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10CFR52.79
10CFR52.63
10CFR50.12

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

LEVY NUCLEAR PLANT, UNITS 1 AND 2
DOCKET NOS. 52-029 AND 52-030
PARTIAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 121 RELATED TO SRP SECTIONS 6.2.5 AND 6.4 FOR THE LEVY NUCLEAR PLANT UNITS 1 AND 2 COMBINED LICENSE APPLICATION

Reference: Letter from Donald Habib (NRC) to Christopher M. Fallon (DEF), dated September 24, 2014, "Request for Additional Information Letter No. 121 Related to Standard Review Plan Sections 6.2.5 and 6.4 for the Levy Nuclear Plant, Units 1 and 2 Combined License Application" (ML14259A096).

Ladies and Gentlemen:

Duke Energy Florida, Inc. (DEF) hereby submits a partial response to the Nuclear Regulatory Commission's (NRC) request for additional information (RAI) provided in the referenced letter. A partial response to question 06.04-2 (eRAI 7661) is addressed in Enclosure 1 to this letter.

The response to the RAI question requires a site-specific departure from the AP1000 Design Control Document (DCD) Revision 19 information. The Levy Nuclear Plant (LNP) Combined License Application (COLA) incorporates the AP1000 DCD by reference. The response also includes a departure from a DCD Tier 1 design description and DCD Technical Specifications. Thus a request for exemption and associated change description is provided in Enclosure 2.

The proposed site-specific revisions to Tier 1 and Tier 2 Licensing Basis Documents are identified in Enclosure 3. The changes to the Final Safety Analysis Report and COLA Parts 4, 7 and 10 are identified in Enclosure 4 and will be included in a future update of the COLA.

If you have any further questions, or need additional information, please contact Bob Kitchen at (704) 382-4046, or me at (704) 382-9248.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 6, 2015.

Sincerely,

Christopher M Fallon

Christopher M. Fallon
Vice President - Nuclear Development

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NRO

Enclosures:

1. Partial Response to NRC RAI Letter No. 121
2. Request for Exemption Regarding Main Control Room Dose
3. Tier 1 and Tier 2 Licensing Basis Documents - Proposed Changes
4. Levy Nuclear Plant Units 1 and 2 COLA Revisions

cc: U.S. NRC Region II, Regional Administrator
Mr. Donald Habib, U.S. NRC Project Manager

Levy Nuclear Plant Units 1 and 2 (LNP)
Partial Response to NRC Request for Additional Information Letter No. 121 Related to Standard Review Plan Sections 6.2.5 and 6.4, dated September 24, 2014

NRC RAI #	Duke Energy RAI #	Duke Energy Response
06.04-2	L-1113	Partial response enclosed – see following pages

NRC Letter No.: LNP-RAI-LTR-121

NRC Letter Date: September 24, 2014

NRC Review of Final Safety Analysis Report

NRC RAI NUMBER: 06.04-2

Text of NRC RAI:

1. At a meeting with the NRC staff on July 23, 2014 (See ADAMS Accession Nos. ML14220A110, ML14220A111, ML14220A113), Westinghouse Electric Company presented some self-identified discrepancies in underlying calculations supporting the AP1000 DCD, Rev. 19, design basis accident main control room (MCR) habitability dose analyses. Westinghouse identified the need to update the analyses in order to show compliance with GDC-19 because

    the analyses did not account for the MCR emergency ventilation system (VES) filter direct dose in the control room,
    the control room ventilation system actuation setpoints did not account for all design basis accident (DBA) release scenarios, and
    the MCR dose contribution from direct radiation and skyshine used methodology that is not up-to-date.

10 CFR Part 50, Appx A, "General Design Criteria;" 10 CFR 52.63, "Finality of standard design certifications;" and 52.97, "Issuance of Combined Licenses" provide the regulatory basis for the following questions. GDC-19 sets out criteria for maintaining a control room to safely operate the plant in normal and accident conditions. Subsection 52.63(a)(1)(4) applies because additional information is needed to ensure that a plant referencing the DCD complies with GDC-19. Subsection 52.97(a)(1) applies because the Commission must have sufficient information to find that all NRC regulations have been met.

1a. Provide a site-specific departure from the DCD that includes control room dose analyses for all DBAs that account for the previously unanalyzed VES filter direct dose contribution to the MCR dose.

1b. Are the radiation monitor setpoints incorporated by reference from DCD Rev. 19 set such that GDC-19 is met for all DBAs for the COL? If not, propose a resolution to the issue. Describe how you determined the answer to this question.

1c. Do you propose to make site-specific revisions to the direct radiation and skyshine dose calculations to use a more detailed analysis methodology as proposed by Westinghouse? Would such revised calculations be necessary to show compliance with GDC-19 for the COL?

2. At the July 23, 2014 meeting, Westinghouse also proposed related changes to the plant design information to add shielding for the main control room filters, increase the VES filter efficiency for organic iodine to 90 percent, and revise the VES and the Nuclear Island Nonradioactive Ventilation System (VBS) actuation radiation monitor setpoints.

In your COL application, do you propose to also make the design changes proposed by Westinghouse, or any other changes related to the Westinghouse self-identified discrepancies in the MCR dose analyses? Are any of these design changes required for the COL in order to show compliance with GDC-19?

Duke RAI ID# L-1113

DEF Partial Response to NRC RAI:

Initially, DEF intended to demonstrate that conformance with GDC-19 requirements could be shown using site specific data contained in Table 2.0-202 of the LNP COLA. DEF subsequently determined that to meet GDC 19 requirements, a more comprehensive change is required to correct the errors in the certified design and that the design change being developed by Westinghouse must be fully implemented. Revised calculations that are required to support this change are in development by Westinghouse but are not expected to be completed in full until March 2015. Main control room (MCR) dose calculations for the large break loss-of-coolant accident (LOCA), main steam line break (MSLB) and spent fuel pool boiling design basis accident analyses have been completed and are reported in this submittal in order to allow NRC to initiate review and audit activities. Thus, the DEF COL application will incorporate the AP1000 generic design changes issued to address the Westinghouse Electric Company's self-identified discrepancies in the underlying calculations supporting the AP1000 DCD, Rev. 19, design basis accident MCR habitability dose analyses.

1a – The exemption required to support the proposed design change is provided in Enclosure 2. Changes to incorporate the departures associated with the LOCA and MSLB are provided in Enclosure 4.

1b – The radiation monitor setpoints are not set such that GDC-19 is met for all design basis accidents. Therefore, site-specific revisions will be included in the DEF COL application. These revised setpoints for MCR emergency habitability system (VES) actuation will be based upon concentrations for any particular monitoring channel (particulate or iodine) not exceeding an operator dose of 1 rem - regardless of release or accident scenario. This methodology will allow for airborne radioactivity in the control room to reach concentrations in each of the three channels at the setpoint and maintain compliance with GDC 19.

1c – Site-specific revisions for direct radiation and skyshine dose will be included in the DEF COL application. These revisions include updated direct radiation and skyshine dose calculations to account for MCR penetrations shielding differences between the AP1000 and AP600 designs. Accounting for the updated direct radiation and skyshine dose is required to show compliance with GDC-19.

2 – A summary of site-specific revisions to fully address the errors identified in the certified design is provided below. The changes to the VES filter design, the initial conditions assumed for the MSLB and the Technical Specification change to limit allowed secondary iodine activity are required to ensure compliance with GDC-19.

Summary of design change

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 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, errors in MCR dose analyses were discovered. The MCR dose analysis presented in AP1000 Design Control Document (DCD) Revision 19 Section 6.4 and Chapter 15 failed to consider MCR operator direct dose contributions from MCR ventilation system filtration unit accumulated radioactive sources. This error adversely affects results reported for all design basis accidents considered in DCD Section 6.4. In addition, another error was identified that adversely impacts the Main Steam Line Break (MSLB) dose consequence results reported in the DCD. The activity release rate modeled in the calculation of MSLB MCR doses was determined to be non-conservative specifically when applied to the AP1000 MCR habitability system design. However, offsite doses were determined to be conservative for this condition.

DEF has developed an exemption request and associated departure (LNP DEP 6.4-1) that accounts for the previously unanalyzed VES filter direct dose contribution. The AP1000 generic design changes for the main control room associated with the revised MSLB analysis are also incorporated into the DEF COL application. For AP1000 MCR operator dose, the revised main steam line break analysis conservatively assumes full power steam generator mass and conditions. This assumption results in a faster blowdown and thus maximizes the activity released prior to MCR isolation which yields higher MCR doses.

In order to address these errors, site-specific revisions to the AP1000 design and associated dose consequence analyses presented in DCD Revision 19 are required. 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 required to address MCR dose analysis errors include:

1. Direct dose contributions from the MCR VES and VBS filters are calculated and included in the total dose for MCR operators when demonstrating compliance with 10 CFR 50 Appendix A General Design Criteria (GDC) 19.

2. An integral shield plate has been added to the AP1000 VES filter unit design to lower direct dose contribution to MCR operators from radioactive material accumulated on the filter.

3. The VES filter efficiency for organic iodine is increased from 30% to 90% resulting in lower inhalation and immersion dose contributions (but increased filter shine dose contributions) to MCR operators for cases involving VES actuation.

4. The VBS radiation monitor setpoints for VBS supplemental filtration mode (SFM) transition and VES actuation are updated. This ensures that doses to MCR operators for all cases involving VBS or VES filtration mode actuation comply with GDC 19.

B. Large Break Loss of Coolant Accident (LOCA) Dose Consequence Changes

AP1000 generic changes impacting the LOCA MCR operator dose evaluations presented in DCD Sections 6.4 and 15.6 required to address MCR dose analysis errors include:

1. MCR dose contributions from adjacent building direct and skyshine are recalculated using AP1000 design parameters and credited shielding details.
2. The passive containment elemental iodine deposition removal coefficient is increased from 1.7 to 1.9 to account for the surface area of the current containment design.

C. Main Steam Line Break (MSLB) Dose Consequence Changes

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

1. The activity release rate model maximizes faulted steam generator releases for MCR dose calculation purposes (offsite dose calculations remain unchanged) which increases iodine activity infiltration into the MCR prior to MCR ventilation system switchover to filtered mode.
2. 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) in order to offset MCR dose impacts associated with other site-specific changes identified above.

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) MCR volumes (MCR and total MCR including support areas normally served by VBS) recalculated based on updated architectural drawings resulting in larger volumes in each case,

b) VBS SFM and VES actuation switchover time assumptions other than setpoint changes previously described resulting in increased switchover delay times,

c) VBS radiation monitor alarm logic changed to initiate VBS SFM on high iodine or particulate activity as well as noble gases,

d) Normal VBS outside air intake flow rate is reduced from 1925 to 1650 cfm

e) Core inventory and RCS source term adjusted to bound advanced first core design

f) VBS ancillary fan initiated MCR air intake flow rate is increased to 1900 cfm (from the minimum 1700 cfm), and

g) Various changes to align parameters reported in the FSAR to the updated analyses.

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.

Associated LNP COL Application Revisions:

See Enclosure 4

Duke Energy
Enclosure 2
Levy Nuclear Plant Units 1 and 2

Request for Exemption Regarding Main Control Room Dose

(8 pages including cover page)

1.0 Summary Description

General Design Criteria 19 requires limiting operator dose to less than 5 rem following design basis accidents (DBAs). The main control room and operator habitability requirements are met by the main control room emergency habitability system (VES).

The 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 DBA which requires protection from the release of radioactivity. In these events, the Nuclear Island Nonradioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 Main Control Room Envelope (MCRE) radiation signal is received, the VES is actuated.

The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air 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.

The operator dose resulting from design basis accidents approved in the certified design are in error. Errors in the certified design result from dose calculation associated with the Main Steam Line (MSL) break accident that did not address the most limiting plant condition. This error results in main control room operator dose that exceeds GDC-19 requirements. In order to demonstrate acceptable radiation exposure to main control room operators in the event of a MSLB accident requires more restrictive limits on secondary iodine activity limits. A revision to Technical Specification 3.7.4 "Secondary Specific Activity" to adjust the allowable secondary coolant iodine activity to one percent of the reactor coolant concentration at maximum equilibrium condition is required to meet GDC-19 requirements for the Main Steam Line break accident. In addition, the design description of the Nuclear Island Nonradioactive Ventilation System was revised to reflect the correct name of the actuation signal (High-2) for isolating the main control room penetrations consistent throughout Tier 1 and Tier 2 DCD content.

2.0 Description of Licensing Basis Impacts

Tier 1 Changes

The design description of the Nuclear Island Nonradioactive Ventilation System in Tier 1 Subsection 2.7.1 was revised to reflect the correct name of the actuation signal (High-2) for isolating the main control room penetrations. This signal isolates the HVAC penetration using the VBS and an additional penetration using the Sanitary Drainage System (SDS). This correction is a conforming change to Tier 2 Chapter 7 actuation setpoint naming convention.

Technical Specification Changes

The Technical Specification (TS) and TS Bases will be updated to lower the allowable specific activity of the secondary coolant. TS 3.7.4 Limiting Condition for Operation (LCO), Surveillance Requirement (SR) 3.7.4.1, Bases Applicable Safety Analyses and LCO for B 3.7.4, and Bases Applicable Safety Analyses for B 3.4.10 are updated.

3.0 Technical Evaluation

General Design Criteria 19 requires limiting operator dose to less than 5 rem following design basis accidents (DBAs). Changing the name of the actuation signal (High-2) for isolating the main control room penetrations in the design description of the Nuclear Island Nonradioactive Ventilation System is a conforming change to Tier 2 Chapter 7 actuation setpoint naming convention.

The current TS 3.7.4 limit for secondary iodine activitiy is 0.1 µC/g dose equivalent (DE) I-131. This is the level of secondary activity that is acceptable as defined by the steamline break analysis. The revised main steam line break analysis uses a secondary iodine activity limit of 0.01 µCi/g DE I-131. The revised TS limit for secondary iodine activity is acceptable under normal plant operation because the realistic DE I-131 value for secondary iodine activity as presented in DCD Table 11.1-8 is orders of magnitude below the revised TS. The revised DE I-131 value is within the detection capability of existing instrumentation, and is significantly above the secondary iodine coolant activities anticipated for AP1000.

4.0 Regulatory Evaluation

4.1 Exemption Justification

4.1.1 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 VES actuation signal 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 coolant 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, as demonstrated below.

<u>Applicable Regulation(s): 10 CFR Part 52, Appendix D, Section III.B</u>
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."

4.1.2 DEF evaluated this exemption request 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.4]. 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 ensures the Nuclear Island Nonradioactive Ventilation System will achieve 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 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 margins for the Nuclear Island Nonradioactive Ventilation System and reducing the TS limit for DE I-131 improves accident consequence margins for DBAs involving secondary coolant release. These changes do 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 continue to be maintained. This exemption request and the associated marked-up TS LCO and TS SR 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 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 actuation signal (from High-High to 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.

4.2 Significant Hazards Consideration

4.2.1 Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes rename the actuation signal (High-2) for isolating the main control room penetrations in the design description of the Nuclear Island Nonradioactive Ventilation System and lower the allowable secondary iodine activity to meet applicable NRC general design criteria requirements. As the proposed changes do not involve any components that could initiate an event by means of component or system failure, the changes do not increase the probability of a previously evaluated accident.

The proposed changes result in reduced consequences for the Main Control Room operator because of lower initial release concentrations for accident scenarios analyzed. The changes do not alter design features available during normal operation or anticipated operational occurrences. The changes do not adversely impact accident source term parameters or affect any release paths used in the safety analyses, which could increase radiological dose consequences. Thus the radiological releases associated with the Chapter 15 accident analyses are not adversely affected.

The proposed changes would not increase the consequences of an accident previously evaluated in the plant-specific DCD. Offsite doses are not adversely affected by the changes proposed. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

4.2.2 Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed changes do not introduce new failure modes, interactions or dependencies, the malfunction of which could lead to new accident scenarios. One of the changes is administrative and the other lowers an initial source term used in existing accident analyses. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

4.2.3 Does the proposed change involve a significant reduction in a margin of safety?

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 functions. The proposed changes lower the initial source term concentrations used in the accident analyses.

The DCD Chapters 6 and 15 analyses results are not adversely affected. No design basis safety analysis or acceptance criterion is challenged or exceeded by the proposed changes. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.3 Applicable Regulatory Requirements/Criteria

10 CFR 52, Appendix D, Section VIII.B.5.a requires that an applicant or licensee who references this appendix may depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of that section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD. This exemption request involves a departure from Tier 1 Subsection 2.7.1 and the generic TS.

4.4 Precedent

No precedent is cited.

4.5 Conclusions

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, and

(2) such activities will be conducted in compliance with the Commission's regulations, and

(3) the issuance of the exemption will not be inimical to the common defense and security or to the health and safety of the public.

The above evaluations demonstrate the requested changes can be accommodated without an increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without a significant reduction in a margin of safety. Having arrived at negative declarations with regard to the criteria of 10 CFR 50.92, this assessment determines the requested change does not involve a Significant Hazards Consideration.

5.0 Risk Assessment

A risk assessment was determined to be not applicable to address the acceptability of this request.

6.0 References

1) Westinghouse Electric Company, AP1000 Design Control Document, Revision 19, June 2011

Duke Energy

Enclosure 3

Levy Nuclear Plant Units 1 and 2

Tier 1 and Tier 2 Licensing Basis Documents – Proposed Changes (Convenience Copy)

TIER 1 DCD Changes

2.7 HVAC Systems

2.7.1 Nuclear Island Nonradioactive Ventilation System

Design Description

The nuclear island nonradioactive ventilation system (VBS) serves the main control room (MCR), control support area (CSA), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room (RSR), reactor coolant pump trip switchgear rooms, adjacent corridors, and passive containment cooling system (PCS) valve room during normal plant operation. The VBS consists of the following independent subsystems: the main control room/control support area HVAC subsystem, the class 1E electrical room HVAC subsystem, and the passive containment cooling system valve room heating and ventilation subsystem. The VBS provides heating, ventilation, and cooling to the areas served when ac power is available. The system provides breathable air to the control room and maintains the main control room and control support area areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations. The VBS monitors the main control room supply air for radioactive particulate and iodine concentrations and provides filtration of main control room/control support area air during conditions of abnormal (high) airborne radioactivity. 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. Additional penetrations from the SDS and Potable Water System (PWS) into the main control room boundary are maintained leak tight using a loop seal in the piping, and the Waste Water System (WWS) is isolated using a normally closed safety related manual isolation valve. These features support operation of the main control room emergency habitability system (VES), and have been included in Tables 2.7.1-1 and 2.7.1-2.

TIER 2 DCD Changes

1.9.4.2.3 New Generic Issues

Issue 83 Control Room Habitability

Discussion:

Loss of control room habitability following an accidental release of external toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor. Use of the remote shutdown workstation outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions.

AP1000 Response:

Habitability of the main control room is provided by the main control room/control support area HVAC subsystem of the nonsafety-related nuclear island nonradioactive ventilation system (VBS). 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). The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.

Appendix 1A

Reg. Guide 1.52, Rev. 3, 6/01 – Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants

C.4.9	Conforms	The credited adsorber efficiencies are 90% for elemental iodine and 90% for organic iodine. These efficiencies assume no humidity control.
Table 1	Conforms	The Technical Specification methyl iodide penetration acceptance limit for the AP1000 activated carbon adsorber is 5%, which correlates to 90% removal efficiency of both organic and elemental iodine. The calculated design basis for the AP1000 passive filtration adsorbers assumes a 90% organic iodine removal efficiency and a 90% elemental iodine efficiency. A bypass leakage is accounted for by testing.

3.1.2 Protection by Multiple Fission Product Barriers

Criterion 19 – Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

AP1000 Compliance

The AP1000 main control room provides the man-machine interfaces required to operate the plant safely and efficiently under normal conditions and to maintain it in a safe manner under accident conditions, including LOCAs. Simplified passive safety-related system designs are provided that do not rely upon operator action to maintain core cooling for design basis accidents. Operator action outside the main control room to mitigate the consequences of an accident is permitted.

The main control room is shielded by the containment and auxiliary building from direct gamma radiation and inhalation doses resulting from the postulated release of fission products inside containment. Refer to Chapter 15 for additional information on accident conditions. The main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system (VBS) allows access to and occupancy of the main control room under accident conditions as described in subsection 9.4.1. Sufficient shielding and the main control room/control support area HVAC subsystem provide adequate protection so that personnel will not receive radiation exposure in excess of 5 rem whole-body or its equivalent to any part of the body for the duration of the accident.

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). The main control room emergency habitability system also allows access to and occupancy of the main control room under accident conditions. The emergency main control room habitability system is designed to satisfy seismic Category I requirements as described in Section 3.2; the system design is described in Section 6.4.

In the event that the operators are forced to abandon the main control room, a workstation is provided with remote shutdown capability. A main control room evacuation is not assumed to occur simultaneously with design basis events. The remote shutdown workstation is described in Section 7.4.

6.4 Habitability Systems

The habitability systems are a set of individual systems that collectively provide the habitability functions for the plant. The systems that make up the habitability systems are the:

- Nuclear island nonradioactive ventilation system (VBS)
- Main control room emergency habitability system (VES)
- Radiation monitoring system (RMS)
- Plant lighting system (ELS)
- Fire Protection System (FPS)

When a source of ac power is available, the nuclear island nonradioactive ventilation system (VBS) provides normal and abnormal HVAC service to the main control room (MCR), control support area (CSA), instrumentation and control rooms, dc equipment rooms, battery rooms, and the nuclear island nonradioactive ventilation system equipment room as described in subsection 9.4.1.

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). The main control room emergency habitability system is capable of providing emergency ventilation and pressurization for the main control room. The main control room emergency habitability system also provides emergency passive heat sinks for the main control room, instrumentation and control rooms, and dc equipment rooms.

Radiation monitoring of the main control room environment is provided by the radiation monitoring system. Smoke detection is provided in the VBS system. Emergency lighting is provided by the plant lighting system. Storage capacity is provided in the main control room for personnel support equipment. Manual hose stations outside the MCR and portable fire extinguishers are provided to fight MCR fires.

6.4.2.6 Shielding Design

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.

6.4.3.2 Emergency Mode

Operation of the main control room emergency habitability system is automatically initiated by either of the following conditions:

- "High-2" particulate or iodine radioactivity in the main control room supply air duct
- Loss of ac power for more than 10 minutes

Operation can also be initiated by manual actuation.

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. At the same time, the main control room emergency habitability system begins to deliver air from the emergency air storage tanks to the main control room by automatically opening the isolation valves located in the supply line. The relief damper isolation valves also open allowing the pressure relief dampers to function and discharge the damper flow to purge the vestibule.

6.4.4 System Safety Evaluation

In the event of an accident involving the release of radioactivity to the environment, the nuclear island nonradioactive ventilation system (VBS) is expected to switch from the normal operating mode to the supplemental air filtration mode to protect the main control room personnel. Although the VBS is not a safety-related system, it is expected to be available to provide the necessary protection for realistic events. However, the design basis accident doses reported in Chapter 15 utilize conservative assumptions, and the main control room doses are calculated based on operation of the safety-related emergency habitability system (VES) since this is the system that is relied upon to limit the amount of activity the personnel are exposed to. The analyses assume that the VBS is initially in operation, but fails to enter the supplemental air filtration mode on a High-1 radioactivity indication in the main control room atmosphere. VES operation is then assumed to be initiated once the High-2 level for control room atmosphere activity is reached.

Doses are also calculated assuming that the VBS does operate in the supplemental air filtration mode as designed, but with no switchover to VES operation. This VBS operating case demonstrates the defense-in-depth that is provided by the system and also shows that, in the event of an accident with realistic assumptions, the VBS is adequate to protect the control room operators without depending on VES operation.

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	2.5 rem TEDE	1.6 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	4.3 rem TEDE	3.1 rem TEDE
(Accident-initiated iodine spike)	1.2 rem TEDE	1.7 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	1.8 rem TEDE	2.2 rem TEDE
Locked Rotor Accident		

(Accident without feedwater available)	0.7 rem TEDE	0.5 rem TEDE
(Accident with feedwater available)	0.5 rem TEDE	1.5 rem TEDE
Small Line Break Outside Containment	0.8 rem TEDE	0.3 rem TEDE

For all events the doses are within the dose acceptance limit of 5.0 rem TEDE. The details of analysis assumptions for modeling the doses to the main control room personnel are delineated in the LOCA dose analysis discussion in subsection 15.6.5.3 for VES operating cases. The analysis assumptions are provided in subsection 9.4.1.2.3.1 for the VBS operating case.

No radioactive materials are stored or transported near the main control room pressure boundary.

As discussed and evaluated in subsection 9.5.1, the use of noncombustible construction and heat and flame resistant materials throughout the plant reduces the likelihood of fire and consequential impact on the main control room atmosphere. Operation of the nuclear island nonradioactive ventilation system in the event of a fire is discussed in subsection 9.4.1.

The exhaust stacks of the onsite standby power diesel generators are located in excess of 150 feet away from the fresh air intakes of the main control room. The onsite standby power system fuel oil storage tanks are located in excess of 300 feet from the main control room fresh air intakes. These separation distances reduce the possibility that combustion fumes or smoke from an oil fire would be drawn into the main control room.

The protection of the operators in the main control room from offsite toxic gas releases is discussed in Section 2.2. The sources of onsite chemicals are described in Table 6.4-1, and their locations are shown on Figure 1.2-2. Analysis of these sources is in accordance with Regulatory Guide 1.78 (Reference 5) and the methodology in NUREG-0570, "Toxic Vapor Concentrations in the Control Room Following a Postulated Accidental Release" (Reference 6), and the analysis shows that these sources do not represent a toxic or flammability hazard to control room personnel.

A supply of protective clothing, respirators, and self-contained breathing apparatus adequate for 11 persons is stored within the main control room pressure boundary.

The main control room emergency habitability system components discussed in subsection 6.4.2.3 are arranged as shown in Figure 6.4-2. The location of components and piping within the main control room pressure boundary provides the required supply of compressed air to the main control room pressure boundary, as shown in Figure 6.4-1.

During emergency operation, the main control room emergency habitability system passive heat sinks are designed to limit the temperature inside the main control room to remain within limits for reliable human performance (References 2 and 3) over 72 hours. The passive heat sinks limit the air temperature inside the instrumentation and control rooms to 120°F and dc equipment rooms to 120°F. The walls and ceilings that act as the passive heat sinks contain sufficient thermal mass to accommodate the heat sources from equipment, personnel, and lighting for 72 hours.

The main control room emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks. Sixty scfm of supplied ventilation flow is sufficient to induce a filtration flow of at least 600 cfm into the passive air filtration line located inside the main control room envelope. This ventilation flow is also sufficient to pressurize the control room to at least positive 1/8-inch water gauge differential pressure with respect to the surrounding areas in addition to limiting the carbon dioxide concentration below one-half percent by volume for a maximum occupancy of 11 persons and maintaining air quality within the guidelines of Table 1 and Appendix C, Table C-1, of Reference 1.

Automatic transfer of habitability system functions from the main control room/control support area HVAC subsystem of the nuclear island nonradioactive ventilation system to the main control room emergency habitability system is initiated by either the following conditions:

- "High-2" particulate or iodine radioactivity in MCR air supply duct
- Loss of ac power for more than 10 minutes

The airborne fission product source term in the reactor containment following the postulated LOCA is assumed to leak from the containment and airborne fission products are assumed to result from spent fuel pool steaming. The concentration of radioactivity, which is assumed to surround the main control room, after the postulated accident, is evaluated as a function of the fission product decay constants, the containment leak rate, and the meteorological conditions assumed. The assessment of the amount of radioactivity within the main control room takes into consideration the radiological decay of fission products and the infiltration/exfiltration rates to and from the main control room pressure boundary.

A single active failure of a component of the main control room emergency habitability system or nuclear island nonradioactive ventilation system does not impair the capability of the systems to accomplish their intended functions. The Class 1E components of the main control room emergency habitability system are connected to independent Class 1E power supplies. Both the main control room emergency habitability system and the portions of the nuclear island nonradioactive ventilation system which isolates the main control room are designed to remain functional during an SSE or design-basis tornado.

In accordance with SECY-77-439 (Reference 13), a single passive failure of a component in the passive filtration line in the main control room emergency habitability system does not impair the capability of the system to accomplish its intended function. There is no source that could create line blockage in the VES line from the air bottles to the eductor. Thus potential blockage in the filtration line does not preclude breathable air from the emergency air storage tanks from being delivered to the main control room envelope for 72 hours during VES operation. Passive filtration using the main control room habitability system is required for the first 24 hours after the initiation of a design basis event to maintain operator doses below the acceptance limit of 5.0 rem TEDE. The doses for the following limiting cases were determined to demonstrate that passive filtration is not required after the first 24 hours after initiation of a design basis event. The following cases are evaluated because 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

Table 6.4-2 **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

9.2.6.1.1 Safety Design Basis

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. The SDS vent line that penetrates the main control room envelope is safety related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a design basis accident. An additional penetration from the SDS into the main control room envelope is maintained leak tight using a loop seal in the safety-related seismic Category I piping.

9.4.1.1.1 Safety Design Basis

The nuclear island nonradioactive ventilation system provides the following nuclear safety-related design basis functions:

- Monitors the main control room supply air for radioactive particulate and iodine concentrations
- 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

9.4.1.1.2 Power Generation Design Basis

Main Control Room/Control Support Area (CSA) Areas

The nuclear island nonradioactive ventilation system provides the following specific functions:

- Controls the main control room and control support area relative humidity between 25 to 60 percent
- Maintains the main control room and CSA areas at a slightly positive pressure with respect to the adjacent rooms and outside environment during normal operations to prevent infiltration of unmonitored air into the main control room and CSA areas
- 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 duct
- Isolates the main control room and/or CSA area from the normal outdoor air intake and provides 100 percent recirculation air to the main control room and CSA areas when a high concentration of smoke is detected in the outside air intake
- Provides smoke removal capability for the main control room and control support area
- Maintains the main control room emergency habitability system passive cooling heat sink below its initial design ambient air temperature limit of 75°F

- Maintains the main control room/control support area carbon dioxide levels below 0.5 percent concentration and the air quality within the guidelines of Table 1 and Appendix C, Table C-1 of Reference 32.

9.4.1.2.1.1 Main Control Room/Control Support Area HVAC Subsystem

The main control room/control support area HVAC subsystem serves the main control room and control support area with two 100 percent capacity supply air handling units, return/exhaust air fans, supplemental air filtration units, associated dampers, instrumentation and controls, and common ductwork. The supply air handling units and return/exhaust air fans are connected to common ductwork which distributes air to the main control room and CSA areas. The main control room envelope consists of the main control room, shift manager's office, operation work area, toilet, and operations break room area. The CSA area consists of the main control support area operations area, conference rooms, NRC room, computer rooms, shift turnover room, kitchen/rest area, and restrooms. The main control room and control support area toilets have separate exhaust fans.

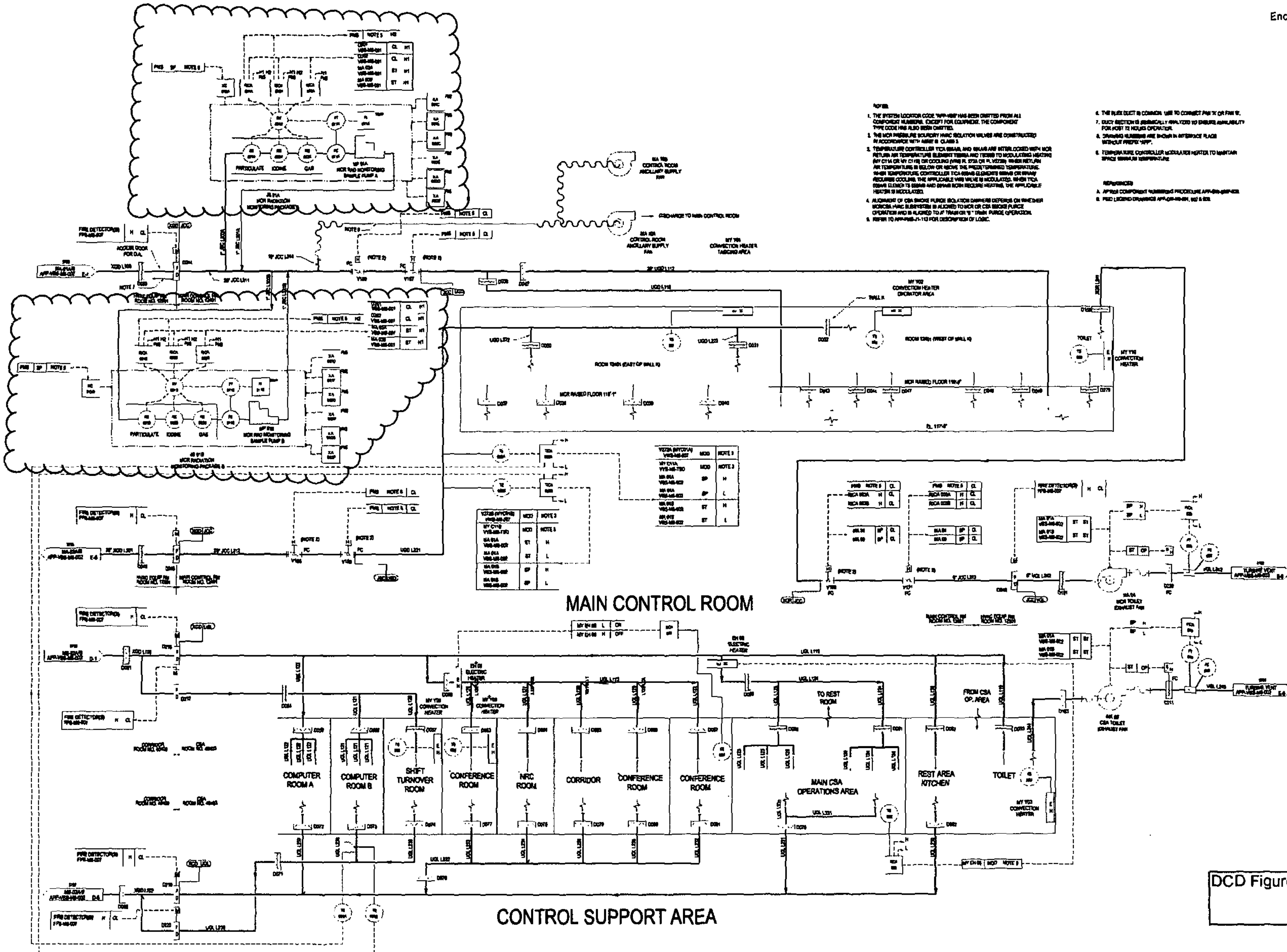
Outside supply air is provided to the plant areas served by the main control room/control support area HVAC subsystem through an outside air intake duct that is protected by an intake enclosure located on the roof of the auxiliary building at elevation 153'-0". The outside air intake duct is located more than 50 feet below and more than 100 feet laterally away from the plant vent discharge. The supply, return, and toilet exhaust are the only HVAC penetrations in the main control room envelope and include redundant safety-related seismic Category I isolation valves that are physically located within the main control room envelope. Redundant safety-related radiation monitor sample line connections are located upstream of the VBS supply air isolation valves. 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. See Section 11.5 for a description of the main control room supply air radiation monitors.

9.4.1.2.3.1 Main Control Room/Control Support Area HVAC Subsystem Abnormal Plant Operation

Control actions are taken at two levels of radioactivity as detected in the main control room supply air duct. 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.

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. The normal outside air makeup duct and the main control room and control support area toilet exhaust duct isolation dampers close. The smoke/purge exhaust isolation dampers close, if open. The main control room/control support area supply air handling unit continues to provide cooling with recirculation air to maintain the main control room passive heat sink below its initial ambient air design temperature and maintains the main control room and CSA areas within their design temperatures. The supplemental air filtration subsystem pressurizes the combined volume of the main control room and control support area concurrently with filtered outside air. A portion of the recirculation air from the main control room and control support area is also filtered for cleanup of airborne radioactivity. The main control room/control support area HVAC equipment and ductwork that form an extension of the main control room/control support area pressure boundary limit the overall infiltration (negative operating pressure) and exfiltration (positive operating pressure) rates to those values shown in Table 9.4.1-1. Based on these values, the system is designed to maintain personnel doses within allowable General Design Criteria (GDC) 19 limits during design basis accidents in both the main control room and the control support area.

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. Main control room habitability is maintained by the main control room emergency habitability system, which is discussed in Section 6.4.



DCD Figure 9.4.1-1, Sheet 5 of 7

Table 11.1-4

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

Table 11.1-5

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	4.0×10^{-7}
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}	Be-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}		

Table 11.1-6

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}

11.5.1.1 Safety Design Basis

While the radiation monitoring system is primarily a surveillance system, certain detector channels perform safety-related functions. The components used in these channels meet the qualification requirements for safety-related equipment as described in subsection 7.1.4.

Channel and equipment redundancy is provided for safety-related monitors to maintain the safety-related function in case of a single failure.

The design objectives of the radiation monitoring system during postulated accidents are:

- Initiate containment air filtration isolation in the event of abnormally high radiation inside the containment (High-1)
- Initiate normal residual heat removal system suction line containment isolation in the event of abnormally high radiation inside the containment (High-2)
- 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)
- Provide long-term post-accident monitoring (using both safety-related and nonsafety-related monitors)

The scope of the radiation monitoring system for post-accident monitoring is set forth in General Design Criterion 64 and in the provisions of Regulatory Guide 1.97.

11.5.2.3.1 Fluid Process Monitors

Main Control Room Supply Air Duct Radiation Monitors

The main control room supply air duct radiation monitors (particulate detectors VBS-JE-RE001A and VBS-JE-RE001B, iodine detectors VBS-JE-RE002A and VBS-JE-RE002B, and noble gas detectors VBS-JE-RE003A and VBS-JE-RE003B) are offline monitors that continuously measure the concentration of radioactive materials in the air that is supplied to the main control room by the nuclear island nonradioactive ventilation system air handling units. The control support area ventilation is also part of this air supply system. The air supply is partially outside air. Refer to subsection 9.4.1 for system details. The main control room supply air duct radiation monitors receive safety-related power. 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. Alarms are also provided in the main control room for these high concentrations.

From DCD Figure 12.3-1 Sheet 6

NOTES:

1. DURING SPENT RESIN WASTE DISPOSAL CONTAINER TRANSFER OR LOADING, THIS AREA CAN BE AS HIGH AS ZONE IX. THE CONTACT DOSE RATE OF THE SPENT RESIN CONTAINER IS >1000 REM/HR.

2. DURING CASK HANDLING OPERATIONS, AREAS OUTSIDE THE RAIL BAY DOORS CAN BE GREATER THAN THEN II LEVEL.

3. UNDERWATER SPENT FUEL ASSEMBLIES ARE AT MINIMUM 2'-6" FROM INSIDE SURFACE OF WEST WALL.

4. UNDERWATER SPENT FUEL ASSEMBLY IS AT MINIMUM 5'-0" FROM NORTH, SOUTH, EAST WALL.

5. ZONE IV WHEN FRESHLY DISCHARGED ASSEMBLIES ARE NOT IN THE OUTER ROW OF SPENT FUEL STORAGE RACK AND ZONE V WHEN FRESHLY DISCHARGED ASSEMBLIES ARE IN THE OUTER ROW OF SPENT FUEL STORAGE RACK

6. DURING UNDERWATER SPENT FUEL TRANSFER OPERATIONS, THIS AREA CAN BE AS HIGH AS ZONE IX.

7. DURING UNDERWATER REACTOR INTERNALS TRANSFER/STORAGE, THIS AREA CAN BE AS HIGH AS ZONE IX.

8. AREAS OUTSIDE SPENT FUEL POOL WALL CAN BE ZONE V WITH FRESHLY DISCHARGED ASSEMBLIES IN OUTER ROW OF SPENT FUEL STORAGE RACK.

9. BLOWDOWN PIPING MAY REACH ZONE III LEVELS WITH CONCURRENT FUEL CLADDING DEFECTS OF 0.25% AND STEAM GENERATOR TUBE LEAKAGE OF ~~500~~ gpd.

10. PASSIVE RHR PIPING MAY REACH ZON $\boxed{300}$ DURING HEAT EXCHANGER TESTING.

11. THE PURPOSE OF THIS DRAWING IS FOR IDENTIFICATION OF RADIATION ZONES ONLY. BACKGROUND INFORMATION MAY CHANGE AND LEGIBILITY OF THE BACKGROUND IS NOT REQUIRED.

Table 14.3-7 (Sheet 2 of 3)

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.	

15.1.5.4.1 Source Term

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. Noble gases are also released to the environment. Their impact is secondary because any noble gases entering the secondary side during normal operation are rapidly released to the environment.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accident occurs.

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.

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.

15.1.5.4.6 Doses

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.

Table 15.1.5-1

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

15.6.5.3.5 Main Control Room Dose Model

There are two approaches used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

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 activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. The bottled air also induces flow through the passive air filtration system which filters contaminated air in the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 scfm ±5 scfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this should result in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. However, no additional credit is taken for dilution of the vestibule via the purge. The projected inleakage into the main control room through ingress/egress is 5 cfm. An additional 10 cfm of unfiltered inleakage is conservatively assumed from other sources.

Activity entering the main control room is assumed to be uniformly dispersed. With the VES in operation, airborne activity is removed from the main control room atmosphere via the passive recirculation filtration portion of the VES.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. 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.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and is reported in Table 15.6.5-3.

Table 15.6.5-2 (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)	8800
– 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 gross volume (ft^3)	3.89E+04
– Gross Volume of HVAC, including main control room and control support area (ft^3)	1.2E+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 15.6.5-2 (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 15.6.5-2 (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

Table 15.6.5-3

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
– Radioactivity accumulated on HVAC filters	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
– Radioactivity accumulated on HVAC filters	0.03
– Spent fuel pooling boiling	0.01
– Total	4.84

15A.3.1.2 Secondary Coolant Source Term

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. This is more conservative than using the design basis secondary coolant source terms listed in Table 11.1-5.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

15B.1 Elemental Iodine Removal

Elemental iodine is removed by deposition onto the structural surfaces inside the containment. The removal of elemental iodine is modeled using the equation from the Standard Review Plan (Reference 1):

$$\lambda_d = \frac{K_w A}{V}$$

where:

λ_d = first order removal coefficient by surface deposition

K_w = mass transfer coefficient (specified in Reference 1 as 4.9 m/hr)

A = surface area available for deposition

V = containment building volume

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}.

Consistent with the guidance of Reference 1, credit for elemental iodine removal is assumed to continue until a decontamination factor (DF) of 200 is reached in the containment atmosphere. Because the source term for the LOCA (defined in subsection 15.6.5.3) is modeled as a gradual release of activity into the containment, the determination of the time at which the DF of 200 is reached needs to be based on the amount of elemental iodine that enters the containment atmosphere over the duration of core activity release.

3.7 PLANT SYSTEMS

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. AND 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

APPLICABLE SAFETY ANALYSES (continued)

assumed to be the LCO of 280 μCi/gm DOSE EQUIVALENT XE-133. 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."

The LCO limits ensure that, in either case, the doses reported in Chapter 15 remain bounding.

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

B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND	Activity in the secondary coolant results from steam generator tube LEAKAGE from the Reactor Coolant System (RCS). Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant. While fission products present in the primary coolant, as well as activated corrosion products, enter the secondary coolant system due to the primary to secondary LEAKAGE, only the iodines are of a significant concern relative to airborne release of activity in the event of an accident or abnormal occurrence (radioactive noble gases that enter the secondary side are not retained in the coolant but are released to the environment via the condenser air removal system throughout normal operation). The limit on secondary coolant radioactive iodines minimizes releases to the environment due to anticipated operational occurrences or postulated accidents.
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.

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System (VES)

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.

The VES consists of compressed air storage tanks, two air delivery flow paths, an eductor, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), associated valves or dampers, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCRE for at least 72 hours. The VES system is designed to maintain CO_2 concentration less than 0.5% for up to 11 MCRE occupants.

The MCRE is the area within the confines of the MCRE boundary that contains the spaces that control room operators inhabit to control the unit during normal and accident conditions. This area encompasses the main control area, operations work area, operational break room, shift supervisor's office, kitchen, and toilet facilities (Ref. 1). The MCRE is protected during normal operation, natural events, and accident conditions. The MCRE boundary is the combination of walls, floor, roof, electrical and mechanical penetrations, and access doors. The OPERABILITY of the MCRE boundary must be maintained to ensure that the inleakage of unfiltered air into the MCRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to MCRE occupants. The MCRE and its boundary are defined in the Main Control Room Envelope Habitability Program.

BACKGROUND (continued)

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCRE, which is initially at or below 75°F. Heat sources inside the MCRE include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCRE pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCRE habitability is maintained by operating one of the two MCRE ancillary fans to supply outside air to the MCRE.

The compressed air storage tanks are initially filled to contain greater than 327,574 scf of compressed air. The compressed air storage tanks, the tank pressure, and the room temperature are monitored to confirm that the required volume of breathable air is stored. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCRE. The MCRE is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas. The VES operation in maintaining the MCRE habitable is discussed in Reference 1.

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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 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.

Duke Energy
Enclosure 4
Levy Nuclear Plant Units 1 and 2

COLA Revisions

(34 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 have been revised. Shielding was added to control room VES filter, VBS signals were added, VES actuation set points 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 Subsection 1.9.4.2.3, Appendix 1A, Subsections 3.1.2, 6.4, 6.4.2.6, 6.4.3.2, and 6.4.4, Table 6.4-2, Subsections 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 9.4.1-1 (Sheet 5 of 7), Subsections 11.5.1.1, 11.5.2.3.1, Table 11.1-4, Table 11.1-5, Table 11.1-6, Figure 12.3-1 (Sheet 6 of 16), Table 14.3-7 (Sheet 2 of 3), Subsections 15.1.5.4.1, 15.1.5.4.6, Table 15.1.5-1, Subsections 15.6.5.3.5, 15.6.5.3.8.2, Table 15.6.5-2 (Sheets 1-3 of 3), Table 15.6.5-3, 15A, 15B, Chapter 16 (TS LCO 3.7.4, TS SR 3.7.4.1, TS Bases B 3.4.10, B 3.7.4 and B 3.7.6)	Chapter 1 (Table of Contents) 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 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) 11.1, 11.5.1.1, 11.5.2.3.1, Table 11.1-201, Table 11.1-202, Table 11.1-203, Chapter 12 (List of Figures), Figure 12.3-201, Chapter 14 (List of Tables) Table 14.3-203, Chapter 15 (Table of

Contents, List of
Tables)
15.1,
15.1.5.4.1,
15.1.5.4.6,
Table 15.1-201,
15.6.5.3.5,
15.6.5.3.8.2,
Table 15.6-201
(Sheets 1-3 of 3),
Table 15.6-202,
15A,
15B,
16 TS LCO 3.7.4,
16 TS SR 3.7.4.1,
16 TS Bases B 3.4.10,
16 TS Bases B 3.7.4,
16 TS Bases B 3.7.6

COLA Part 2, FSAR Sections 1.9, Appendix 1AA, 3.1, 6.4, 9.2, 9.4, 11.1, 11.5, 12.3, 14.3, 15.1, 15.6, 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 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).

3. 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	The credited adosorber efficiencies are 90% for elemental iodine and 90% for organic iodine. These efficiencies assume no humidity control.
Table 1	Conforms	The Technical Specification methyl iodide penetration acceptance limit for the AP1000 activated carbon adsorber is 5%, which correlates to 90% removal efficiency of both organic and elemental iodine. The calculated design basis for the AP1000 passive filtration adsorbers assumes a 90% organic iodine removal efficiency and a 90% elemental iodine efficiency. A bypass leakage is accounted for by testing.

4. 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.

5. 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).

6. 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).

7. 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

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	2.5 rem TEDE	1.6 rem TEDE
Steam Generator Tube Rupture		
(Pre-existing iodine spike)	4.3 rem TEDE	3.1 rem TEDE
(Accident-initiated iodine spike)	1.2 rem TEDE	1.7 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	1.8 rem TEDE	2.2 rem TEDE
Locked Rotor Accident		
(Accident without feedwater available)	0.7 rem TEDE	0.5 rem TEDE
(Accident with feedwater available)	0.5 rem TEDE	1.5 rem TEDE
Small Line Break Outside Containment	0.8 rem TEDE	0.3 rem TEDE

11. 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 add:

    6.4.4 System Safety Evaluation

    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

12. 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 add:

    6.4.4 System Safety Evaluation

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

    Passive filtration using the main control room habitability system is required for the first 24 hours after the initiation of a design basis event to maintain operator doses below the acceptance limit of 5.0 rem TEDE. The doses for the following limiting cases were determined to demonstrate that passive filtration is not required after the first 24 hours after initiation of a design basis event. The following cases are evaluated because 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

13. 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.

14. 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

15. 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

16. 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.

17. 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.

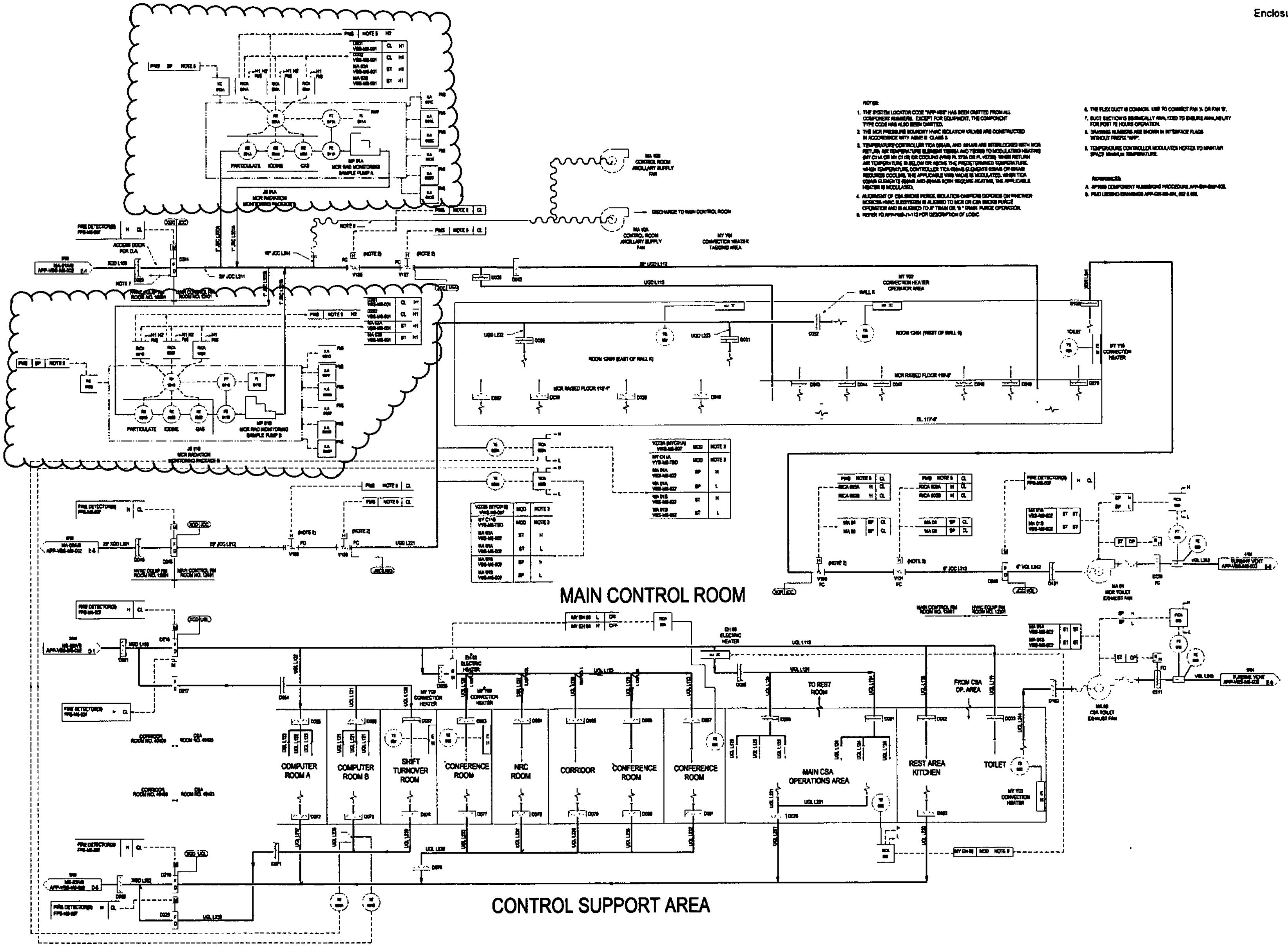
Revise the first sentence of the second paragraph of the Abnormal Plant Operation section of DCD Subsection 9.4.1.2.3.1 to read as follows:

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.

18. 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:


MAIN CONTROL ROOM
CONTROL SUPPORT AREA
PARTICULATE
IODINE
GAS
COMPUTER ROOM A
COMPUTER ROOM B
SHIFT TURNOVER ROOM
CONFERENCE ROOM
NRC ROOM
CORRIDOR
CONFERENCE ROOM
CONFERENCE ROOM
MAIN CSA OPERATIONS AREA
TO REST ROOM
REST AREA KITCHEN
FROM CSA OP. AREA
TOILET

19. 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.

20. 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

21. 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	4.0×10^{-7}
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}		

22. 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}

23. 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 bullet 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)

24. 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.

25. 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.

26. 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.	

27. 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.

28. 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.

29. 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.

30. 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.

31. 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 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.

32. 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.

33. 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 15.6-201 (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 15.6-201 is shown below:

Table 15.6-201 (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)	8800
– 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 gross volume (ft^3)	3.89 E+04

– Gross 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 15.6-201 (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 15.6-201 (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

34. 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 15.6-202, 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

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
– Radioactivity accumulated on HVAC filters	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
– Radioactivity accumulated on HVAC filters	0.03
– Spent fuel pooling boiling	0.01
– Total	4.84

Note:

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

35. 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.

36. 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.

37. 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}.

38. 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. AND	6 hours
	A.2 Be in MODE 5.	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

39. 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."

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

BASES

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.

41. 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.

42. 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 Subsection 1.9.4.2.3, Appendix 1A, Subsection 3.1.2, Subsections 6.4, 6.4.2.6, 6.4.3.2, and 6.4.4, Table 6.4-2, Subsections 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 9.4.1-1 (Sheet 5 of 7), Subsections 11.5.1.1, 11.5.2.3.1, Table 11.1-4, Table 11.1-5, Table 11.1-6, Figure 12.3-1 (Sheet 6 of 16), Table 14.3-7 (Sheet 2 of 3), Subsections 15.1.5.4.1, 15.1.5.4.6, Table 15.1.5-1, 15.6.5.3.5, 15.6.5.3.8.2, Table 15.6.5-2 (Sheets 1-3 of 3), Table 15.6.5-3, 15A, 15B, Chapter 16 (TS LCO 3.7.4, TS SR 3.7.4.1, TS Bases B 3.4.10, B 3.7.4 and B 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 would lead to exceeding General Design Criterion 19 operator dose limits (5 rem), the protection and safety monitoring system (PMS) automatically actuates the Main Control Room (MCR) emergency habitability system (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. They work to remove particulate and iodine from the air to reduce potential control room dose during VES operation.

During AP1000 design finalization, errors in MCR dose analyses were discovered. The MCR dose analysis presented in AP1000 Design Control Document (DCD) Revision 19 Section 6.4 and Chapter 15 safety evaluations failed to consider MCR operator direct dose contributions from MCR ventilation system filtration unit accumulated radioactive sources. This error adversely affects results reported for all design basis accidents considered in DCD Section 6.4. In addition, another error was identified that adversely impacts the Main Steam Line Break (MSLB) dose consequence results reported in the DCD. The activity release rate modeled in the calculation of MSLB MCR doses was determined to be non-conservative when applied to the AP1000 MCR habitability system design. However, offsite doses were determined to be conservative for this condition.

Scope/Extent of Departure:

In order to address these errors, changes to the AP1000 design and associated dose consequence analyses presented in DCD Revision 19 are required. 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. Generic Changes

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

1. Direct dose contributions from the MCR VES and VBS filters are calculated and included in the total dose for MCR operators when demonstrating compliance with 10 CFR 50 Appendix A General Design Criteria (GDC) 19.
2. A shield plate has been added to the AP1000 VES filter unit design to lower direct dose contribution to MCR operators from radioactive material accumulated on the filter.
3. The VES filter efficiency for organic iodine is increased from 30% to 90% resulting in lower inhalation and immersion dose contributions (but increased change A.1 filter-shine dose contributions) to MCR operators for cases involving VES actuation.
4. The VBS radiation monitor setpoints for VBS supplemental filtration mode (SFM) transition and VES actuation are updated. This ensures that doses to MCR operators for all cases involving VBS or VES filtration mode actuation comply with GDC 19 and are as low as reasonably achievable.

B. Large Break Loss of Coolant Accident (LOCA) Dose Consequence Changes

AP1000 changes impacting the LOCA MCR operator dose evaluations presented in DCD Sections 6.4 and 15.6 required to address MCR dose analysis errors include:

1. MCR dose contributions from adjacent building direct and skyshine are recalculated using AP1000 design parameters and credited shielding details.

2. The passive containment elemental iodine deposition removal coefficient is increased from 1.7 to 1.9 to account for the surface area of the current containment design.

C. Main Steam Line Break (MSLB) Dose Consequence Changes

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

1. The activity release rate model maximizes faulted steam generator releases for MCR dose calculation purposes (offsite dose calculations remain unchanged) which increases iodine activity infiltration into the MCR prior to MCR ventilation system switchover to filtered mode.
2. The Technical Specification (TS) limit for secondary iodine activity is reduced from 0.1 to 0.01 microcurie/gram dose equivalent (DE) I-131 (LCO 3.7.4) in order to offset MCR dose impacts associated with the changes A.1 and C.1.

In addition to the required changes summarized above, other generic changes associated with design finalization are incorporated in revised MCR dose calculations. These include, a) MCR volumes (MCR and total MCR including support areas normally served by VBS) recalculated based on updated architectural drawings resulting in larger volumes in each case, b) VBS SFM and VES actuation switchover time assumptions other than setpoint changes previously described resulting in increased switchover delay times, c) VBS radiation monitor alarm logic changed to initiate VBS SFM on high iodine or particulate activity as well as noble gases, d) normal VBS outside air intake flow rate is reduced from 1925 to 1650 cfm, e) VBS ancillary fan initiated MCR air intake flow rate is increased to 1900 cfm (from the minimum 1700 cfm), and f) various changes to align parameters reported in the FSAR to the updated analyses. Although these changes are considered as part of the updated MCR dose calculations, they are being implemented as general detailed design updates at this time. Even though these changes have an effect on dose consequence results, the changes are not specifically implemented to offset impacts of errors otherwise being addressed as part of this departure evaluation.

Departure Justification:

The proposed change does not involve a significant reduction in the margin of safety. The proposed change does 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 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 adds a shield plate to the AP1000 VES filter unit design to lower direct dose contribution to MCR operators from radioactive material accumulated on the filter. The MCR dose analysis now includes the direct dose contributions from the MCR VES and VBS filters. The VBS radiation monitor setpoints for VBS supplemental filtration mode transition and VES actuation are optimized. The VES filter efficiency for organic iodine is increased from 30% to 90%. The passive containment elemental iodine deposition removal coefficient is increased. The plant mode of operation for MSLB analysis is changed from Hot Zero-Power to Full Power. The Tech Spec limits for secondary iodine activity is reduced from 0.1 to 0.01 microcurie/gram dose equivalent I-131. 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 a LBLOCA decreased from 4.41 rem to 4.33 rem and continues to meet the GDC 19 limit of 5 rem. Therefore, this departure does not 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 an increase in the elemental iodine deposition removal coefficient for iodine within containment during a LOCA. This new model creates more margin in the radiological dose safety analysis and 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.

43. 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 marked-up TS LCO and TS SR 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 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.

44. 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.