AND</u> A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify the specific activity of the secondary coolant ≤ 0.01 μCi/gm DOSE EQUIVALENT I-131.	31 days

82. COLA Part 4, Technical Specifications Bases Section 3.4.10 Applicable Safety Analyses will be revised; the last sentence of the third paragraph will be revised as follows:

RCS Specific Activity
B 3.4.10

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.01 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.4, "Secondary Specific Activity."

83. COLA Part 4, Technical Specifications Bases Section 3.7.4 Applicable Safety Analyses and LCO will be revised as follows:

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (SLB) as discussed in Chapter 15 (Ref. 1) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.01 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a postulated SLB are within the acceptance criteria in SRP Section 15.0.1, and within the exposure guideline values of 10 CFR Part 50.34.

Secondary specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit of the secondary coolant is required to be ≤ 0.01 μCi/gm DOSE EQUIVALENT I-131 to maintain the validity of the analyses reported in Chapter 15 (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

84. COLA Part 4, Technical Specifications Bases Section 3.7.6 will be revised, specifically, the first paragraph of the Background and the first four paragraphs of the Applicable Safety Analyses will be revised as follows:

Main Control Room Emergency Habitability System (VES)
B 3.7.6

BASES

BACKGROUND	The Main Control Room Emergency Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 iodine or particulate Main Control Room Envelope (MCRE) radiation signal is received, the VES is actuated. The MCRE radioactivity is measured by detectors in the MCR supply air duct, downstream of the filtration units. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air (Ref. 4) for the MCRE occupants; 2) to provide forced ventilation to maintain the MCRE at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; 3) provide passive filtration to filter contaminated air in the MCRE; and 4) to limit the temperature increase of the MCRE equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.
APPLICABLE SAFETY ANALYSES	The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation. Operation of the VES is automatically initiated by either of the following safety related signals: • Control Room Air Supply Iodine or Particulate Radiation - High-2. • Loss of all AC power for more than 10 minutes In the event that a High-1 radioactivity setpoint value is reached, the non-safety VBS re-aligns to supplemental filtration mode, providing MCRE pressurization, cooling, and filtration. Upon high-2 particulate or iodine radioactivity setpoint, a safety related signal is generated to isolate the MCRE and to initiate air flow from the VES storage tanks. Isolation of the MCRE consists of closing safety related valves in the lines that penetrate the MCRE pressure boundary. Valves in the VBS supply and exhaust ducts, and the Sanitary Drainage System (SDS) vent lines are automatically isolated. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

85. COLA Part 7, Departures and Exemption Requests, will be revised to add the following exemption and departure:

A. STD and LNP Departures

This Departure Report includes deviations in the Levy Nuclear Plant, Units 1 and 2 COLA FSAR from the Tier 2 information in the applicable Design Control Document (DCD), pursuant to 10 CFR Part 52, Appendix D, Section VIII and Section X.B.1.

The following Departures are described and evaluated in detail in this report.

Departure Number	Description
STD DEP 1.1-1	Administrative departure for organization and numbering for the FSAR sections
LNP DEP 1.8-1	Correction of an inconsistency in regulatory citation in an interface description
LNP DEP 3.2-1	Addition of downspouts to the condensate return portion of the Passive Core Cooling System
LNP DEP 3.7-1	Use of site-specific horizontal seismic response spectra for the design of drilled shafts that support the seismic Category II portions of the Annex and Turbine Buildings.
LNP DEP 3.11-1	Revision of "Envir. Zone" numbers for Spent Fuel Pool Level instruments
LNP DEP 6.3-1	Quantification of the term "indefinitely" as used in the DCD for maintenance of safe shutdown conditions using the PRHR HX during non-LOCA accidents.
LNP DEP 6.4-1	MCR operator dose
STD DEP 8.3-1	Class 1E voltage regulating transformer current limiting features

Departure LNP DEP 6.4-1 is a departure from AP1000 Tier 1 information, in addition to Tier 2 information in the DCD; an exemption request and NRC approval is required prior to implementation.

Departure Number LNP DEP 6.4-1:

Affected DCD/FSAR Sections: Tier 1 Subsection 2.7.1, Tier 2 Table 1.6-1, Subsection 1.9.4.2.3, Appendix 1A, Subsection 3.1.2, Subsection 6.4, Subsection 6.4.2.6, Subsection 6.4.3.2, Subsection 6.4.4, Table 6.4-2, Subsection 7.3.1.2.17, Subsection 9.2.6.1.1, Subsection 9.4.1.1.1, Subsection 9.4.1.1.2, Subsection 9.4.1.2.1.1, Subsection 9.4.1.2.3.1, Figure 9.4.1-1 (Sheet 5 of 7), Table 11.1-4, Table 11.1-5, Table 11.1-6, Subsection 11.5.1.1, Subsection 11.5.2.3.1, Subsection 12.2.1.3.1, Subsection 12.2.1.3.2, Subsection 12.3.2.2.7, Table 12.2-28, Table12.2-29, Figure 12.3-1 (Sheet 6 of 16), Table 14.3-7 (Sheet 2 of 3), Subsection 15.0.11.1, Subsection 15.0.11.6, Table 15.0-2, Subsection 15.1.5.4.1, Subsection 15.1.5.4.6, Table 15.1.5-1, Subsection 15.3.3.3.1, Table 15.3-3 (Sheet 1 of 2), Subsection 15.4.8.1.1.3, Subsection 15.4.8.1.2, Subsection 15.4.8.2, Subsection 15.4.8.2.1, Subsection 15.4.8.2.1.1, Subsection 15.4.8.2.1.2, Subsection 15.4.8.2.1.3, Subsection 15.4.8.2.1.4, Subsection 15.4.8.2.1.5, Subsection 15.4.8.2.1.7, Subsection 15.4.8.2.1.8, Subsection 15.4.8.2.1.9, Subsection 15.4.8.3, Subsection 15.4.8.3.1, Subsection 15.4.8.3.5, Subsection 15.4.8.3.6, Subsection 15.4.10, Table 15.4-1 (Sheet 2 of 3), Table 15.4-3 (Deleted - Not Used), Table 15.4-4 (Sheet 1 and 2 of 2), Figure 15.4.8-1, Figure 15.4.8-2, Figure 15.4.8-3, Figure 15.4.8-4 (Not used), Subsection 15.6.2.6, Subsection 15.6.3.3.1, Subsection 15.6.3.3.6, Subsection 15.6.5.3.5, Subsection 15.6.5.3.8.1, Subsection 15.6.5.3.8.2, Table 15.6.2-1, Table 15.6.5-2 (Sheets 1-3 of 3), Table 15.6.5-3,Table 15.6.3-3, Subsection 15.7.4.5, Table 15.7.1, Subsection 15A.3.1.2, Subsection 15B.1, Chapter 16 LCO 3.7.4, SR 3.7.4.1, Bases 3.4.10, Bases 3.7.4, Bases 3.7.6.

Summary of Departure:

If high levels of particulate or iodine radioactivity are detected in the main control room supply air duct that could lead to exceeding General Design Criterion (GDC) 19 operator dose limits (5 rem), the protection and safety monitoring system (PMS) automatically actuates the VES to ensure compliance. The VES design includes a passive filtration feature consisting of a HEPA filter in series with a charcoal adsorber and a postfilter which work to remove particulate and iodine from the air to reduce potential control room dose during VES operation.

During AP1000 design finalization, a number of issues were identified challenging the ability of the certified design to limit operator dose to less than 5 Rem. In order to address these issues, site-specific revisions to the AP1000 design and associated dose consequence analyses presented in DCD Revision 19 are made to ensure that operator dose following a DBA is maintained below the 5 rem GDC limit for the duration of the event. Some design changes apply to all MCR design basis accidents and ventilation system alignments evaluated in DCD Section 6.4, while others are design basis accident specific.

A. Changes Impacting All MCR Design Basis Events

AP1000 generic changes impacting all MCR operator dose evaluations presented in DCD Section 6.4 and Chapter 15 required to address MCR dose analysis errors include:

1. Radiation contributions from HVAC filters were not considered in MCR dose calculation results reported in DCD Revision 19 Section 6.4. Regulatory Guide 1.183 indicates that these contributions should be considered in plant design. The radiological dose analyses are therefore revised to include direct radiation contributions from radioactive material postulated to accumulate on filters in the VES and VBS HVAC systems during design basis events.

2. In order to reduce the MCR operator direct radiation dose contribution from radioactive material postulated to accumulate on VES filter media during design basis events, shielding is added around the filters and is accounted for in the revised radiation analysis model. Consequence analyses considering filter contributions assume that control room occupants are located below these filters, using the defined occupancy factors (DCD Revision 19, Table 15.6.5-2, sheet 2 of 3).

3. In order to partially offset increases in calculated MCR operator dose due to consideration of direct radiation from VES filter media and other corrections identified in this response, the VES filter efficiency for organic iodine is increased from 30% to 90% (DCD Revision 19 Table 15.6.5-2, sheet 2 of 3). DCD Revision 19 post accident dose analyses applied an organic iodine filter efficiency of 30% to VES filtration units based on Regulatory Guide 1.52 Revision 2 and a conservative assumption that relative humidity within the MCR could exceed 95% following an accident. As part of AP1000 detailed design, environmental conditions have been evaluated to show that the humidity within the MCR is not expected to exceed 95%. Further, humidity is not expected to exceed 60% within the first 72 hours of an event, the time frame during which the filter would be

operating, or exceed 95% at any time post accident. Thus, the higher filter efficiency can be credited in the MCR dose analyses consistent with Regulatory Guide 1.52 Revision 2. Additionally, it is noted that the analyses model an "overall" efficiency for each chemical form being filtered (elemental, organic, particulate). This overall efficiency accounts for filter media sizing (e.g. charcoal bed depth) and the potential for bypass around the filter.

4. During AP1000 detailed design and re-evaluation of MCR doses to include consideration of HVAC filter contributions (Item A.1.), it was determined that the VBS radiation monitor setpoints applied in MCR dose calculations supporting DCD Revision 19 were not selected in a manner that a) ensures compliance with the GDC-19 for all postulated accident conditions including Design Basis Accidents (DBAs) evaluated in DCD Revision 19 Chapter 15, or b) fully supports the AP1000 design objective to use VBS supplemental filtration mode (SFM) when available rather than VES actuation to provide the MCR radiological protection function.

For postulated accident conditions involving a reduced source term or release rate other than evaluated for DBAs as part of the certified design, there may not be sufficient radioactivity within the MCR Envelope to prompt actuation of VES, and yet, enough radioactivity could exist that would lead to operator doses in excess of 5 rem without manual actuation. The radiation monitor setpoint values are therefore updated to ensure VBS or VES filtration mode actuation occurs for any radiological release event that could result in MCR operator doses in excess of GDC-19.

One of the fundamental objectives of VBS as described in DCD Revision 19 Section 9.4.1 is to "...minimize the potential for actuation of the main control room emergency habitability system...". This change uses a non-safety High-1 signal to actuate VBS SFM and the existing safety-related signal (High-2) to actuate VES in a manner that ensures High-2 would only be reached if VBS SFM was not functioning properly or is insufficient. This change also addresses release scenarios where high concentrations of particulates or iodine may exist with low levels of noble gas. If such a release occurred without this VBS setpoint logic change, direct VES actuation could be induced without the opportunity for VBS SFM to be actuated.

B. <u>Large Break Loss of Coolant Accident (LOCA) Dose Consequence Changes</u>

AP1000 generic changes impacting the LOCA MCR operator dose evaluations presented in DCD Sections 6.4 and 15.6.5 required to address MCR dose analysis errors include:

1. Dose contributions from adjacent structure direct and skyshine radiation included MCR operator dose results for LOCA as reported in DCD Revision 19 are based upon AP600 post-accident dose calculations and assume the presence of shielding that was not included in the AP1000 design. Post-accident radiological dose calculations are therefore changed to use updated AP1000 detailed design inputs and

analyses for skyshine and direct radiation. The added dose incurred by this change is partially offset by other proposed changes.

2. In order to partially offset increases in calculated MCR operator dose due to consideration of direct radiation from VES filter media and other corrections identified in this response, changes are made to the containment elemental iodine removal coefficient and re-suspension models supporting the DCD Revision 19 LOCA dose analysis.

Changes are made to the IRWST iodine re-evolution model. The changes involve a) the water/vapor partition factor modeled for elemental iodine and b) the timing associated with the conversion of elemental iodine to organic iodine and its availability for release. These refinements and modeling changes define the production of organic iodine based on re-evolved elemental iodine.

The iodine source term applied in the LOCA dose analysis supporting DCD Revision 19 is based upon the NUREG-1465 source term described in Regulatory Guide 1.183. The analysis models a staged release of core activity (i.e. gap release and early in-vessel) to the containment atmosphere over the first ~2 hours following the start of the event. The chemical form of iodine released is assumed to be 95% particulate, 4.85% elemental, and 0.15% organic, consistent with Regulatory Guide 1.183. Particulate removal via passive processes (i.e. diffusiophoresis, thermophoresis, and sedimentation) and elemental iodine removal via deposition are modeled. Organic iodine removal via processes other than decay or leakage from containment is not modeled.

Particulates removed to the containment shell are assumed to be washed off the shell by the flow of water resulting from condensing steam (i.e. condensate flow). The particulates may be either washed into the sump, which is controlled to a pH ≥7 post-accident or into the IRWST, which is not pH controlled post-accident. Due to the assumed conditions in the IRWST, the particulate iodine washed into the IRWST may chemically convert to an elemental form and re-evolve, subject to partitioning, as airborne. A portion (3%) of that airborne elemental iodine is then assumed to convert to an organic form. This is consistent with elemental organic split assumed for the initial release from the core (4.85/0.15 = 97/3) and is consistent the Regulatory Guide 1.183 guidance for other events.

The calculational approach to account for the iodine that is assumed to re-evolve from the IRWST post-LOCA is overly conservative in the certified design analysis. The certified design analysis applies a water-vapor partition factor of 5 for elemental iodine and neglects the time dependent formation of organic iodine from elemental iodine; the organic iodine that would be formed over time is assumed to be present at time zero.

NUREG-1465 states that "It is unduly conservative to assume that organic iodine is not removed at all from containment atmosphere, once generated, since such an assumption can result in an overestimate of the long-term doses to the thyroid." The revised analysis approach applies a conservative water/vapor elemental iodine partition factor of 10, selected to conservatively bound the time-dependent partition factors calculated using the NUREG/CR-5950 models and IRWST temperature and pH as a function of time. Additionally, the conversion of elemental iodine to organic iodine is modeled on a time-dependent basis in which 3% of the evolved elemental iodine is assumed to convert to an organic form upon its release to containment. It is noted that this does not impact the percentage of iodine assumed to convert to the organic form.

The passive containment elemental iodine deposition removal coefficient is also increased from the 1.7 hr^{-1} value applied in DCD Revision 19 LOCA dose calculations to 1.9 hr^{-1}. The larger elemental iodine removal rate constant is calculated based on a larger containment deposition surface area documented during the AP1000 detailed design. The DCD Revision 19 elemental iodine removal rate constant was based on an assumed 219,000 ft^2 deposition surface area. Updated detailed design calculations have documented a 251,000 ft^2 deposition surface area.

C: <u>Main Steam Line Break (MSLB) Dose Consequence Changes</u>

AP1000 generic changes impacting the MSLB MCR operator dose evaluations presented in DCD Sections 6.4 and 15.1.5 required to address MCR dose analysis errors include:

1. The AP1000 steam line break accident analysis described in DCD Revision 19 assumes a 10 minute faulted steam generator (SG) blowdown based on a Hot Zero Power (HZP) SG mass released at an average rate. This HZP case is conservative for offsite dose. It was determined, however, that a full power SG mass could lead to SG dry-out occurring at ~200 seconds. Earlier dry-out is more limiting for the purposes of operator post-accident dose calculations. To ensure a conservative dose for both offsite and MCR, the HZP initial mass was retained, a bounding release rate was modeled until 300 seconds, and any remaining activity was released thereafter.

2. In order to offset increases in calculated MCR operator dose for MSLB due to consideration of a bounding release rate and other corrections identified in this response, the Technical Specification limit for secondary iodine activity is reduced from 0.1 to 0.01 microcurie/gram dose equivalent (DE) I-131 (Limiting Condition for Operation (LCO) 3.7.4).

The current Technical Specification (TS) limit for secondary iodine activity is 0.1 μCi/g dose equivalent (DE) I-131 (LCO 3.7.4). This is a standard value, however the TS bases refer to the steam line break analysis; in other words, the steam line break analysis defines the level of secondary activity that is acceptable. The maximum secondary activity is also limited by other TS limits. Specifically, the primary coolant specific activity concentration limit of 1 μCi/g DE I-131 (based on the design basis fuel defect of 0.25%) and the TS primary to secondary leakage limit of 300 gallons per day.

Using these values, the secondary side coolant activity is calculated to be 8.3E-4 μCi/g DE I-131, which is orders of magnitude below the current TS limit. The TS limit for secondary iodine activity is therefore revised from 0.1 μCi/g DE I-131 to 0.01 μCi/g DE I-131. The change does not impact the operational margin, as the secondary side specific activity is limited to values lower than the new proposed TS limit by the TS primary to secondary leakage limit and the design basis fuel defect. The revised value of 0.01 μCi/g DE I-131 is within the detection capability of existing instrumentation and is significantly above the typical secondary coolant activities observed at operating plants. No additional sampling or modifications to the frequency of LCO 3.7.4 are needed.

D. Rod Ejection Accident (REA) Dose Consequence Changes

AP1000 generic changes impacting the REA MCR operator dose evaluations presented in DCD Sections 6.4 and 15.4.8 required to address MCR dose analysis errors include:

1. The method for performing the REA dose analysis has changed from that applied in DCD Revision 19. As stated in NUREG-1793, the NRC accepted the use of NUREG-0800 Section 4.2 Revision 2 for design certification of the AP1000 plant. However, in NUREG-1793 Supplement 2 it is stated that:

> "For COL applicants or licensees who reference the AP1000 or AP600 certified designs, the staff will review any change or departure from the certified design that requires prior NRC approval as specified in Section VIII of Appendices C and D to 10 CFR Part 52, respectively.
>
> The staff will evaluate the reactivity-initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to NUREG-0800 Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of final safety evaluation report Chapter 4, "Reactor," or Chapter 15, "Transient and Accident Analysis."

Due to the need to incorporate other design changes in the REA MCR operator dose calculations, NUREG-0800 Section 4.2 Revision 3 is used for recalculation of the rod ejection dose analysis, which results in a significant impact to the rod ejection dose analysis. NUREG-0800 Section 4.2 Revision 3 precludes fuel melt, providing a dose benefit, but also connects the source term to the fuel enthalpy increase, which is a significant dose penalty. The dominant contributor to the increased dose is the increase by a factor of more than 5 in alkali metal releases.

2. The full-power moisture carryover from the steam generators used in the AP1000 REA dose analysis was increased from 0.1% to 0.35%. This input is used to calculate alkali metal releases from the SGs in the AP1000 REA dose analysis. This input was updated to be consistent with the updated AP1000 plant design.

3. The REA dose analysis results are updated to account for a reduction in the Technical Specification limit for secondary iodine activity from 0.1 to 0.01 microcurie/gram dose equivalent (DE) I-131 as described for the updated MSLB analysis in Item C.2., above.

4. The radial peaking factor has been adjusted to a value of 1.75. This provides a more conservative input than the DCD Revision 19 value and provides additional future core design margin.

5. The passive containment elemental iodine deposition removal coefficient is also increased from the 1.7 hr^{-1} value applied in DCD Revision 19 LOCA dose calculations to 1.9 hr^{-1}. The larger elemental iodine removal rate constant is calculated based on a larger containment deposition surface area documented during the AP1000 detailed design. The DCD Revision 19 elemental iodine removal rate constant was based on an assumed 219,000 ft^2 deposition surface area. Update detailed design calculations have documented a 251,000 ft^2 deposition surface area.

E. Steam Generator Tube Rupture (SGTR) Dose Consequence Changes

AP1000 generic changes impacting the SGTR MCR operator dose evaluations presented in DCD Sections 6.4 and 15.6.3 required to address MCR dose analysis errors include:

1. The full-power moisture carryover from the intact steam generator used in the AP1000 SGTR dose analysis was increased from 0.1% to 0.35%. This input is used to calculate alkali metal releases from non-faulted SG in the AP1000 SGTR dose analysis. This input was updated to be consistent with the updated AP1000 plant design.

2. The full-power moisture carryover from the ruptured loop steam generator used in the AP1000 SGTR dose analysis was increased from 0.1% to 0.35%. This input is used to calculate alkali metal releases from the faulted SG in the AP1000 SGTR dose analysis. This input was updated to be consistent with the updated AP1000 plant design.

3. The SGTR results are updated to account for a reduction in the Technical Specification limit for secondary iodine activity from 0.1 to 0.01 microcurie/gram dose equivalent (DE) I-131 as described for the updated MSLB analysis in Item C.2., above.

F. Locked Rotor Accident Dose Consequence Changes

AP1000 generic changes impacting the locked rotor MCR operator dose evaluations presented in DCD Sections 6.4 and 15.3.3 required to address MCR dose analysis errors include:

1. The full-power moisture carryover from the steam generators used in the AP1000 locked rotor dose analysis was increased from 0.1% to 0.35%. This input was updated to be consistent with the updated AP1000 plant design.

2. The locked rotor results are updated to account for a reduction in the Technical Specification limit for secondary iodine activity from 0.1 to 0.01 microcurie/gram dose equivalent (DE) I-131 as described for the updated MSLB analysis in Item C.2., above.

3. The radial peaking factor has been adjusted to a value of 1.75. This provides a more conservative input than the DCD Revision 19 value and provides additional future core design margin.

G. Small Line Break Outside Containment Dose Consequence Changes

AP1000 generic changes impacting the small line break outside containment MCR operator dose evaluations presented in DCD Sections 6.4 and 15.6.2 required to address MCR dose analysis errors include:

1. The fraction of reactor coolant flashed was increased from the value used in DCD Revision 19 supporting calculations of 0.41 to 0.47 based on the updated detailed design. The certified design analysis used vessel average temperature (Tavg) as the basis for the flashing fraction. It was determined that the sample lines draw from the hot leg, thus, a hot leg temperature (which is greater than Tavg) was used.

H. Fuel Handling Accident (FHA) Dose Consequence Changes

AP1000 generic changes impacting the FHA MCR operator dose evaluations presented in DCD Sections 6.4 and 15.7.4 required to address MCR dose analysis errors include:

1. The radial peaking factor has been adjusted to a value of 1.75. This provides a more conservative input than the DCD Revision 19 value and provides additional future core design margin.

2. No unique accident specific changes.

In addition to the required changes summarized above, other generic changes associated with AP1000 detailed design are incorporated in revised MCR dose calculations. These include:

a) The MCR pressure boundary consists of the main control area, operator work area, mezzanine, operator break room, shift manager's office, kitchen area, and restrooms. The vestibule to enter the MCR and stairwell to the remote shutdown room is outside of the MCR pressure boundary. The MCR and MCR HVAC volumes are recalculated based on updated detailed design data, and are used as input to revised post-accident operator dose analyses.

b) The VBS normal operation to VBS SFM switchover time and the response time to actuate VES used in DCD Revision 19 supporting analyses have been determined to be non-bounding. The certified design analyses used assumed/expected values that were ultimately not supportable by the updated detailed design. System-level requirements are developed for switch over and response times; these system-level requirements account for sample transport time, radiation detector response time, I&C response times, and VBS/VES equipment actuations (e.g. valves, dampers, etc.). The dose analyses for cases considering VBS SFM are revised to include a longer delay time between the point when airborne radioactivity in the control room reaches the High-1 setpoint concentration and when the VBS SFM is operational. The dose analyses for cases considering VES are revised to include a longer response time between the point when the High-2 setpoint is reached and when the VES is operational.

c) DCD Revision 19 post accident dose calculations model a normal VBS inflow rate of 1925 cfm until MCR isolation. This is the path for the released activity to enter the MCR. This value is based on an assumed preliminary design value of 1750 cfm and adding a 10% penalty to account for instrumentation uncertainty. A VBS outside air intake flow rate calculated as part of detailed design indicates a nominal outside air flow rate of 1320 cfm. The normal VBS outside air flow rate assumption used in the accident operator dose calculations is therefore decreased accordingly. It is noted that during plant start-up, the HVAC system is balanced and dampers that have specific criteria under certain modes of operation are adjusted to have a preset position which can then be controlled from the MCR. To address a potential inaccuracy of the damper positioning, a nominal value of 1500 cfm is established for the normal VBS outside air flow rate. For dose analyses, the assumed normal VBS outside air flow rate will therefore be 1650 cfm, which corresponds to 1500 cfm +10%.

d) The VBS ancillary fan MCR air intake flow rate is also increased to 1900 cfm. The previous assumption of 1700 cfm had been specified as a minimum as part of the detailed design. For conservatism, the 1700 cfm was increased by 10% and rounded up to 1900 cfm.

Although these changes are considered as part of the updated MCR dose calculations, they are being implemented as general detailed design updates and are not specifically implemented to offset impacts of errors otherwise being addressed as part of this RAI response.

Departure Justification:

The proposed changes do not involve a significant reduction in the margin of safety. The proposed changes do not reduce the redundancy or diversity of any safety-related SSCs. The proposed changes improve the mitigating capabilities of the MCR Habitability System and address the MCR dose analysis errors. The MCR dose to the operators slightly decreases for the limiting DBA (LBLOCA) and the analysis shows that the results do not exceed the GDC-19 requirements of 5 rem.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) approval of the change will not be inimical to the health and safety of the public or to the common defense and security.

Departure Evaluation:

This Tier 2 departure makes the changes stated above. The departure does not involve a significant reduction in the margin of safety and does not reduce the redundancy or diversity of any safety-related SSCs. Analysis results show that the MCR dose does not exceed GDC requirements of 5 rem. Therefore:

1. This proposed departure does not impact the frequency of occurrence of an accident previously evaluated in the plant-specific DCD. Therefore there is not more than a minimal increase in the frequency of occurrence.
2. This proposed departure does not impact the likelihood of a malfunction of an SSC. Shielding is a passive function that does not impact HVAC function and is designed to remain in place under seismic conditions. The switchover times from normal HVAC in the control room (VBS) to either the VBS supplemental filtration mode or to the emergency habitability system (VES) are analyzed to determine conservative setpoints to establish bounding system-level requirements for each system participating in the switchover.
3. New analyses determined that the radiation dose to the operator during the limiting DBA (LBLOCA) decreased from 4.41 rem to 4.33 rem and continues to meet the GDC limit of 5 rem. However, the radiation dose to the operator during the rod ejection accident increased from 1.8 to 3.6 rem. Therefore, this departure does result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD.
4. Potential malfunctions of the HVAC system operation and switchover modes were analyzed and evaluated, and there were no design changes affecting or increasing source terms. Therefore, this departure will not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD.
5. This departure does not impact the possibility of accidents, and therefore does not create a possibility for an accident of a different type than previously evaluated in the plant-specific DCD.
6. The operability of the HVAC system with the different modes of operation (VBS w/SFM vs. VES) along with the interface of the RMS was analyzed and evaluated for adverse effects. It was determined that this departure would not create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD.
7. This departure does not result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered.
8. This departure proposes a refinement of the iodine evolution/re-evolution model, and a new computer code is used for recalculation of the rod ejection dose analysis. The use of the new iodine model and new rod ejection computer code is therefore considered to be a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

This departure does not affect resolution of a severe accident issue identified in the plant-specific DCD. Therefore, this departure has no safety significance.

NRC Approval Requirement:

This departure requires an exemption from the requirements of 10 CFR Part 52, Appendix D, Section III.B, which requires compliance with Tier 1 requirements of the AP1000 DCD and the generic Technical Specifications. Therefore, an exemption is requested in Part B of this COL Application Part. This departure also requires NRC approval pursuant to 10 CFR Part 52, Appendix D, Section VIII.B.5.

86. COLA Part 7, Departures and Exemption Requests, Exemption Request 5 will be added as follows:

B. Levy Nuclear Plant, Units 1 and 2 Exemption Requests

Duke Energy Florida, Inc. (DEF) requests the following exemptions related to:

1. Not used, and
2. Combined License (COL) Application Organization and Numbering
3. Special Nuclear Material (SNM) Material Control and Accounting Program Description
4. Containment Cooling Changes in regard to Passive Core Cooling System Condensate Return
5. Main Control Room Dose

Discussion and justification for each of these requests is provided in the following pages.

5) Main Control Room Dose

Applicable Regulation(s): 10 CFR Part 52, Appendix D, Section III.B
Specific wording from which exemption is requested:

"III. Scope and Contents

B. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix, including Tier 1, Tier 2 (including the investment protection short-term availability controls in Section 16.3 of the DCD), and the generic TS except as otherwise provided in this appendix. Conceptual design information in the generic DCD and the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

Pursuant to 10 CFR §52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for a plant-specific Tier 1 departure from the AP1000 DCD for Tier 1 information and for a material departure from the generic TS. The Tier 1 departure is contained in Tier 1 Subsection 2.7.1 and involves the revision of the VES actuation name to align with Tier 2 Chapter 7 naming convention. The departures also include a change to TS LCO 3.7.4 and TS SR 3.7.4.1 which involves lowering allowable secondary iodine activity. This exemption request is in accordance with the provisions of 10 CFR §50.12, 10 CFR §52.7, and 10 CFR Part 52, Appendix D.

Discussion:

The changes requested to Tier 1 Subsection 2.7.1, TS LCO 3.7.4 and TS SR 3.7.4.1 provide reasonable assurance that the facility has been constructed and will be operated in conformity with the applicable design criteria, codes and standards, and demonstrate acceptable main control room operator dose during design basis scenarios.

Conclusion:

This exemption request is evaluated in accordance with 10 CFR Part 52, Appendix D, Section VIII.A.4, 10 CFR §50.12, 10 CFR §52.7 and 10 CFR §52.63, which state that the NRC may grant exemptions from the requirements of the regulations provided the following six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, Appendix D, VIII.A.1]. The requested exemption satisfies the criteria for granting specific exemptions, as described below.

1. This exemption is authorized by law

The NRC has authority under 10 CFR §§ 50.12, 52.7, and 52.63 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR §§50.12 and 52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

Accordingly, this requested exemption is "authorized by law," as required by 10 CFR §50.12(a)(1).

2. This exemption will not present an undue risk to the health and safety of the public

The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information and a change to a TS LCO and SR to depart from the AP1000 certified (Tier 2) information. The plant-specific Tier 1 DCD will continue to reflect the approved licensing basis for the applicant, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the plant-specific DCD. Because the change maintains the capability of the Nuclear Island Nonradioactive Ventilation System to perform its design functions, the changed design will ensure the protection of the health and safety of the public. Therefore, no adverse safety impact which would present any additional risk to the health and safety is present.

Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3. The exemption is consistent with the common defense and security

The exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would change elements of the plant-specific Tier 1 DCD by departing from the AP1000 certified (Tier 1) design information relating to the Nuclear Island Nonradioactive Ventilation System and departing from the generic TS to lower the allowable secondary iodine activity. The exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

4. Special circumstances are present

10 CFR §50.12(a)(2) lists six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR §50.12(a)(2)(ii). That subsection defines special circumstances as when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The rule under consideration in this request for exemption from Tier 1 Subsection 2.7.1 and the generic TS is 10 CFR 52, Appendix D, Section III.B, which requires that an applicant referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information and generic TS. The Levy Units 1 and 2 COLA references the AP1000 Design Certification Rule and incorporates by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information and generic TS. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D to maintain the level of safety in the design.

The proposed change to the name of the actuation signal does not impact the design functions of the Nuclear Island Nonradioactive Ventilation System. This change does not impact the ability of any structures, systems, or components to perform their functions or negatively impact safety. Accordingly, this exemption from the certification information in Tier 1 Subsection 2.7.1, TS LCO 3.7.4 and TS SR 3.7.4.1 will enable the applicant to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 and the generic TS as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

5. The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption

Based on the nature of the changes to the plant-specific Tier 1 information and generic TS and the understanding that these changes support the design function of the Nuclear Island Nonradioactive Ventilation System and establish limits for the specific activity in the secondary system, it is likely that other AP1000 applicants and licensees will request this exemption. However, if this is not the case, the special circumstances continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the Nuclear Island Nonradioactive Ventilation System associated with this request will be maintained with the implementation of these changes. This exemption request and the associated TS LCO and TS SR changes demonstrate that the Nuclear Island Nonradioactive Ventilation System function continues to be maintained following implementation of the change from the generic AP1000 DCD, thereby minimizing the safety impact resulting from any reduction in standardization.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. In fact, as described in condition 6 below, the exemption will result in no reduction in the level of safety.

6. The design change will not result in a significant decrease in the level of safety.

The exemption revises the plant-specific DCD Tier 1 information by changing the name of the of the actuation signal (High-2) for isolating the main control room penetrations in Subsection 2.7.1 This change does not alter the ability of the Nuclear Island Nonradioactive Ventilation System to maintain its design functions. This exemption also revises the generic TS LCO 3.7.4 and TS SR 3.7.4.1 to lower the allowable secondary iodine activity. Because these functions are met, there is no reduction in the level of safety.

Therefore, the design change and change to the TS will not result in a significant decrease in the level of safety.

As demonstrated above, this exemption request satisfies NRC requirements for an exemption to the design certification rule for the AP1000 design.

87. COLA Part 10, License Conditions and ITAAC, Appendix B will be revised to add the following information:

Nuclear Island Nonradioactive Ventilation System ITAAC

Revise the sixth and seventh sentences of the Design Description information in DCD Tier 1 Section 2.7.1 to read as follows:

In addition, the VBS isolates the HVAC penetrations in the main control room boundary on "High-2" particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes. The Sanitary Drainage System (SDS) also isolates a penetration in the main control room boundary on “High-2” particulate or iodine radioactivity in the main control room supply air duct or on a loss of ac power for more than 10 minutes.