INDIANA MICHIGAN POWER®
A unit of American Electric Power

Indiana Michigan Power
Cook Nuclear Plant
One Cook Place
Bridgman, MI 49106
IndianaMichiganPower.com

November 14, 2014

AEP-NRC-2014-65
10 CFR 50.90
10 CFR 50.67

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant, Units 1 and 2
LICENSE AMENDMENT REQUEST TO ADOPT TSTF-490, REVISION 0,
"DELETION OF E BAR DEFINITION AND REVISION TO REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY TECHNICAL SPECIFICATION" AND IMPLEMENT FULL-SCOPE ALTERNATIVE SOURCE TERM

References:

1. TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," dated September 13, 2005.

2. Federal Register Notice of Availability published on March 19, 2007 (72 FR 12838), Notice of Availability of Model Application Concerning Technical Specification Improvement Regarding Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification Using the Consolidated Line Item Improvement Process.

3. Summary of the August 8, 2014, Pre-Application Meeting to Discuss Pending License Amendment Request Associated with the Implementation of Alternative Source Term And Technical Specification Task Force Traveler (TSTF)-490 (TAC NOS. MF4483 AND MF4484), dated August 27, 2014.

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Indiana Michigan Power Company (I&M) is submitting a request for an amendment to the Technical Specifications (TS) for Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. The proposed changes would replace the current CNP Units 1 and 2 TS 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xe-133 definition that would replace the current E Bar average disintegration energy definition. In addition, the current dose equivalent I-131 definition would be revised to allow the use of additional thyroid dose conversion factors.

A001
NRR

Additionally, I&M proposes to revise the CNP Units 1 and 2 licensing basis and TS to adopt the alternative source term (AST) as allowed in 10 CFR 50.67. This amendment request represents full implementation of the AST as described in U. S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0. Full implementation revises the plant licensing basis to specify the AST is used in place of the previous source term for calculating the radiological consequences of design basis accidents (DBA). I&M performed AST analyses for the six DBAs identified in RG 1.183 that could potentially result in significant control room (CR) and off-site doses. These include the loss of coolant accident, main steam line break accident, fuel handling accident, steam generator tube rupture, reactor coolant pump locked rotor, and control rod ejection accident. The analyses for the accidents described in RG 1.183 demonstrate that use of AST methodologies maintains post-accident CR and off-site doses within regulatory acceptance limits. During evaluation of DBAs for AST methodology, rupture accidents for the waste gas decay tank and volume control tank were also evaluated. The analyses for these two accidents will continue to use AST for CR habitability but will not implement AST for off-site dose consequences.

The proposed RCS gross specific activity changes are consistent with NRC-approved Industry Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to Reactor Coolant System Specific Activity Technical Specification" (Reference 1). The availability of this TS improvement was announced in the Federal Register on March 19, 2007 (Reference 2), as part of the Consolidated Line Item Improvement Process.

The proposed changes related to AST are consistent with RG 1.183, Regulatory Information Summary 2006-04, "Experience with Implementation of Alternative Source Terms," and Section 15.0.1 of the Standard Review Plan, "Radiological Consequence Analyses Using Alternative Source Terms." This proposed amendment will modify TS requirements related to the use of an AST in analyzing off-site and CR accident dose consequences. Upon approval, I&M will implement this proposed change through a revision to the CNP licensing basis, including the TS and associated Bases. TS Bases changes are informational only and, upon approval, will be made in accordance with the TS Bases Control Program. Conforming changes will be made to the CNP Updated Final Safety Analysis Report (UFSAR) and subsequently submitted to the NRC as part of the regular UFSAR update process in accordance with 10 CFR 50.71(e). I&M's previously approved CR habitability dose analysis based on AST is being revised with an updated analysis for this submittal.

The following enclosures are provided with this letter:

- Enclosure 1, Affirmation
- Enclosure 2, Evaluation of Proposed Changes
- Enclosure 3, CNP Unit 1 TS Pages Marked to Show Proposed Changes
- Enclosure 4, CNP Unit 1 TS Bases Pages Marked to Show Proposed Changes
- Enclosure 5, CNP Unit 1 Final TS Pages
- Enclosure 6, CNP Unit 2 TS Pages Marked to Show Proposed Changes.
- Enclosure 7, CNP Unit 2 TS Bases Pages Marked to Show Proposed Changes
- Enclosure 8, CNP Unit 2 Final TS Pages

- Enclosure 9, D. C. Cook AST Radiological Analyses Technical Report, prepared by Red Wolf Associates, August 15, 2014
- Enclosure 10, D. C. Cook AST Regulatory Guide 1.183 Compliance Matrix
- Enclosure 11, D. C. Cook AST Regulatory Issue Summary 2006-04 Compliance Matrix
- Enclosure 12, D. C. Cook AST Accident Analyses Input Values Comparison Tables
- Enclosure 13, D. C. Cook AST Accident Analyses Meteorological Data

On August 8, 2014, a pre-submittal public meeting was held between I&M and the NRC in which no regulatory decisions or commitments were made (Reference 3). In this meeting, I&M agreed to provide meteorological data and a comparison of input values related to the analyses. This information is included in Enclosures 12 and 13.

I&M requests approval of the proposed license amendment by October 2015. Once approved, the amendment will be implemented within 180 days.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated Michigan state officials.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Joel P.Gebbie
Site Vice President

TLC/amp

Enclosures:

1. Affirmation
2. Evaluation of Proposed Changes
3. CNP Unit 1 TS Pages Marked to Show Proposed Changes
4. CNP Unit 1 TS Bases Pages Marked to Show Proposed Changes
5. CNP Unit 1 Final TS Pages
6. CNP Unit 2 TS Pages Marked to Show Proposed Changes
7. CNP Unit 2 TS Bases Pages Marked to Show Proposed Changes
8. CNP Unit 2 Final TS Pages
9. D. C. Cook AST Radiological Analyses Technical Report, prepared by Red Wolf Associates, August 15, 2014
10. D. C. Cook AST Regulatory Guide 1.183 Compliance Matrix
11. D. C. Cook AST Regulatory Issue Summary 2006-04 Compliance Matrix
12. D. C. Cook AST Accident Analyses Input Values Comparison Tables
13. D. C. Cook AST Accident Analyses Meteorological Data

c: M. L. Chawla, NRC Washington DC
J. T. King - MPSC
MDEQ- RMD/RPS
NRC Resident Inspector
C. D. Pederson, NRC Region III
A. J. Williamson - AEP Ft. Wayne, w/o enclosures

Enclosure 1 to AEP-NRC-2014-65

AFFIRMATION

I, Joel P. Gebbie, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the U. S. Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Joel P. Gebbie
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 14 DAY OF November 2014

Notary Public

DANIELLE BURGOYNE
Notary Public, State of Michigan
County of Berrien
My Commission Expires 04-04-2018
Acting in the County of Berrien

My Commission Expires 04-04-2018

Enclosure 2 to AEP-NRC-2014-65

Evaluation of Proposed Changes

1.0 DESCRIPTION

This letter is a request to amend Operating License Numbers DPR-58 and DPR-74 for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, respectively. In this request, Indiana Michigan Power Company (I&M) proposes to implement Technical Specification Task Force (TSTF) change traveler TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec" and adopt Alternative Source Term (AST) for control room (CR) habitability and off-site dose consequence analyses at CNP.

TSTF-490

The proposed changes would replace the current limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would be based on DOSE EQUIVALENT Xe-133 and would take into account only the noble gas activity in the primary coolant. The changes were approved by the U. S. Nuclear Regulatory Commission (NRC) staff Safety Evaluation (SE), dated March 19, 2007 (72 FR 12838) (Reference 1). TSTF change traveler TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," was announced for availability in Reference 1 as part of the consolidated line item improvement process (CLIIP). By memorandum from the Chief, Licensing Processes Branch, to the Plant Licensing Branch Chiefs, dated March 14, 2012, the NRC staff indicated that license amendment requests (LAR) related to TSTF-490 can be accepted for review, but will be handled through the normal LAR review process, instead of the expedited six-month CLIIP schedule.

AST

The proposed amendment will also modify the CNP licensing basis for Units 1 and 2 to adopt the use of an AST associated with accident off-site and CR dose consequences pursuant to Section 50.67 of Title 10 of the Code of Federal Regulations (10 CFR). This submittal represents full implementation of AST in accordance with NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0 (Reference 4). The current licensing bases for the accident radiological consequences analyses for off-site dose are based on source methodologies and assumptions derived from Technical Information Document-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." I&M previously submitted a request and received NRC approval for a Technical Specification (TS) amendment to implement AST methodology for the CR habitability dose analysis (Reference 3). However, the CR habitability dose analysis is being revised with updated information for this submittal. This submittal does not request adoption of AST for environmental qualification of safety-related equipment.

In order to minimize the deviations from TSTF-490, this submittal will follow the format of the TSTF model application in describing proposed changes for both TSTF-490 and AST.

2.0 PROPOSED CHANGES

TSTF-490

Consistent with NRC-approved TSTF-490, Revision 0, the proposed TS changes:

- Revise the definition of DOSE EQUIVALENT I-131. (Note: Because this submittal also contains a request to implement Alternative Source Term methodology, the definition of DOSE EQUIVALENT I-131 is based on the Committed Dose Equivalent or Committed Effective Dose Equivalent dose conversion factors.)
- Delete the definition of "E Bar, AVERAGE DISINTEGRATION ENERGY."
- Add a new TS definition for DOSE EQUIVALENT Xe-133.
- Revise LCO 3.4.16, "RCS Specific Activity" to delete references to gross specific activity; add limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133; and delete Figure 3.4.16–1, "Reactor Coolant DOSE EQUIVALENT I–131 Specific Activity Limit versus Percent of RATED THERMAL POWER."
- Revise LCO 3.4.16 "Applicability" to specify the LCO is applicable in MODES 1, 2, 3, and 4.
- Modify ACTIONS Table as follows:
  - A. Condition A is modified to delete the reference to Figure 3.4.16-1, and define an upper limit that is applicable at all power levels.
  - B. Condition B is added to provide a Condition and Required Action for DOSE EQUIVALENT Xe-133 instead of gross specific activity. The Completion Time is 48 hours. A Note allowing the applicability of LCO 3.0.4.c is also added, consistent with the Note to Required Action A.1.
  - C. Condition C (was condition B) is modified based on the changes to Conditions A and B and to reflect the change in the LCO Applicability.
- Revise SR 3.4.16.1 to verify the limit for DOSE EQUIVALENT Xe-133. A Note is added, consistent with SR 3.4.16.2 to allow entry into MODES 2, 3, and 4 prior to performance of the SR.
- Delete SR 3.4.16.3.

The following clarifying deviations from the TSTF-490 model application published in the FR notice are being made:

1) Reference to the NRC staff SE, dated September 27, 2006 (ADAMS ML062700612), is changed to refer to the NRC staff SE in Reference 1, because the SE dated September 27, 2006, that is referred to in the model application, is not publically available. The SE posted in the Federal Register on March 19, 2007, is publically available and approved for use.

2) For Surveillance Requirement (SR) 3.4.16.1, the TSTF-490 proposed note "Only required to be performed in Mode 1" will not be added.

3) The existing note for SR 3.4.16.2 will be deleted, consistent with other licensees that have adopted TSTF-490.

AST

In addition to the changes based on TSTF-490, the following changes to TS are proposed in conjunction with the implementation of AST methodology:

- The accident induced leakage performance criterion established by the Steam Generator (SG) Program in Section 5.5.7.b.2 is revised to clarify that SG primary to secondary leakage is limited to 0.25 gallon per minute (gpm) in any one SG, for a total leakage limit of 1.0 gpm from all SGs.
- Section 5.5.9.c. is revised to increase the maximum allowable methyl iodide penetration for the CR Emergency Ventilation charcoal adsorber from 1 percent (%) to 2.5%.
- The maximum allowable leakage rate, L_a at P_a, specified by the Containment Leakage Rate Program in Section 5.5.14.c is reduced from 0.25% per day to 0.18% per day.

3.0 BACKGROUND

TSTF-490

The background for the TSTF-490 portion of this application is as stated in the model SE in the NRC Notice of Availability, published on March 19, 2007 (72 FR 12838), the NRC Notice for Comment, published on November 20, 2006 (71 FR 67170), and TSTF-490, Revision 0.

AST

I&M is requesting to implement the AST methodology for CNP radiological dose consequence analyses for off-site dose analyses and update the dose analyses for CR habitability.

For the off-site doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), the current CNP licensing basis for the radiological consequences analyses of accidents discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) is based on the criteria stated in 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion 19, and the methodologies prescribed in RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors."

By separate amendment request, I&M previously submitted a request and received NRC approval for a TS amendment to implement AST methodology for the CR habitability dose analysis (Reference 3). Therefore, the current licensing basis for the CR radiological consequence analysis is 10 CFR 50.67, Accident Source Term. The UFSAR Chapter 14 design basis accident (DBA) analyses are being revised to fully implement the AST methodology for off-site dose consequences and update the CR analyses using current information.

Implementation of AST is being initiated, in part, because of technical discrepancies that have been identified in the current licensing basis dose consequence analyses described in the UFSAR. To resolve these discrepancies, I&M is submitting this LAR pursuant to 10 CFR 50.67 to incorporate AST methodology into the dose consequence calculations and licensing basis. Full implementation revises the plant licensing basis to specify the AST is used in place of the

previous source term for calculating the radiological consequences of DBAs. These analyses for CR habitability and off-site radiological dose consequences are being updated using the AST methodology outlined in RG 1.183. The new dose analyses being submitted for NRC review and approval include the following accident scenarios:

- Loss of Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor Accident
- Control Rod Ejection (CRE)
- Waste Gas Decay Tank (WGDT) Rupture*
- Volume Control Tank (VCT) Rupture*

(* Rupture accidents for the waste gas decay tank and the volume control tank were evaluated but AST will not be implemented for off-site dose consequence analyses. AST methodology will continue to be applied for CR dose analyses related to these two accidents.)

RG 1.183 recommends submitting changes to the UFSAR that reflect the revised analyses or the actual calculation documentation to the NRC staff. In lieu of providing the NRC staff with proposed UFSAR changes or supporting DBA calculations, I&M is providing DBA calculation summary information in Enclosure 9 and input values in Enclosures 12 and 13 to this letter. Upon issuance of a license amendment, conforming UFSAR changes will be completed as required by CNP procedures and submitted to the NRC staff in accordance with the regular UFSAR update process as required by 10 CFR 50.71(e).

Implementation of TSTF-490 and AST for CNP Units 1 and 2 does not include other programs that involve determination of integrated dose. Examples of programs, processes, or procedures that will not use TSTF-490 definitions or AST include, but are not limited to, Environmental Qualification of safety related equipment, Emergency Response Facility habitability, and the Off-Site Dose Calculation Manual.

4.0 TECHNICAL ANALYSIS

TSTF-490

I&M has reviewed References 1 and 2, and the model SE published in Reference 1. I&M has applied the methodology in Reference 1 to develop the proposed TS changes. I&M has concluded that the justifications presented in TSTF-490, Revision 0 and the model SE prepared by the NRC staff are applicable to CNP Units 1 and 2, and justify this amendment for the incorporation of the changes to the CNP Unit 1 and Unit 2 TS.

AST

Pursuant to 10 CFR 50.67, I&M is submitting this LAR for approval to implement AST. This amendment request was prepared using the guidance of RG 1.183 (Reference 4), with

additional guidance provided in Regulatory Issue Summary (RIS) 2006-04, "Experience with Implementation of Alternative Source Term." There are no plant modifications or changes to plant operation associated with implementation of AST at CNP.

I&M has performed analyses to support a full implementation of the AST as defined in RG 1.183. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the Total Effective Dose Equivalent (TEDE) dose as the new acceptance criteria. Full implementation of an AST will upgrade the existing radiological analyses by considering the impact of all five characteristics of the AST; radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release.

To develop the information required for this submittal, I&M employed the vendor Red Wolf Associates to conduct the radiological analyses. Their report, which provides technical information regarding the five characteristics of AST and forms the basis for the new analyses, is included as Enclosure 9 to this letter. Preparation of that report adhered to the main text and the appropriate appendices of RG 1.183. This report is supplemented with compliance matrixes for RG 1.183 and RIS 2006-04, presented in Enclosures 10 and 11 to this letter, respectively. In addition, a table that compares the current licensing basis input values to the new input values for AST is provided in Enclosure 12 to this letter.

The revised design basis dose analyses were performed using the direction of RG 1.183 with additional guidance provided in RIS 2006-04. This single set of analyses is applicable to both Units 1 and 2, with a limiting set of inputs applied to the analyses that is bounding for both units. Consistent with that approach, releases from either unit consider the dose impact on all receptor locations applicable to both units. The following UFSAR Chapter 14 accidents are evaluated:

- LOCA
- FHA
- MSLB
- SGTR
- Locked Rotor
- CRE

Additionally, accidents involving WGDT rupture and VCT rupture were evaluated even though RG 1.183 does not include guidance for the analysis of the WGDT and VCT rupture accidents. These events were evaluated for completeness to assess the CR TEDE doses for these events against the acceptance criteria provided in 10 CFR 50.67. The accident analyses for WGDT rupture and VCT rupture will not be revised to implement AST for off-site dose consequences. However, the CR dose consequence analyses for WGDT and VCT rupture events will continue to use the AST methodology and satisfy the acceptance criteria of 10 CFR 50.67.

The D. C. Cook AST Radiological Analysis Technical Report (Enclosure 9 to this letter) provides all the technical information necessary to conduct an evaluation of the radiological analyses. The report includes a description of the development of source terms, updated meteorological data, revision of atmospheric dispersion factors, analysis of DBAs for AST, and the computer

codes used in the analyses. The report forms the basis for this change request. A brief description of each of the report components is provided below.

Source Term

A new source term has been developed for the reactor core using the ORIGEN-ARP code described below. The fuel handling accident source term is derived from the core source term, modifying the value based on the number of fuel assemblies and radial peaking factor. The Reactor Coolant System (RCS) source term is established with input from the reactor core source term and modeling in the GOTHIC code.

Meteorological Data and Atmospheric Dispersion Factors

During development of the new source term for these accidents, the atmospheric dispersion factors (X/Q) utilized as inputs in the accident analyses were also revised. The dose analyses address releases from either unit and must consider the dose impact on all receptor locations applicable to both units. As such, X/Qs are developed for all possible release-receptor pairs, and the values applied in the analysis reflect the most limiting combination without regard to the unit in which the event occurs.

In developing the X/Qs, five years' worth of meteorological data is used, which meets the guidance set forth in Regulatory Position 3.1 of RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." The years of meteorological data provided are based on the five most recent years that have valid data for both the primary and shoreline towers. For these analyses, the years of 2002, 2004, 2005, 2007, and 2010 are the most recent years for which valid data is available. The raw data for the years listed is captured in a spreadsheet for use with ARCON96 and PAVAN. This meteorological data is provided on a compact disc as Enclosure 13 to this letter.

Prior to use, the raw meteorological data was examined to identify and flag bad or missing data to ensure that the data used in the X/Q determination were of high quality. However, neither RG 1.194 nor NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," provides guidance on the valid meteorological data recovery rate required for use in determining onsite X/Q values. Therefore, other regulatory guidance, which specifies a 90% data recovery threshold for measuring and capturing meteorological data, was used. The 90% data recovery rate applies to the composite of all variables needed to model atmospheric dispersion for each potential release pathway.

Accident Analysis

CR and off-site doses are calculated for the various events using the methodology outlined in Appendixes A, B, E, F, G, and H of RG 1.183. The dose contribution from the different radionuclide release characteristics is determined and then combined to obtain the total dose for the event.

Computer Codes

The following computer codes are used in performing the CNP radiological dose analyses:

RADTRAD is used to determine the control room and off-site doses for each analyzed event using the source term and X/Q inputs. The code considers the release timing, filtration, hold-up, and chemical form of the nuclides released into the environment.

ARCON96 ((NUREG/CR-6331) is used to determine the X/Qs at the CR intakes for selected release locations from plant meteorological data.

PAVAN provides X/Qs for various time periods at the EAB and LPZ boundaries using plant meteorological data.

JFREQ is a program in the METD suite of programs that is used to compute the joint frequency distribution of wind speed, wind direction, and atmospheric stability class for use as input to the PAVAN program.

MicroShield is used to determine the direct shine dose to the operators in the CR from the activity on the CR ventilation system filters.

ORIGEN-ARP calculates the fission product isotopic activity of the reactor core used in the development of the core and RCS source terms.

The GOTHIC code is used to simulate the RCS purification system to determine the relative concentrations of nuclides in the reactor coolant, and is also used to calculate the time-dependent refueling water storage tank temperature due to back leakage from the containment sump.

5.0 REGULATORY ANALYSIS

Precedent

The NRC has previously approved LARs for other plants to implement TSTF-490 and AST.

Submittals by the following plants to request implementation of TSTF-490 were reviewed, along with the corresponding requests for additional information (RAI). In addition, the letters for issuance of amendment were also reviewed to establish the final version of the approved amendment.

- Arkansas Nuclear One
- Braidwood Station and Byron Station
- Duke Energy Carolinas
- Palo Verde Nuclear Generating Station
- Prairie Island Nuclear Generating Plant

As a result of those reviews, a deviation was taken from TSTF-490 related to mode applicability of surveillance requirements.

Submittals by the following plants to request implementation of AST were reviewed, along with the corresponding RAIs. In addition, the letters for issuance of amendment were also reviewed to establish the final version of the approved amendment.

- Arkansas Nuclear One
- McGuire Nuclear Station
- Prairie Island Nuclear Generating Plant
- Turkey Point
- Virgil C. Summer Nuclear Station

As a result of those reviews, along with the pre-submittal public meeting between the NRC and CNP staff, the need to prepare compliance matrixes for RG 1.183 and RIS 2006-04 was reinforced. In addition, the reviews and meeting indicated that a comparison table of old and new values would be helpful to the reviewers, and is therefore included as Enclosure 12 to this letter.

TSTF-490

A description of these proposed changes and their relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published in References 1 and 2, and TSTF-490, Revision 0.

AST

I&M proposes to revise the CNP Unit 1 and Unit 2 licensing bases and TS to adopt the AST as allowed in 10 CFR 50.67. This amendment request represents full implementation of the AST as described in RG 1.183. I&M performed AST analyses for the six DBAs identified in RG 1.183 that could potentially result in significant CR and off-site doses. These include the LOCA, the MSLB accident, FHA, SGTR, RCP locked rotor, and the CRE accident. AST analyses have also been performed for WGDT rupture and VCT rupture for evaluation of CR habitability.

Justification for the TS changes that are proposed to implement AST methodology is discussed below:

TS Section 5.5.7.b.2

- The accident induced leakage performance criterion established by the SG Program is revised to clarify that SG primary to secondary leakage is limited to 0.25 gpm in any one SG and 1.0 gpm total for all SGs.

  Justification: The 1.0 gpm limit that previously existed in this TS is unchanged. This is simply a clarification that the 1.0 gpm is evenly divided among the four SGs.

TS Section 5.5.9.c.

- CR Emergency Ventilation charcoal adsorber is revised to increase the maximum allowable methyl iodide penetration from 1% to 2.5%.

Justification: The basis for the penetration value change is the dose analyses. Using a value of 95% (adjusted to 94.05% for filter bypass) for elemental and organic filter efficiency provides justification for increasing the penetration value. The value of 95% filter efficiency represents a safety factor of two (i.e., 2 x 2.5%) per RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants."

TS Section 5.5.14.c

- The maximum allowable leakage rate, L_a at P_a, specified by the Containment Leakage Rate Program is reduced from 0.25% per day to 0.18% per day.

Justification: The maximum allowable containment leakage value is being lowered to 0.18% to provide margin for the dose analyses. This is a more restrictive value, and therefore more conservative value, for testing.

6.0 NO SIGNIFICANT HAZARDS CONSIDERATION

TSTF-490

I&M has reviewed the proposed no significant hazards consideration determination published in Reference 1 as part of the CLIIP. I&M has concluded that the proposed determination presented in the notice is applicable to CNP Units 1 and 2 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

AST

As provided by 10 CFR 50.67, I&M is implementing the use of an AST and the dose calculation methodology described in RG 1.183 to calculate accident doses to CR and off-site personnel following postulated events that result in the release of radioactive material from the reactor fuel. The AST and associated methodology define the amount, isotopic composition, physical and chemical characteristics, and timing of radioactive material releases following postulated events. Transport of the material to the CR and off-site is modeled, and the resulting TEDE is determined. Regulatory acceptance criteria account for the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). In accordance with 10 CFR 50.67(b), licensees wishing to adopt an AST must apply for a license amendment in accordance with 10 CFR 50.90.

In support of the revised analysis applying AST, TS limits for RCS and secondary system specific activity are being revised as part of this amendment request.

As required by 10 CFR 50.91(a), the CNP analysis of the issue of no significant hazards consideration for adoption of AST is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

There are no physical changes to the plant being introduced by the proposed changes to the accident source term. Implementation of AST and the associated proposed TS changes and new atmospheric dispersion factors have no impact on the probability for initiation of any DBAs. Once the occurrence of an accident has been postulated, the new accident source term and atmospheric dispersion factors are an input to analyses that evaluate the radiological consequences. The proposed changes do not involve a revision to the design or manner in which the facility is operated that could increase the probability of an accident previously evaluated in Chapter 14 of the UFSAR.

Based on the AST analyses, there are no proposed changes to performance requirements and no proposed revision to the parameters or conditions that could contribute to the initiation of an accident previously discussed in Chapter 14 of the UFSAR. Plant-specific radiological analyses have been performed using the AST methodology and new X/Qs have been established. Based on the results of these analyses, it has been demonstrated that the CR and off-site dose consequences of the limiting events considered in the analyses meet the regulatory guidance provided for use with the AST, and the doses are within the limits established by 10 CFR 50.67.

Therefore, it is concluded that the proposed amendment does not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

No new modes of operation are introduced by the proposed changes. The proposed changes will not create any failure mode not bounded by previously evaluated accidents. Implementation of AST and the associated proposed TS changes and new X/Qs have no impact to the initiation of any DBAs. These changes do not affect the design function or modes of operation of structures, systems and components in the facility prior to a postulated accident. Since structures, systems and components are operated no differently after the AST implementation, no new failure modes are created by this proposed change. The alternative source term change itself does not have the capability to initiate accidents.

Consequently, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The AST analyses have been performed using approved methodologies to ensure that analyzed events are bounding and safety margin has not been reduced. Also, new X/Qs, which are based on site specific meteorological data, were calculated in accordance with the guidance of RG 1.194 to utilize more recent data and improved calculational methodologies. The dose consequences of these limiting events are within the acceptance criteria presented in 10 CFR 50.67. Thus, by meeting the applicable regulatory limits for AST, there is no significant reduction in a margin of safety.

Therefore, because the proposed changes continue to result in dose consequences within the applicable regulatory limits, the proposed amendment does not involve a significant reduction in margin of safety.

Based on the above, I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

Conclusion

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 NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 ENVIRONMENTAL EVALUATION

TSTF-490

I&M has reviewed the environmental consideration included in the model SE published in Reference 1 as part of the CLIIP. I&M has concluded that the staff's findings presented therein are applicable to CNP Units 1 and 2 and the determination is hereby incorporated by reference for this application.

AST

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released off-site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

8.0 REFERENCES

1. Federal Register Notice of Availability published on March 19, 2007 (72 FR 12838).
2. Federal Register Notice for Comment published on November 20, 2006 (71 FR 67170).
3. Letter from U. S. Nuclear Regulatory Commission to Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Units 1 And 2 - Issuance of Amendments TAC Nos. MB5318 AND MB5319)," dated November 14, 2002 (ML022980619).
4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, dated July 2000.

Enclosure 3 to AEP-NRC-2014-65

CNP Unit 1 TS Pages Marked to Show Proposed Changes

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or those listed in ICRP-30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source.
~~Ē - AVERAGE DISINTEGRATION ENERGY~~	~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits.~~ RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES ~~1 and 2,~~ 1, 2, 3, and 4.
~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.~~

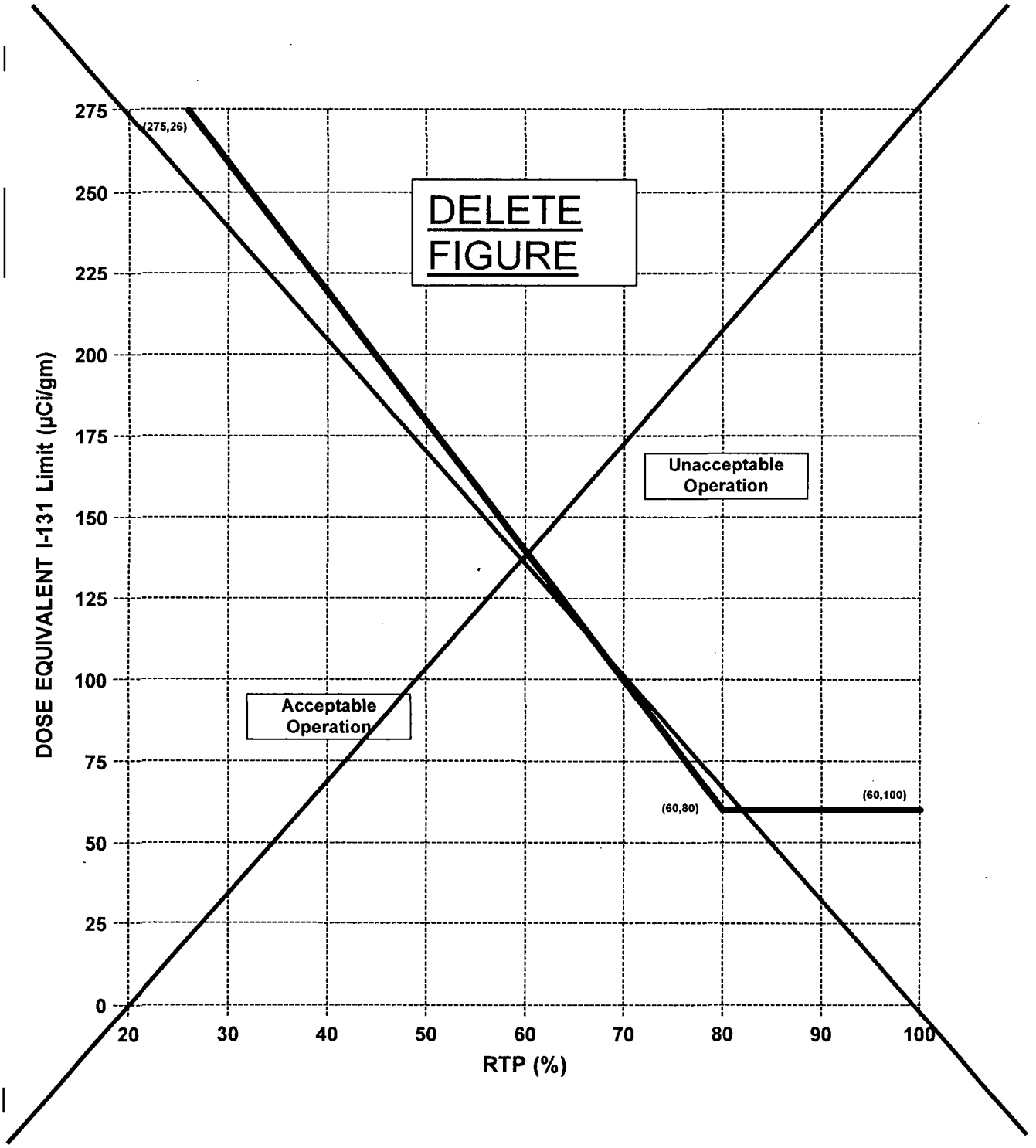
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 ~~> 1.0 µCi/gm.~~ not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1.~~ ≤ 60 µCi/gm.	Once per 4 hours
	AND	48 hours
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
~~B~~C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3 ~~with Tavg < 500°F~~.	6 hours
	AND	

CONDITION	REQUIRED ACTION	COMPLETION TIME
OR DOSE EQUIVALENT I- 131 ~~in the unacceptable region of Figure 3.4.16-1.~~ > 60 µCi/gm. ~~OR~~ ~~Gross specific activity of the reactor coolant not within limit.~~	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant ~~gross~~ DOSE EQUIVALENT XE-133 specific activity ≤ ~~100/Ē~~ 215.1 µCi/gm.	7 days
SR 3.4.16.2	~~NOTE~~ ~~Only required to be performed in MODE 1.~~	
	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
~~SR 3.4.16.3~~	~~NOTE~~ ~~Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	
	~~Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	~~184 days~~



~~Figure 3.4.16-1 (page 1 of 1)~~
~~Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity~~
~~Limit Versus Percent of RATED THERMAL POWER~~

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm for an individual SG, for a total leakage of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	~~1~~2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in October 1992.

   2. A one-time exception to the requirement to perform post-modification Type A testing is allowed for the steam generators and associated piping, as components of the containment barrier. For this case, ASME Section XI leak testing will be used to verify the leak tightness of the repaired or modified portions of the containment barrier. Entry into MODES 3 and 4 following the extended outage that commenced in 1997 may be made to perform this testing.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be ~~0.25~~0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Enclosure 4 to AEP-NRC-2014-65

CNP Unit 1 TS Bases Pages Marked to Show Proposed Changes

(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to Plant Specific Design Criterion (PSDC) 9, "Reactor Coolant Pressure Boundary" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The RCS, in conjunction with its control and protective provisions, was designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation or anticipated system interactions, and to maintain the stresses within allowable code stress limits. Also, in accordance with PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 1), the reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The design pressure of the RCS is 2485 psig. During normal operation and anticipated operational transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" (Ref. 4).

BASES

APPLICABILITY	SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.
SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100~~50.67, ~~"Reactor Site Criteria,"~~ limits (Ref. 4). The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.
REFERENCES	1. UFSAR, Sections 1.4.2 and 1.4.6. 2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000. 3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000. 4. 10 CFR ~~100~~50.67. 5. UFSAR, Section 7.2. 6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

BASES

APPLICABLE SAFETY ANALYSES (continued)

post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

c. Rod ejection.

Each of these events is discussed below.

The boron dilution analysis covers operation during shutdown, refueling, startup, and power operation. The purpose of the analysis is to show that, from initiation of the event, sufficient time is available to allow the operator to determine the cause of the dilution and to take corrective action before the SDM is lost.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level, high pressurizer pressure, overtemperature ΔT, overpower ΔT, or pressurizer water level trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 3) and the boron dilution (Ref. 4) analyses are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100, "Reactor Site Criteria,"~~ 50.67 limits (Ref. 5). For the boron dilution accident, if the LCO is violated, the time

assumed for operator action to terminate dilution may no longer be applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

b. Bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC); and

h. Boron penalty (MODES 4 and 5 only).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The boron penalty must be applied in MODES 4 and 5 since all reactor coolant pumps may be stopped in these MODES. This extra amount of boron ensures that minimum response times are met for the operator to diagnose and mitigate an inadvertent boron dilution event prior to loss of SDM.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. UFSAR, Section 1.4.5.

2. UFSAR, Chapter 14.

3. UFSAR, Section 14.2.5.

4. UFSAR, Section 14.1.5.

5. 10 CFR ~~100~~50.67.

BASES

BACKGROUND (continued)

the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

During anticipated operational transients, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50~~ and 10 CFR 100~~ criteria during anticipated operational transients.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into four distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 2), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the valves isolated by this instrumentation has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The Containment Purge Supply and Exhaust System isolation radiation monitors act as backup to the SI signal to ensure closing of the containment purge supply and exhaust valves. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR ~~100~~50.67 (Ref. 2) limits.

The Containment Purge Supply and Exhaust System isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge Supply and Exhaust System isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires one channel per train to be OPERABLE. The operator can initiate Containment Purge Supply and Exhaust System isolation at any time by using either of two switches (manual Containment Isolation - Phase A actuation or manual Containment Spray, Containment Isolation - Phase B actuation) in either Train "A" or Train "B" in the control room. Each switch actuates its associated train. This action will cause actuation of components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one switch and the interconnecting wiring to the actuation logic. These switches are common to ESFAS Containment Isolation, Phase A and B Manual Initiation switches.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS

BASES

SURVEILLANCE REQUIREMENTS (continued)

operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 24 months. The Frequency is acceptable based on instrument reliability and operating experience.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 24 months. Each Manual Initiation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., valves cycle).

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no setpoints associated with it.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every 24 months. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience.

REFERENCES

1. UFSAR, Section 5.5.3.

2. 10 CFR ~~100.11~~50.67

3. WCAP-15376, Rev. 0, October 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

Plant Specific Design Criterion 16 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~150 gpd per steam generator primary to secondary LEAKAGE as the initial condition.~~that primary to secondary LEAKAGE from an individual SG is

0.25 gpm (1.0 gpm for all SGs) as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. ~~and~~ To a lesser extent, primary to secondary LEAKAGE is a factor in the dose releases outside containment in other accidents or transients involving secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the steam generator power operated relief valves (and safety valves if their setpoint is reached) if offsite power is not available or if the condenser steam dump system fails to operate. The safety analysis for the SLB accident assumes ~~150 gpd~~ 0.25 gpm per steam generator (1.0 gpm for all SGs) primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from events resulting in a steam discharge to the atmosphere are within ~~a small fraction of~~ the limits defined in 10 CFR ~~100~~ 50.67~~and within GDC 19~~.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

The 0.8 gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air particulate monitoring equipment can detect within a reasonable time period. The limit is established for the pressurizer surge line in the leak before break methodology. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

~~The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.~~

~~The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.~~

~~The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~

~~The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY

~~The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small~~

ANALYSES

~~fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gpd per SG. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."~~

~~The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity.~~

~~The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent increase in iodine evolution that increases the I-131 activity in the reactor coolant based on an evolution rate that is 335 times the normal equilibrium rate for a spike duration of 8 hours after the accident.~~

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 0.25 gpm per SG (1 gpm for all SGs) exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.~~

~~The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.~~

~~The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves (if their setpoint is reached). The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.~~

~~The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131 and, for the concurrent iodine spike case, has a linear increasing DOSE EQUIVALENT I-131 level beginning immediately after the accident and reaching a maximum level in 8 hours.~~

~~The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.~~

~~RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~

The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 215.1 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a rise in pressure in the ruptured SG which causes radioactively contaminated steam to discharge to the atmosphere

through the power operated relief valves or the main steam safety valves. The atmospheric discharge stops when the primary to secondary leakage is halted via operator action. The unaffected SG removes core decay heat by venting steam until Residual Heat Removal (RHR) system entry conditions are reached.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until RHR system entry conditions are reached.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 µCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

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

LCO

~~The specific iodine activity is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of µCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design~~

The iodine specific activity in the reactor coolant is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 215.1 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

BASES

LCO (continued)

~~Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

~~The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.~~

APPLICABILITY

~~In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.~~

~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

~~With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that the limits of Figure 3.4.16-1 are not exceeded. An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.~~

~~The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.~~

~~A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.~~

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ 60.0 µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event that is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

BASES

ACTIONS (continued)

~~B.1~~

~~If any Required Action and associated Completion Time of Condition A is not met, if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, or if gross specific activity of the reactor coolant is not within limit, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging unit systems.~~

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60.0 µCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

~~SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.~~

~~Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.~~

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating

BASES

SURVEILLANCE REQUIREMENTS (continued)

conditions. The 7-day Frequency considers the low probability of a gross fuel failure during this time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

~~This Surveillance requires the verification that the reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. This Surveillance is accomplished by performing an isotopic analysis of a reactor coolant sample. This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.~~

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14-day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change > 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.4.16.3~~

~~A radiochemical analysis for Ē determination is required every 184 days with the unit operating in MODE 1 equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify unit operation within the specified gross activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes Ē does not change rapidly.~~

~~This SR has been modified by a Note that indicates sampling is not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.~~

REFERENCES

~~1. 10 CFR 100.11.~~

1. 10 CFR 50.67.

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

3.~~2.~~ UFSAR, Section 14.2.4.

4. UFSAR, Section 14.2.5.

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the ~~operational~~ accident induced LEAKAGE rate ~~limits in LCO 3.4.13, "RCS Operational LEAKAGE,"~~ of 0.25 gpm (1.0 gpm for all SGs) plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via the SG power operated relief valves.

The analysis for design basis accidents and transients other than an SGTR assumes the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on ~~150 gpd~~ the total primary to secondary LEAKAGE from an individual SG of 0.25 gpm (1.0 gpm for all SGs)~~per SG primary to secondary LEAKAGE as an initial condition.~~as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of ~~GDC 19 (Ref. 2),~~ 10 CFR ~~100~~ 50.67 (Ref. 3)~~ or the NRC approved licensing basis (e.g., a small fraction of these limits)~~.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the

evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed ~~150 gpd per SG~~ 0.25 gpm per SG (1.0 gpm for all SGs). The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. ~~10 CFR 50 Appendix A, GDC 19.~~Not Used

3. 10 CFR~~100~~50.67.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

BASES

BACKGROUND (continued)

d. The sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE (i.e., OPERABLE such that the containment leakage limits are met).

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA (Ref. 2) or a rod ejection accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment ~~was~~ is designed with an allowable leakage rate of ~~0.25~~0.18% of containment air weight per day (Ref. 4). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~0.25~~0.18% per day in the safety analysis at P_a = 12 psig (Ref. 4).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The Containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide local indication of door position. Additionally, a control room alarm is provided for each air lock to alert the operator whenever an air lock door is open for greater than approximately 5 minutes.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Refs. 1 and 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment ~~was~~ is designed with an allowable leakage rate of ~~0.25~~0.18% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 4), as L_a = ~~0.25~~0.18% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 12 psig following

BASES

APPLICABLE SAFETY ANALYSES (continued)

Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an SGSV to close.

The SGSVs serve only a closed safety function and remain open during power operation. These valves operate during a SLB and steam generator tube rupture.

The SGSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that four SGSVs in the steam lines be OPERABLE. The SGSVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the SGSVs will perform their design safety function to mitigate the consequences of accidents ~~that could result in~~ such that offsite exposures ~~comparable to a small fraction of~~ are less than 10 CFR ~~100~~ 50.67 (Ref. 3) limits.

APPLICABILITY

The SGSVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed, when there is significant mass and energy in the RCS and steam generators. When the SGSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low, thus the probability of a SLB is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the SGSVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one SGSV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the SGSV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SGSVs.

B.1

If the SGSV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each SGSV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of SGSV testing is every 24 months. The 24 month Frequency for testing is based on equipment reliability. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.2.
2. UFSAR, Section 14.2.5.
3. 10 CFR ~~100.11~~50.67.
4. Technical Requirements Manual
5. ASME, Operations and Maintenance Standards and Guides (OM Codes).

B 3.7 PLANT SYSTEMS

B 3.7.12 Engineered Safety Features (ESF) Ventilation System

BASES

BACKGROUND

The ESF Ventilation System filters air from the enclosures for the ESF equipment (containment spray pump, residual heat removal (RHR) pump, safety injection pump, RHR heat exchanger, containment spray heat exchanger, and reciprocating and centrifugal charging pump enclosures) during normal operation, transients, and accidents. The ESF Ventilation System, in conjunction with other systems, also provides adequate cooling in the ESF enclosure areas.

The ESF Ventilation System consists of two independent and redundant trains. Each train consists of a roll media roughing filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system.

The design of each train includes a bypass of the charcoal adsorber section. There are two independent air operated, fail-closed, dampers in the charcoal adsorber section bypass. These dampers are arranged in parallel. Normally, one train is in operation, directing the exhaust air through the roughing and HEPA filters, bypassing the charcoal adsorber section, and discharging it to the unit vent, while the other train is in standby. In the event of a Phase B isolation (Containment Pressure - High High) signal: a) for the standby train, the fan automatically starts (via a containment spray pump closed breaker signal); and b) for both the operating and standby trains, the charcoal adsorber section bypasses are automatically closed and the air is directed through the charcoal adsorber section in addition to the roughing and HEPA filters. The standby train also starts on any train related ESF system pump start signal, or upon receipt of a Safety Injection signal. The roughing filters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ESF Ventilation System is discussed in UFSAR, Section 9.9.3.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ESF Ventilation System is established by the large break LOCA. The system evaluation assumes leakage from the Emergency Core Cooling System (ECCS) and Containment Spray System components during the recirculation mode. ~~In such a case~~Although not credited in dose consequence analyses, the system will assist in ~~limits~~ limiting radioactive release to within the 10 CFR ~~100~~ 50.67 (Ref. 2) limits and to 5 rem total effective dose equivalent (TEDE) for control room

BASES

APPLICABLE SAFETY ANALYSES (continued)

operators (Ref. 3). The analysis of the effects and consequences of a large break LOCA is presented in Reference 4.

The ESF Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the ESF Ventilation System are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ESF enclosure areas exceeding 10 CFR ~~100~~ 50.67 limits in the event of a Design Basis Accident (DBA).

ESF Ventilation System is considered OPERABLE when the individual components necessary to maintain the ESF enclosure areas filtration are OPERABLE in both trains.

An ESF Ventilation System train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE and air flow can be maintained.

In addition, a train is allowed to be operating since, if a loss of power occurs, it will automatically restart when power is restored.

The LCO is modified by a Note allowing the ESF enclosure boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ESF enclosure isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESF Ventilation System is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ESF Ventilation System is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

BASES

ACTIONS

A.1

With one ESF Ventilation train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ESF Ventilation System function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

If the ESF enclosure boundary is inoperable, the ESF Ventilation trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ESF enclosure boundary within 24 hours. During the period that the ESF enclosure boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR ~~Part 100~~50.67 should be utilized to protect plant personnel from potential hazards. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ESF enclosure boundary.

C.1 and C.2

If the ESF Ventilation train or ESF enclosure boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every 92 days provides an adequate check on this system. Operating the ESF

BASES

REFERENCES

1. UFSAR, Section 9.9.3.1.
2. 10 CFR ~~100.11~~50.67.
3. 10 CFR 50, Appendix A, GDC 19.
4. UFSAR, Section 14.3.5.19.

BASES

APPLICABLE SAFETY ANALYSES

The FHAEV System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHAEV System is operating and the exhaust flow is directed through the charcoal adsorber section and the Fuel Handling Area Supply Air System fans are automatically shutdown upon receipt of a Fuel Handling Area Radiation - High signal. The amount of fission products available for release from the auxiliary building is determined for a fuel handling accident. These assumptions and the analysis follow the guidance discussed in the UFSAR (Ref. 2).

The FHAEV System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One train of the FHAEV System is required to be OPERABLE and in operation. The required FHAEV train is in operation when one fan is operating and all charcoal adsorber section bypass dampers are closed and inlet dampers are open. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR ~~100~~ 50.67 (Ref. 3) limits in the event of a fuel handling accident.

The FHAEV train is considered OPERABLE when the individual components necessary to control exposure in the auxiliary building are OPERABLE. Thus, the required FHAEV train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained; and

d. Fuel Handling Area Supply Air System fans must be capable of being stopped upon receipt of a Fuel Handling Area Radiation - High signal.

The LCO is modified by a Note allowing the auxiliary building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary building isolation is indicated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

This SR verifies that the required FHAEV System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum and maximum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.4

This SR verifies that the required FHAEV train actuates on an actual or simulated actuation signal. The test must verify that the signal automatically shuts down each of the Fuel Handling Area Supply Air System fans. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.13.5

This SR verifies the integrity of the auxiliary building enclosure. The ability of the pool storage area to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHAEV train. During the accident mode of operation, the FHAEV train is designed to maintain a slight negative pressure in the FHAEV train, to prevent unfiltered leakage. The FHAEV train is designed to maintain a pressure ≥ 0.125 inches of vacuum water gauge with respect to atmospheric pressure at a flow rate of ≤ 27,000 cfm. The Frequency of 24 months is consistent with industry practice and with other filtration system SRs.

REFERENCES

1. UFSAR, Section 9.9.3.2.

2. UFSAR, Section 14.2.1.

3. 10 CFR ~~100~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the UFSAR, Section 9.7.2 (Ref. 1). A description of the Spent Fuel Pool Cooling System is given in the UFSAR, Section 9.4 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 14.2.1 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in the UFSAR (Ref. 3). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is ~~a small fraction of~~ less than the 10 CFR ~~100~~ 50.67 (Ref. 4) limits. According to Reference 3, there is 23 ft of water above the top of the damaged fuel bundle during a fuel handling accident. With 23 ft of water, the assumptions discussed in Reference 3 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle (due to the width of the bundle). To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The Fuel Storage Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

BASES

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

REFERENCES

1. UFSAR, Section 9.7.2.
2. UFSAR, Section 9.4.
3. UFSAR, Section 14.2.1.
4. 10 CFR ~~100.11~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational transients, and accidents.

This limit is lower than the activity value that might be expected from a tube leak allowed by LCO 3.4.13, "RCS Operational LEAKAGE" of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

~~With the specified activity limit, the resultant thyroid dose to a person at the site boundary would be about 2.2 rem following a trip from full power coincident with a loss of offsite power and venting steam from the intact steam generators for 30 days.~~

Operating a unit at the allowable limits ~~could result~~ would not allow the ~~in a~~ 2 hour site boundary exposure or the control room exposure ~~of a small fraction of~~ to exceed the 10 CFR ~~100~~ 50.67 (Ref. 1) total effective dose equivalent (TEDE) limits ~~and a control room dose limit of 5 rem total effective dose equivalent (TEDE)~~.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Section 14.2.7 (Ref. ~~3~~2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 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 an MSLB do not exceed ~~a small fraction of~~ the unit site boundary limits (Ref. 1) for ~~whole body and thyroid~~ dose ~~rates~~ and a control room dose limit of 5 rem TEDE (Ref. ~~2~~1).

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) and steam generator (SG) power operated relief valves (PORVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to discharge steam and any entrained activity through the MSSVs and SG PORVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary Specific Activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to ~~a small fraction of~~ less than the required site boundary limit (Ref. 1) and a control room dose limit of 5 rem TEDE (Ref. ~~2~~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.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

Specific activity of the secondary coolant exceeding the allowable value is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR ~~100.11~~50.67.

~~2. 10 CFR 50, Appendix A, GDC 19.~~

~~3~~2. UFSAR, Section 14.2.7.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR ~~100~~50.67 (Ref. 2). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but at least one air lock door must always remain capable of being closed.

BASES

BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Containment Purge Supply and Exhaust System includes a 24 inch purge supply penetration and a 30 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge supply and exhaust penetrations are normally maintained closed. The Containment Purge Supply and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The Containment Purge Supply and Exhaust System is used for this purpose.

The containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 1, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 120 hours prior to irradiated fuel movement with containment closure capability, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are ~~a small fraction of~~ less than the ~~guideline~~ values specified in 10 CFR ~~100~~50.67 (Ref. 2).

Containment Penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge supply and exhaust penetrations and the containment personnel air locks. For the OPERABLE containment purge supply and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge Supply and Exhaust System. The OPERABILITY requirements for this LCO ensure that the automatic purge

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations is in its required status. The LCO 3.9.3.c.2 status requirement, which requires penetrations to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System, can be verified by ensuring each required valve operator is capable of closing automatically if needed. This Surveillance does not require cycling of the valves since this is performed at the appropriate frequency in accordance with SR 3.9.3.2.

The Surveillance is performed every 7 days during movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess ~~of a small fraction~~ of the ~~guideline~~ values specified in 10 CFR ~~100~~50.67 (Ref. 2).

SR 3.9.3.2

This Surveillance demonstrates that each required containment purge supply and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar valve testing requirements. LCO 3.3.6, "Containment Purge Supply and Exhaust System Isolation Instrumentation," provides additional Surveillance Requirements for the containment purge supply and exhaust valve actuation circuitry. Ensuring these Surveillances are met during movement of irradiated fuel assemblies within containment will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. UFSAR, Section 14.2.1.5.
2. 10 CFR 50.67

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~a small fraction of~~ less than the 10 CFR ~~100~~ 50.67 limits.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as described in the UFSAR (Ref. 1). A minimum water level of 23 ft assures an acceptable decontamination factor for iodine. The fuel handling accident analysis inside containment is described in Reference 1. With a minimum water level of 23 ft and a minimum decay time of 120 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 2). Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

BASES

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 14.2.1.

2. 10 CFR ~~100.10~~50.67.

Enclosure 5 to AEP-NRC-2014-65

CNP Unit 1 Final TS Pages

(For Information Only)

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External

1.1 Definitions

Exposure to Radionuclides in Air, Water, and Soil" or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	AND	
OR	C.2 Be in MODE 5.	36 hours
DOSE EQUIVALENT I-131 > 60 µCi/gm.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 215.1 μCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm in an individual SG, for a total leakage rate of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in October 1992.

   2. A one-time exception to the requirement to perform post-modification Type A testing is allowed for the steam generators and associated piping, as components of the containment barrier. For this case, ASME Section XI leak testing will be used to verify the leak tightness of the repaired or modified portions of the containment barrier. Entry into MODES 3 and 4 following the extended outage that commenced in 1997 may be made to perform this testing.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

Enclosure 6 to AEP-NRC-2014-65

CNP Unit 2 TS Pages Marked to Show Proposed Changes

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	~~DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites," those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or those listed in ICRP 30, Supplement to Part 1, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity".~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" or the average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source.
~~Ē - AVERAGE DISINTEGRATION ENERGY~~	~~Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 ~~The specific activity of the reactor coolant shall be within limits.~~ RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES ~~1 and 2,~~ 1, 2, 3, and 4.
~~MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.~~

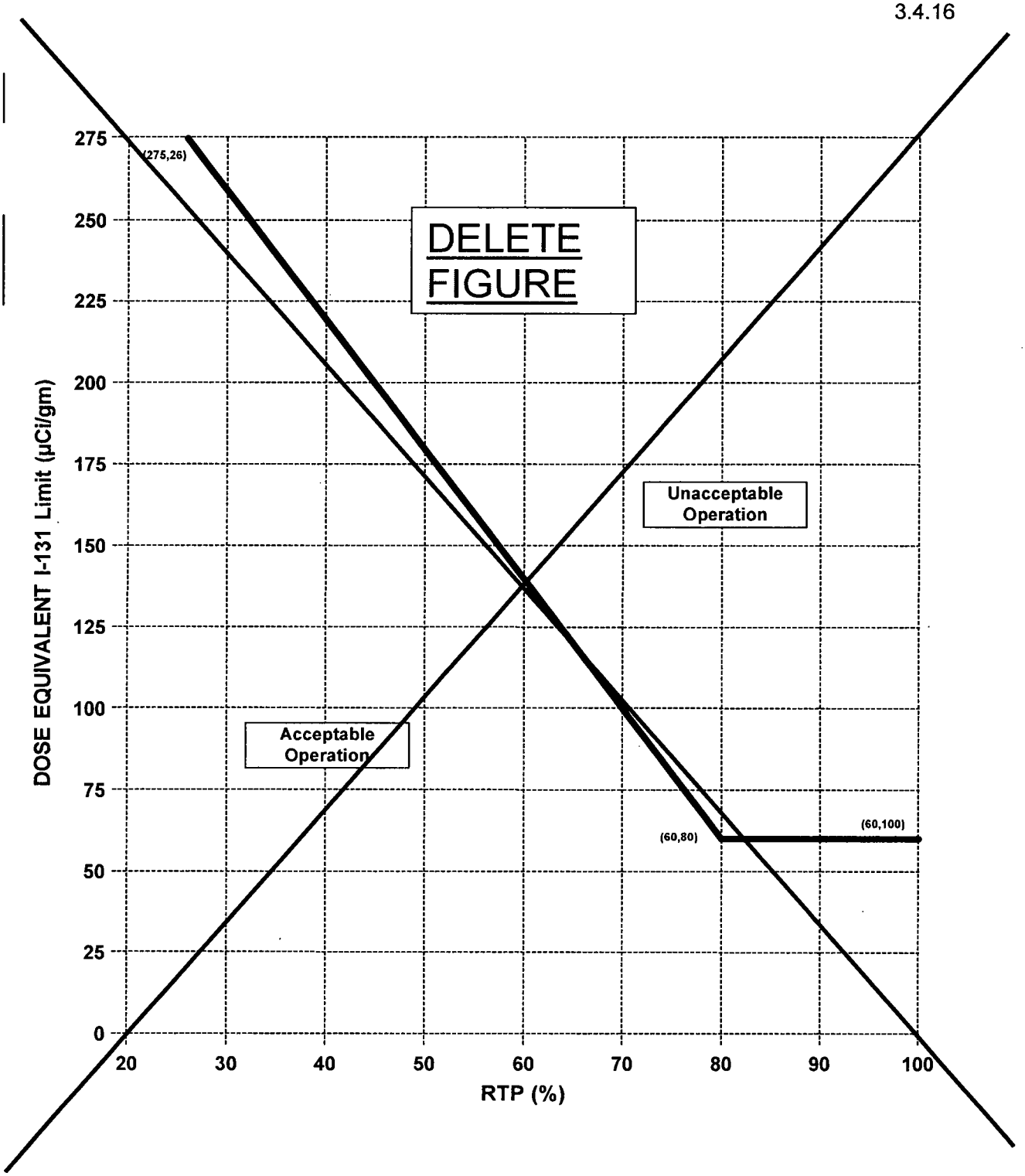
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 ~~> 1.0 µCi/gm~~ not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1.~~ ≤ 60 µCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
~~B~~C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3 ~~with Tavg < 500°F~~	6 hours
	AND	
OR	C.2 Be in MODE 5.	36 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
DOSE EQUIVALENT I-131 ~~in the unacceptable region of Figure 3.4.16-1.~~ > 60 µCi/gm. ~~OR~~ ~~Gross specific activity of the reactor coolant not within limit.~~		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant ~~gross~~ DOSE EQUIVALENT XE-133 specific activity ≤ ~~100/Ē~~ 215.1 µCi/gm.	7 days
SR 3.4.16.2	~~NOTE~~ ~~Only required to be performed in MODE 1.~~	
	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
~~SR 3.4.16.3~~	~~NOTE~~ ~~Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	
	~~Determine Ē from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.~~	~~184 days~~



~~Figure 3.4.16-1 (page 1 of 1)~~
~~Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity~~
~~Limit Versus Percent of RATED THERMAL POWER~~

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm in an individual SG, for a total leakage rate of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	~~1~~2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in May 1992.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be ~~0.25~~0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the Type B and C tests and ≤ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.15 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

Enclosure 7 to AEP-NRC-2014-657

CNP Unit 2 TS Bases Pages Marked to Show Proposed Changes

(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to Plant Specific Design Criterion (PSDC) 9, "Reactor Coolant Pressure Boundary" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. The RCS, in conjunction with its control and protective provisions, was designed to accommodate the system pressures and temperatures attained under the expected modes of plant operation or anticipated system interactions, and to maintain the stresses within allowable code stress limits. Also, in accordance with PSDC 33, "Reactor Coolant Pressure Boundary Capability" (Ref. 1), the reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The design pressure of the RCS is 2485 psig. During normal operation and anticipated operational transients, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR ~~100~~50.67, "~~Reactor Site Criteria~~Accident Source Term" (Ref. 4).

BASES

APPLICABILITY	SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.
SAFETY LIMIT VIOLATIONS	If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria," limits~~50.67 (Ref. 4). The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized. If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. UFSAR, Sections 1.4.2 and 1.4.6.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR ~~100~~50.67.
5. UFSAR, Section 7.2.
6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

BASES

APPLICABLE SAFETY ANALYSES (continued)

post trip return to power may occur; however, no fuel damage occurs as a result of the post trip return to power, and THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement must also protect against:

a. Inadvertent boron dilution;

b. An uncontrolled rod withdrawal from subcritical or low power condition; and

c. Rod ejection.

Each of these events is discussed below.

The boron dilution analysis covers operation during shutdown, refueling, startup, and power operation. The purpose of the analysis is to show that, from initiation of the event, sufficient time is available to allow the operator to determine the cause of the dilution and to take corrective action before the SDM is lost.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled rod withdrawal transient is terminated by either a high power level, high pressurizer pressure, overtemperature ΔT, overpower ΔT, or pressurizer water level trip. In all cases, power level, RCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

The ejection of a control rod rapidly adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The ejection of a rod also produces a time dependent redistribution of core power.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

SDM is a core design condition that can be ensured during operation through control rod positioning (control and shutdown banks) and through the soluble boron concentration.

The MSLB (Ref. 3) and the boron dilution (Ref. 4) analyses are the most limiting analyses that establish the SDM value of the LCO. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR ~~100, "Reactor Site Criteria,"~~50.67 limits (Ref. 5). For the boron dilution accident, if the LCO is violated, the time

assumed for operator action to terminate dilution may no longer be applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

b. Bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC); and

h. Boron penalty (MODES 4 and 5 only).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. The boron penalty must be applied in MODES 4 and 5 since all reactor coolant pumps may be stopped in these MODES. This extra amount of boron ensures that minimum response times are met for the operator to diagnose and mitigate an inadvertent boron dilution event prior to loss of SDM.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required SDM. This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. UFSAR, Section 1.4.5.

2. UFSAR, Chapter 14.

3. UFSAR, Section 14.2.5.

4. UFSAR, Section 14.1.5.

5. 10 CFR ~~100~~50.67.

BASES

BACKGROUND (continued)

the trip setpoint should be left adjusted to a value within the established trip setpoint calibration tolerance band, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned. If the actual setting of the device is found to have exceeded the Allowable Value the device would be considered inoperable from a Technical Specification perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required.

During anticipated operational transients, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

2. Fuel centerline melt shall not occur; and

3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50~~ and 10 CFR 100~~ criteria during anticipated operational transients.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR ~~100~~ 50.67 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RTS instrumentation is segmented into four distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 2), and as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the valves isolated by this instrumentation has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The Containment Purge Supply and Exhaust System isolation radiation monitors act as backup to the SI signal to ensure closing of the containment purge supply and exhaust valves. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR ~~100~~50.67 (Ref. 2) limits.

The Containment Purge Supply and Exhaust System isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge Supply and Exhaust System isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

   The LCO requires one channel per train to be OPERABLE. The operator can initiate Containment Purge Supply and Exhaust System isolation at any time by using either of two switches (manual Containment Isolation - Phase A actuation or manual Containment Spray, Containment Isolation - Phase B actuation) in either Train "A" or Train "B" in the control room. Each switch actuates its associated train. This action will cause actuation of components in the same manner as any of the automatic actuation signals.

   The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

   Each channel consists of one switch and the interconnecting wiring to the actuation logic. These switches are common to ESFAS Containment Isolation, Phase A and B Manual Initiation switches.

2. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

   Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS

BASES

SURVEILLANCE REQUIREMENTS (continued)

operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 24 months. The Frequency is acceptable based on instrument reliability and operating experience.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Initiation Function and is performed every 24 months. Each Manual Initiation Function is tested up to, and including, the master relay coils. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (i.e., valves cycle).

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no setpoints associated with it.

The Frequency is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.6.7

A CHANNEL CALIBRATION is performed every 24 months. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience.

REFERENCES

1. UFSAR, Section 5.5.3.

2. 10 CFR ~~100.11~~50.67

3. WCAP-15376, Rev. 0, October 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During unit life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

Plant Specific Design Criterion 16 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~150 gpd per steam generator (SG) primary to secondary LEAKAGE as the initial condition.~~that primary to secondary LEAKAGE from an individual SG is

0.25 gpm (1.0 gpm for all SGs) as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. ~~and~~ To a lesser extent, primary to secondary LEAKAGE is a factor in the dose releases outside containment in other accidents or transients involving secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the steam generator power operated relief valves (and safety valves if their setpoint is reached) if offsite power is not available or if the condenser steam dump system fails to operate. The safety analysis for the SLB accident assumes ~~150 gpd~~ 0.25 gpm per SG (1.0 gpm for all SGs) primary to secondary LEAKAGE as an initial condition. The dose consequences resulting from events resulting in a steam discharge to the atmosphere are within ~~a small fraction of~~ the limits defined in 10 CFR ~~100~~50.67. ~~and within GDC-19.~~

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air particulate monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

~~The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.~~

~~The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.~~

~~The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~

~~The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

The maximum dose that an individual at the exclusion area boundary can receive for 2 hours following an accident, or at the low population zone outer boundary for the radiological release duration, is specified in 10 CFR 50.67 (Ref. 1). Doses to control room operators must be limited per GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the dose consequences in the event of a steam line break (SLB) or steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133. The allowable levels are intended to ensure that offsite and control room doses meet the appropriate acceptance criteria in the Standard Review Plan (Ref. 2).

APPLICABLE SAFETY

~~The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small~~

ANALYSES

~~fraction of the 10 CFR 100 dose guideline limits following a SGTR accident. The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gpd per SG. The safety analysis assumes the specific activity of the secondary coolant at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."~~

~~The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity.~~

~~The analysis is for two cases of reactor coolant specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent increase in iodine evolution that increases the I-131 activity in the reactor coolant based on an evolution rate that is 335 times normal equilibrium rate for a spike duration of 8 hours after the accident.~~

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate SRP acceptance criteria following a SLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits, and an existing reactor coolant steam generator (SG) tube leakage rate of 0.25 gpm per SG (1 gpm for all SGs) exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of 0.1 μCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.17, "Secondary Specific Activity."

The analyses for the SLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at 1.0 μCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a SLB (by a factor of 500), or SGTR (by a factor of 335), respectively.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē μCi/gm for gross specific activity.~~

~~The analysis also assumes a loss of offsite power at the same time as the SGTR event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.~~

~~The coincident loss of offsite power causes the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG power operated relief valves and the main steam safety valves (if their setpoint is reached). The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends.~~

~~The safety analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has pre-accident iodine spiking levels up to 60.0 μCi/gm DOSE EQUIVALENT I-131 and, for the concurrent iodine spike case, has a linear increasing DOSE EQUIVALENT I-131 level beginning immediately after the accident and reaching a maximum level in 8 hours.~~

~~The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.~~

~~RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).~~

The second case assumes the initial reactor coolant iodine activity at 60.0 μCi/gm DOSE EQUIVALENT I-131 due to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be 215.1 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis assumes a rise in pressure in the ruptured SG which causes radioactively contaminated steam to discharge to the atmosphere

through the power operated relief valves or the main steam safety valves. The atmospheric discharge stops when the primary to secondary leakage is halted via operator action. The unaffected SG removes core decay heat by venting steam until Residual Heat Removal (RHR) system entry conditions are reached.

The SLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until RHR system entry conditions are reached.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed 60.0 µCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

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

LCO

~~The specific iodine activity is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of µCi/gm equal to 100 divided by Ē (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design~~

The iodine specific activity in the reactor coolant is limited to 1.0 µCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to 215.1 µCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate SRP acceptance criteria (Ref. 2).

The SLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a SLB or SGTR, lead to doses that exceed the SRP acceptance criteria (Ref. 2).

BASES

LCO (continued)

~~Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.~~

~~The SGTR accident analysis (Ref. 2) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.~~

APPLICABILITY

~~In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.~~

~~For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.~~

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a SLB or SGTR to within the SRP acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

~~With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to verify that the limits of Figure 3.4.16-1 are not exceeded. An isotopic analysis of a reactor coolant sample must be performed for at least I-131, I-133, and I-135. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.~~

~~The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.~~

~~A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit remains at, or proceeds to power operation.~~

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ 60.0 µCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event that is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

BASES

ACTIONS (continued)

~~B.1~~

~~If any Required Action and associated Completion Time of Condition A is not met, if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, or if gross specific activity of the reactor coolant is not within limit, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging unit systems.~~

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 60.0 µCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

~~SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.~~

~~Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.~~

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in the noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7-day Frequency considers the low probability of a gross fuel failure during this time.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

A Note modifies the SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

SR 3.4.16.2

~~This Surveillance requires the verification that the reactor coolant DOSE EQUIVALENT I-131 specific activity is within limit. This Surveillance is accomplished by performing an isotopic analysis of a reactor coolant sample. This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.~~

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during normal operation and following fast power changes when iodine spiking is more apt to occur. The 14-day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change > 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

The Note modifies this SR to allow entry into and operation in MODE 4, MODE 3, and MODE 2 prior to performing the SR. This allows the Surveillance to be performed in those MODES, prior to entering MODE 1.

~~SR 3.4.16.3~~

~~A radiochemical analysis for Ē determination is required every 184 days with the unit operating in MODE 1 equilibrium conditions. The Ē determination directly relates to the LCO and is required to verify unit~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~operation within the specified gross activity LCO limit. The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes Ē does not change rapidly.~~

~~This SR has been modified by a Note that indicates sampling is not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.~~

REFERENCES

~~1. 10 CFR 100.11.~~
1. 10 CFR 50.67.

2. Standard Review Plan (SRP) Section 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms."

3. ~~2.~~ UFSAR, Section 14.2.4.

4. UFSAR, Section 14.2.5

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the ~~operational~~ accident induced LEAKAGE rate ~~limits in LCO 3.4.13, "RCS Operational LEAKAGE,"~~ of 0.25 gpm (1.0 for all SGs) plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via the SG power operated relief valves.

The analysis for design basis accidents and transients other than an SGTR assumes the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from an individual SG of 0.25 gpm (1.0 gpm for all SGs) ~~of 1 gallon per minute (gpm) or is assumed to increase to 1 gpm~~ as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of ~~GDC 19 (Ref. 2),~~ 10 CFR ~~100~~ 50.67 (Ref. 3) ~~or the NRC approved licensing basis (e.g., a small fraction of these limits)~~.

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. If a tube was determined to satisfy the plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

An SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.7, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the

evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than an SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed ~~150 gpd per SG~~ 0.25 gpm per SG (1.0 gpm for all SGs). The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. ~~10 CFR 50 Appendix A, GDC 19.~~Not Used

3. 10 CFR ~~100~~50.67.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

BASES

BACKGROUND (continued)

d. The sealing mechanism associated with each containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE (i.e., OPERABLE such that the containment leakage limits are met).

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA (Ref. 2) or a rod ejection accident (Ref. 3). In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment ~~was~~ is designed with an allowable leakage rate of ~~0.25~~0.18% of containment air weight per day (Ref. 4). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~0.25~~0.18% per day in the safety analysis at P_a = 12 psig (Ref. 4).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The Containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

Each personnel air lock is provided with limit switches on both doors that provide local indication of door position. Additionally, a control room alarm is provided for each air lock to alert the operator whenever an air lock door is open for greater than approximately 5 minutes.

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Refs. 1 and 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment ~~was~~ is designed with an allowable leakage rate of ~~0.25~~0.18% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 4), as L_a = ~~0.25~~0.18% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 12 psig following

BASES

APPLICABLE SAFETY ANALYSES (continued)

Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an SGSV to close.

The SGSVs serve only a closed safety function and remain open during power operation. These valves operate during a SLB, steam generator tube rupture, and feedwater line break.

The SGSVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that four SGSVs in the steam lines be OPERABLE. The SGSVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

This LCO provides assurance that the SGSVs will perform their design safety function to mitigate the consequences of accidents ~~that could result in~~ such that offsite exposures ~~comparable to a small fraction of~~ are less than 10 CFR ~~100~~ 50.67 (Ref. 3) limits.

APPLICABILITY

The SGSVs must be OPERABLE in MODE 1, and in MODES 2 and 3 except when closed, when there is significant mass and energy in the RCS and steam generators. When the SGSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low, thus the probability of a SLB is low.

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the SGSVs are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one SGSV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the SGSV can be made with the unit hot. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SGSVs.

B.1

If the SGSV cannot be restored to OPERABLE status within 8 hours, the unit must be placed in a MODE in which the LCO does not apply. To

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency is in accordance with the Inservice Testing Program.

This test is conducted in MODE 3 with the unit at operating temperature and pressure. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows a delay of testing until MODE 3, to establish conditions consistent with those under which the acceptance criterion was generated.

SR 3.7.2.2

This SR verifies that each SGSV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of SGSV testing is every 24 months. The 24 month Frequency for testing is based on equipment reliability. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.2.
2. UFSAR, Section 14.2.5.
3. 10 CFR ~~100.11~~50.67.
4. Technical Requirements Manual
5. ASME, Operations and Maintenance Standards and Guides (OM Codes).

B 3.7 PLANT SYSTEMS

B 3.7.12 Engineered Safety Features (ESF) Ventilation System

BASES

BACKGROUND

The ESF Ventilation System filters air from the enclosures for the ESF equipment (containment spray pump, residual heat removal (RHR) pump, safety injection pump, RHR heat exchanger, containment spray heat exchanger, and reciprocating and centrifugal charging pump enclosures) during normal operation, transients, and accidents. The ESF Ventilation System, in conjunction with other systems, also provides adequate cooling in the ESF enclosure areas.

The ESF Ventilation System consists of two independent and redundant trains. Each train consists of a roll media roughing filter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system.

The design of each train includes a bypass of the charcoal adsorber section. There are two independent air operated, fail-closed, dampers in the charcoal adsorber section bypass. These dampers are arranged in parallel. Normally, one train is in operation, directing the exhaust air through the roughing and HEPA filters, bypassing the charcoal adsorber section, and discharging it to the unit vent, while the other train is in standby. In the event of a Phase B isolation (Containment Pressure - High High) signal: a) for the standby train, the fan automatically starts (via a containment spray pump closed breaker signal); and b) for both the operating and standby trains, the charcoal adsorber section bypasses are automatically closed and the air is directed through the charcoal adsorber section in addition to the roughing and HEPA filters. The standby train also starts on any train related ESF system pump start signal, or upon receipt of a Safety Injection signal. The roughing filters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

The ESF Ventilation System is discussed in UFSAR, Section 9.9.3.1 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The design basis of the ESF Ventilation System is established by the large break LOCA. The system evaluation assumes leakage from the Emergency Core Cooling System (ECCS) and Containment Spray System components during the recirculation mode. ~~In such a case~~Although not credited in dose consequence analyses, the system will assist in limiting ~~limits~~ radioactive release to within the 10 CFR ~~100~~ 50.67 (Ref. 2) limits and to 5 rem total effective dose equivalent (TEDE) for control room

BASES

APPLICABLE SAFETY ANALYSES (continued)

operators (Ref. 3). The analysis of the effects and consequences of a large break LOCA is presented in Reference 4.

The ESF Ventilation System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the ESF Ventilation System are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. Total system failure could result in the atmospheric release from the ESF enclosure areas exceeding 10 CFR ~~100~~ 50.67 limits in the event of a Design Basis Accident (DBA).

ESF Ventilation System is considered OPERABLE when the individual components necessary to maintain the ESF enclosure areas filtration are OPERABLE in both trains.

An ESF Ventilation System train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE and air flow can be maintained.

In addition, a train is allowed to be operating since, if a loss of power occurs, it will automatically restart when power is restored.

The LCO is modified by a Note allowing the ESF enclosure boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for ESF enclosure isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, the ESF Ventilation System is required to be OPERABLE consistent with the OPERABILITY requirements of the ECCS.

In MODE 5 or 6, the ESF Ventilation System is not required to be OPERABLE since the ECCS is not required to be OPERABLE.

BASES

ACTIONS

A.1

With one ESF Ventilation train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ESF Ventilation System function.

The 7 day Completion Time is appropriate because the risk contribution is less than that for the ECCS (72 hour Completion Time), and this system is not a direct support system for the ECCS. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1

If the ESF enclosure boundary is inoperable, the ESF Ventilation trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ESF enclosure boundary within 24 hours. During the period that the ESF enclosure boundary is inoperable, appropriate compensatory measures consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR ~~Part 100~~50.67 should be utilized to protect plant personnel from potential hazards. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ESF enclosure boundary.

C.1 and C.2

If the ESF Ventilation train or ESF enclosure boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.12.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every 92 days provides an adequate check on this system. Operating the ESF

BASES

REFERENCES

1. UFSAR, Section 9.9.3.1.
2. 10 CFR ~~100.11~~50.67.
3. 10 CFR 50, Appendix A, GDC 19.
4. UFSAR, Section 14.3.5.19.

BASES

APPLICABLE SAFETY ANALYSES

The FHAEV System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FHAEV System is operating and the exhaust flow is directed through the charcoal adsorber section and the Fuel Handling Area Supply Air System fans are automatically shutdown upon receipt of a Fuel Handling Area Radiation - High signal. The amount of fission products available for release from the auxiliary building is determined for a fuel handling accident. These assumptions and the analysis follow the guidance discussed in the UFSAR (Ref. 2).

The FHAEV System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One train of the FHAEV System is required to be OPERABLE and in operation. The required FHAEV train is in operation when one fan is operating and all charcoal adsorber section bypass dampers are closed and inlet dampers are open. Total system failure could result in the atmospheric release from the fuel handling building exceeding the 10 CFR ~~100~~ 50.67 (Ref. 3) limits in the event of a fuel handling accident.

The FHAEV train is considered OPERABLE when the individual components necessary to control exposure in the auxiliary building are OPERABLE. Thus, the required FHAEV train is considered OPERABLE when its associated:

a. Fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained; and

d. Fuel Handling Area Supply Air System fans must be capable of being stopped upon receipt of a Fuel Handling Area Radiation - High signal.

The LCO is modified by a Note allowing the auxiliary building boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary building isolation is indicated.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.3

This SR verifies that the required FHAEV System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum and maximum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.13.4

This SR verifies that the required FHAEV train actuates on an actual or simulated actuation signal. The test must verify that the signal automatically shuts down each of the Fuel Handling Area Supply Air System fans. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.13.5

This SR verifies the integrity of the auxiliary building enclosure. The ability of the pool storage area to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FHAEV train. During the accident mode of operation, the FHAEV train is designed to maintain a slight negative pressure in the FHAEV train, to prevent unfiltered leakage. The FHAEV train is designed to maintain a pressure ≥ 0.125 inches of vacuum water gauge with respect to atmospheric pressure at a flow rate of ≤ 27,000 cfm. The Frequency of 24 months is consistent with industry practice and with other filtration system SRs.

REFERENCES

1. UFSAR, Section 9.9.3.2.

2. UFSAR, Section 14.2.1.

3. 10 CFR ~~100~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.14 Fuel Storage Pool Water Level

BASES

BACKGROUND	The minimum water level in the fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. A general description of the fuel storage pool design is given in the UFSAR, Section 9.7.2 (Ref. 1). A description of the Spent Fuel Pool Cooling System is given in the UFSAR, Section 9.4 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 14.2.1 (Ref. 3).
APPLICABLE SAFETY ANALYSES	The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in the UFSAR (Ref. 3). The resultant 2 hour ~~thyroid~~ dose per person at the exclusion area boundary is ~~a small fraction of~~ less than the 10 CFR ~~100~~ 50.67 (Ref. 4) limits. According to Reference 3, there is 23 ft of water above the top of the damaged fuel bundle during a fuel handling accident. With 23 ft of water, the assumptions discussed in Reference 3 can be used directly. In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle (due to the width of the bundle). To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop. The Fuel Storage Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The fuel storage pool water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for movement within the fuel storage pool.
APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the fuel storage pool, since the potential for a release of fission products exists.

BASES

ACTIONS

A.1

When the initial conditions for prevention of an accident cannot be met, steps should be taken to preclude the accident from occurring. When the fuel storage pool water level is lower than the required level, the movement of irradiated fuel assemblies in the fuel storage pool is immediately suspended to a safe position. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

This SR verifies sufficient fuel storage pool water is available in the event of a fuel handling accident. The water level in the fuel storage pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

REFERENCES

1. UFSAR, Section 9.7.2.

2. UFSAR, Section 9.4.

3. UFSAR, Section 14.2.1.

4. 10 CFR ~~100.11~~50.67.

B 3.7 PLANT SYSTEMS

B 3.7.17 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational transients, and accidents.

This limit is lower than the activity value that might be expected from a tube leak allowed by LCO 3.4.13, "RCS Operational LEAKAGE" of primary coolant at the limit of 1.0 µCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

~~With the specified activity limit, the resultant thyroid dose to a person at the site boundary would be about 2.2 rem following a trip from full power coincident with a loss of offsite power and venting steam from the intact steam generators for 30 days.~~

Operating a unit at the allowable limits ~~could result~~would not allow the ~~in a~~ 2 hour site boundary exposure or the control room exposure ~~of a small fraction of~~ to exceed the 10 CFR ~~100~~ 50.67 (Ref. 1) total effective dose equivalent (TEDE) limits ~~and a control room dose limit of 5 rem total effective dose equivalent (TEDE)~~.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Section 14.2.7 (Ref. ~~3~~2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 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 an MSLB do not exceed ~~a small fraction of~~ the unit site boundary limits (Ref. 1) for ~~whole body and thyroid~~ dose ~~rates~~ and a control room dose limit of 5 rem TEDE (Ref. ~~2~~1).

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) and steam generator (SG) power operated relief valves (PORVs). The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generators are assumed to discharge steam and any entrained activity through the MSSVs and SG PORVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary Specific Activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to ~~a small fraction of~~ less than the required site boundary limit (Ref. 1) and a control room dose limit of 5 rem TEDE (Ref. ~~2~~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.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS

A.1 and A.2

Specific activity of the secondary coolant exceeding the allowable value is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR ~~100.11~~50.67.

~~2. 10 CFR 50, Appendix A, GDC 19.~~

~~3~~2. UFSAR, Section 14.2.7.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR ~~100~~50.67 (Ref. 2). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but at least one air lock door must always remain capable of being closed.

BASES

BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Containment Purge Supply and Exhaust System includes a 24 inch purge supply penetration and a 30 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge supply and exhaust penetrations are normally maintained closed. The Containment Purge Supply and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The Containment Purge Supply and Exhaust System is used for this purpose.

The containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 1, involve dropping a single irradiated fuel assembly and handling tool. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 120 hours prior to irradiated fuel movement with containment closure capability, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are ~~a small fraction of~~ less than the ~~guideline~~ values specified in 10 CFR ~~100~~50.67 (Ref. 2).

Containment Penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge supply and exhaust penetrations and the containment personnel air locks. For the OPERABLE containment purge supply and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge Supply and Exhaust System. The OPERABILITY requirements for this LCO ensure that the automatic purge

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations is in its required status. The LCO 3.9.3.c.2 status requirement, which requires penetrations to be capable of being closed by an OPERABLE Containment Purge Supply and Exhaust System, can be verified by ensuring each required valve operator is capable of closing automatically if needed. This Surveillance does not require cycling of the valves since this is performed at the appropriate frequency in accordance with SR 3.9.3.2.

The Surveillance is performed every 7 days during movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess ~~of a small fraction~~ of the ~~guideline~~ values specified in 10 CFR ~~100~~50.67 (Ref. 2).

SR 3.9.3.2

This Surveillance demonstrates that each required containment purge supply and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar valve testing requirements. LCO 3.3.6, "Containment Purge Supply and Exhaust System Isolation Instrumentation," provides additional Surveillance Requirements for the containment purge supply and exhaust valve actuation circuitry. Ensuring these Surveillances are met during movement of irradiated fuel assemblies within containment will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. UFSAR, Section 14.2.1.5.
2. 10 CFR 50.67

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Cavity Water Level

BASES

BACKGROUND	The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in the containment, refueling canal, fuel transfer canal, refueling cavity, and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Ref. 1). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~a small fraction of~~ less than the 10 CFR ~~100~~ 50.67 limits.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment, as described in the UFSAR (Ref. 1). A minimum water level of 23 ft assures an acceptable decontamination factor for iodine. The fuel handling accident analysis inside containment is described in Reference 1. With a minimum water level of 23 ft and a minimum decay time of 120 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and offsite doses are maintained within allowable limits (Ref. 2). Refueling cavity water level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	A minimum refueling cavity water level of 23 ft above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits.
APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies within containment. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.14, "Fuel Storage Pool Water Level."

BASES

ACTIONS

A.1

With a water level of < 23 ft above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within the containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. UFSAR, Section 14.2.1.

2. 10 CFR ~~100.10~~50.67.

Enclosure 8 to AEP-NRC-2014-65

CNP Unit 2 Final TS Pages

(For Information Only)

1.1 Definitions

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel OPERABILITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
CORE OPERATING LIMITS REPORT (COLR)	The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Unit operation within these limits is addressed in individual Specifications.
DOSE EQUIVALENT I-131	DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using thyroid dose conversion factors from Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."
DOSE EQUIVALENT XE-133	DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil" or the

1.1 Definitions

average gamma disintegration energies as provided in ICRP Publication 38, "Radionuclide Transformations" or similar source.

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank,

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE, or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 60 µCi/gm.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	----------NOTE---------- LCO 3.0.4.c is applicable. ----------	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3	6 hours
	AND	
OR	C.2 Be in MODE 5.	36 hours
DOSE EQUIVALENT I-131 > 60 µCi/gm.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 215.1 µCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 µCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.25 gpm in an individual SG, for a total leakage rate of 1 gpm for all SGs.

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Face Velocity (fpm)	Penetration (%)	RH (%)
CREV System	NA	2.5	95
ESF Ventilation System	45.5	5	95
FHAEV System	46.8	5	95

In addition, the carbon samples not obtained from test canisters shall be prepared by either:

1. Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed; or

2. Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches water gauge)	Flowrate (cfm)
CREV System	4	≥ 5,400 and ≤ 6,600
ESF Ventilation System	4	≥ 22,500 and ≤ 27,500
FHAEV System	4	≥ 27,000 and ≤ 33,000

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. The Type A testing Frequency specified in NEI 94-01, Revision 0, Paragraph 9.2.3, as "at least once per 10 years based on acceptable performance history" is modified to be "at least once per 15 years based on acceptable performance history." This change applies only to the interval following the Type A test performed in May 1992.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 12 psig.

c. The maximum allowable containment leakage rate, L_a at P_a, shall be 0.18% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and C tests and $\leq$ 0.75 L_a for Type A tests.

   2. Air lock testing acceptance criterion is overall air lock leakage rate is $\leq$ 0.05 L_a when tested at $\geq P_a$.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.15 Battery Monitoring and Maintenance Program

This program provides for battery restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer including the following:

a. Actions to restore battery cells with float voltage < 2.13 V; and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

Enclosure 9 to AEP-NRC-2014-65

D. C. Cook AST Radiological Analyses Technical Report, prepared by Red Wolf Associates, August 15, 2014

Report Number: RWA-1313-015	**Revision:** 0

D. C. Cook AST Radiological Analyses Technical Report

Preparer: Mark Pope	**Date:**
Reviewer: Myra Manning	**Date:**
Reviewer: Joe Sinodis	**Date:**
Project Manager: Joe Sinodis	**Date:**

Table of Contents

List of Figures

List of Tables

1 Introduction

The current D. C. Cook licensing basis for the radiological consequences analyses of accidents discussed in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) is based on methodologies prescribed in Reg. Guide 1.195 for the offsite doses at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ), and are based upon the Alternative Source Term (AST) methodology from Regulatory Guide 1.183 for the control room doses. These analyses are being updated to fully implement the AST methodology for both the onsite and offsite dose locations.

1.1 Overview

The revised design basis dose analyses are performed using the direction of Reg. Guide 1.183 (Reference [4.1]) with additional guidance provided in Regulatory Issues Summary (RIS) 2006-04 (Reference [4.2]). This single set of analyses is applicable to both Units 1 and 2, and as such, a limiting set of inputs are applied that is bounding for both units. In addition, releases from either unit must consider the dose impact on all receptor locations applicable to both units.

The following UFSAR Chapter 14 accidents are evaluated:

- Loss-of-Coolant Accident (LOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor
- Control Rod Ejection (CRE)
- Waste Gas Decay Tank (WGDT) Rupture
- Volume Control Tank (VCT) Rupture

It is important to note that Reg. Guide 1.183 does not include guidance for the analysis of the WGDT and VCT rupture accidents. These events are evaluated for completeness to apply a consistent source term to all of the radiological consequences presented in the D. C. Cook UFSAR and to assess the control room TEDE doses for these events against the acceptance criteria provided in 10CFR50.67. However, the offsite dose consequences resulting from the WGDT and VCT rupture events will continue to satisfy the acceptance criteria of 10CFR20 as discussed in Item 11 of Reference [4.2].

1.2 Proposed Changes to the Licensing Basis

As part of the full implementation of AST and the update of the D. C. Cook radiological dose analysis, the following changes to the Technical Specifications will be proposed.

- The definition of Dose Equivalent I-131 in Section 1.1 is revised to reference Federal Guidance Report No. 11 as the source of effective dose conversion factors.
- Section 1.1 is revised to replace the definition of $\bar{E}$ Average Disintegration Energy with the definition of Dose Equivalent Xe-133 using dose conversion factors from the effective column of Table III.1 of Federal Guidance Report No. 12.
- The Limiting Condition for Operation related to RCS Activity in Section 3.4.16 is modified to replace the $100/\bar{E}$ gross specific activity criterion with the Dose Equivalent Xe-133 limit.
- The accident induced leakage performance criterion established by the Steam Generator Program in Section 5.5.7.b.2 is revised to be 1 gpm for all steam generators and 0.25 gpm to any one steam generator.
- The maximum allowable leakage rate, L_a at P_a specified by the Containment Leakage Rate Program in Section 5.5.14.c is reduced to 0.18%/day.
- The maximum allowable methyl iodide penetration for the Control Room Emergency Ventilation charcoal adsorber is increased to 2.5% in Section 5.5.9.c.

1.3 Computer Codes

The following computer codes are used in performing the Cook radiological dose analyses:

Computer Code	Version	Reference
RADTRAD	3.10	[4.3] - [4.6]
ARCON96	1997	[4.7]
PAVAN	2.0	[4.8]
JFREQ (METD)	1982	[4.29]
MicroShield	8.03	[4.10]
ORIGEN-ARP	6.1.3	[4.9]
GOTHIC	7.2a	[4.11] - [4.13]

RADTRAD is used to determine the control room and offsite doses for each analyzed event using the source term and *X/Q* inputs. The code considers the release timing, filtration, hold-up, and chemical form of the nuclides released into the environment.

ARCON96 is used to determine the atmospheric dispersion factors (*X/Qs*) at the control room intakes for selected release locations from plant meteorological data.

PAVAN provides atmospheric dispersion factors (*X/Qs*) for various time periods at the EAB and LPZ boundaries using plant meteorological data.

JFREQ is a program in the METD suite of programs that is used to compute the joint frequency distribution of wind speed, wind direction, and atmospheric stability class for use as input to the PAVAN program.

MicroShield is used to determine the direct shine dose to the operators in the control room from the activity on the control room ventilation system filters.

ORIGEN-ARP calculates the fission product isotopic activity of the reactor core used in the development of the core and RCS source terms.

The GOTHIC code is used to simulate the RCS purification system to determine the relative concentrations of nuclides in the reactor coolant, and is also used to calculate the time-dependent RWST temperature due to backleakage from the containment sump.

2 Common Input Parameters, Assumptions, and Methods

2.1 Control Room Dose Calculation Model

During normal operation, 880 cfm of unfiltered air enters the control room through the normal outside air intake. Following a safety injection signal, the control room ventilation system is automatically placed into recirculation after applicable delays for signal processing, emergency power restoration, and damper repositioning. In this configuration, the control room pressurization/cleanup fans circulate 5400 cfm of air through the control room filters, with 880 cfm of this flow supplied by fresh air from the emergency outdoor air intake and the remaining 4520 cfm taken from the control room envelope.

Unfiltered inleakage is assumed to enter the control room at a constant rate of 40 cfm during all modes of system operation. When the control room ventilation system is aligned in the pressurization/cleanup mode, the control room envelope is at a positive pressure with respect to the surrounding areas and leakage is predominantly out of the control room. However, this flow configuration creates a negative pressure in the system ducting downstream of the isolated normal intake dampers. Therefore, the control room unfiltered inleakage is assumed to enter the control room at the location of the normal intakes. Control room occupancy and breathing rates are taken from Position 4.2.6 of Reference [4.1]. The control room model parameters used in the analyses are listed in Table 2.1-1.

Table 2.1-1: Control Room Parameters

Parameter	Value
Control Room Volume	50,616 ft^3
Normal Operation	
Filtered Make-up Flow Rate	0 cfm
Filtered Recirculation Flow Rate	0 cfm
Unfiltered Make-up Flow Rate	880 cfm
Unfiltered Inleakage	40 cfm
Emergency Operation	
Recirculation Mode:	
Filtered Make-up Flow Rate	880 cfm
Filtered Recirculation Flow Rate	4520 cfm
Unfiltered Make-up Flow Rate	0 cfm
Unfiltered Inleakage	40 cfm
Filter Efficiencies	
Elemental	94.05%
Organic	94.05%
Particulate	98.01%
Occupancy	
0 – 24 hrs 1 – 4 days 4 – 30 days	1.0 0.6 0.4
Breathing Rate	3.5×10^{-4} m^3/sec

2.2 Source Terms

2.2.1 Core Source Term

Consistent with the guidance of Position 3.1 of Reference [4.1], the inventory of fission products in the reactor core available for release is based upon the product of the maximum full power operation of the core at the licensed rated thermal power and the ECCS evaluation uncertainty. An ORIGEN-ARP model is created using a single fuel assembly as the mass basis with an assembly burnup selected which conservatively exceeds the expected end-of-cycle core average burnup. Separate ORIGEN-ARP cases are run covering the full range of licensed fuel enrichments, and the maximum activity for each dose-significant isotope is selected from among

these cases. In this manner, the fission product activity of a single assembly is derived which bounds the anticipated core design values for enrichment and burnup. The total core average source term is then obtained by multiplying the single-assembly isotopic activities by the number of fuel assemblies in the core. Key source term inputs are presented in Table 2.2-1.

The list of dose significant isotopes is generally based upon Table 5 of Reference [4.1], which identifies the radionuclide elements that should be considered in the design basis AST dose analysis. This element list is the same as that specified for the revised source term in Table 3.8 of NUREG-1465 (Reference [4.14]). Section 1.4.3.2 of Reference [4.4] documents the creation of a 60-isotope, 9-element list of radionuclides which meets the requirements of NUREG-1465. However, some early AST submittals included a more comprehensive list of isotopes. Specifically, Table 3.1-4 of Reference [4.15] presents a core inventory which contains the standard 60 isotopes from Table 1.4.3.2-2 of Reference [4.4] plus an additional 61 nuclides. This extended list serves as the refined basis for the dose-significant isotopes selected for this analysis; however, the number of isotopes is reduced to a maximum of 100 to accommodate the limitations of RADTRAD 3.10 identified in Section 1.1 of Reference [4.3]. The reduction from 121 to 100 nuclides is based upon the availability of valid dose conversion factors from References [4.16] and [4.17], nuclide half life, and significance of daughter products. The final list of the 100 dose significant isotopes and corresponding core source term activities is presented in Table 2.2-2. This source term is used in combination with dose conversion factors from Federal Guidance Report No. 11 (FGR 11) (Reference [4.16]) and Federal Guidance Report No. 12 (FGR 12) (Reference [4.17]), which is consistent with Position 4.1 of Reference [4.1].

2.2.2 Fuel Handling Accident Source Term

The fuel handling accident source term is developed from the core source term and follows the guidance of Position 3.1 of Reference [4.1], which states that the fission product inventory of each damaged fuel rod for DBA events that do not involve the entire core is determined by dividing the total core inventory by the number of rods in the core. To account for differences in power level across the core, the radial peaking factor is applied to the inventory of the damaged rods. Since the fuel handling accident event involves the failure of all of the rods in a single fuel assembly, the FHA source term is derived by dividing each nuclide activity in the core source term by the number of fuel assemblies and multiplying by the radial peaking factor.

Two additional adjustments are needed to finalize the FHA source term. The gap inventory fractions for I-131 and Kr-85 specified in Table 3 of Reference [4.1] are greater than the respective halogen and noble gas gap fractions. To accommodate these differences, the activities of these two isotopes are increased so that the proper quantities are released when the group gap fractions are applied. Specifically, the activity of I-131 is increased by a factor of 0.08/0.05 = 1.6, and the activity of Kr-85 is increased by 0.1/0.05 = 2.0. Thus, using the core average activities from Table 2.2-2, the radial peaking factor of 1.65 from Table 2.2-1, and 193 fuel assemblies in the core, the fuel handling source term shown in Table 2.2-3 is obtained.

2.2.3 RCS Source Term

The equilibrium nuclide concentration in the RCS is calculated based on the core inventory described in Section 2.2.1. The rate of nuclide release from the core to the reactor coolant for applicable isotopes is calculated from fission product escape rate coefficients and assumes that 1% of fuel rods have defects. With this isotopic production rate in the coolant established, RCS concentrations are calculated with a hydraulic

model of the RCS purification system using GOTHIC. This model accounts for radioactive decay and daughter production, removal of nuclides by the demineralizers, degassing in the volume control tank (VCT), and dilution of the nuclide concentration by normal makeup for RCS boron control. The GOTHIC model is run until the radionuclide concentrations reach equilibrium values. The GOTHIC output provides the relative distribution of isotopic concentrations in the RCS. These values are then manually scaled such that the iodine activities match the Dose Equivalent I-131 limit of 1.0 μCi/gm specified in the Technical Specifications, and the non-iodine isotopes are adjusted separately to meet the gross specific activity limits of 100/Ē. The resulting RCS equilibrium source term is shown in Table 2.2-4. The noble gas concentrations from Table 2.2-4 correspond to a Dose Equivalent Xe-133 of 215.1 μCi/gm based upon the definition of Dose Equivalent Xe-133 provided in TSTF-490 (Reference [4.30]).

2.2.4 Steam Generator Secondary Source Term

The specific iodine activity on the secondary side of the steam generators is limited to 0.1 μCi/gm Dose Equivalent I-131, as shown in Table 2.2-1, which is one-tenth of the RCS iodine activity limit. Therefore, the secondary activities are developed by taking the iodine activities from Table 2.2-4 and reducing them by a factor of ten. The resulting secondary source term is presented in Table 2.2-5.

Table 2.2-1: Source Term Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3480 MWt (3468 MWt plus 0.34% uncertainty)
Core Average Fuel Burnup	43,000 MWD/MTU
Fuel Enrichment	1.5 –5.0 w/o
Uranium Mass per Fuel Assembly	498 kg
Number of Assemblies in the Core	193
Assembly Radial Peaking Factor	1.65
RCS Iodine Specific Activity Limit	1.0 μCi/gm Dose Equivalent I-131
SG Secondary Iodine Specific Activity Limit	0.1 μCi/gm Dose Equivalent I-131
Non-Iodine Gross Specific Activity Limit	100/Ē

Table 2.2-2: Core Source Term

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
Co-58	8.884E+05	Pr-143	1.398E+08
Co-60	6.796E+05	Nd-147	6.178E+07
Kr-85	1.280E+06	Np-239	2.609E+09
Kr-85m	2.364E+07	Pu-238	4.130E+05
Kr-87	4.661E+07	Pu-239	3.727E+04
Kr-88	6.222E+07	Pu-240	6.637E+04
Rb-86	2.272E+05	Pu-241	1.603E+07
Sr-89	8.677E+07	Am-241	1.707E+04
Sr-90	1.002E+07	Cm-242	7.417E+06
Sr-91	1.100E+08	Cm-244	1.838E+06
Sr-92	1.184E+08	Kr-83m	1.119E+07
Y-90	1.038E+07	Br-82	3.972E+05
Y-91	1.142E+08	Br-83	1.106E+07
Y-92	1.197E+08	Br-84	2.009E+07
Y-93	1.358E+08	Rb-89	8.303E+07
Zr-95	1.566E+08	Y-91m	6.384E+07
Zr-97	1.586E+08	Y-95	1.496E+08
Nb-95	1.578E+08	Nb-95m	1.795E+06
Mo-99	1.742E+08	Nb-97	1.596E+08
Tc-99m	1.546E+08	Rh-103m	1.849E+08
Ru-103	1.850E+08	Pd-109	5.749E+07
Ru-105	1.491E+08	Sb-124	1.434E+05
Ru-106	9.480E+07	Sb-125	1.231E+06
Rh-105	1.309E+08	Sb-126	5.873E+04
Sb-127	1.067E+07	Te-125m	2.725E+05
Sb-129	3.215E+07	Te-131	8.174E+07
Te-127	1.054E+07	Te-133	1.024E+08
Te-127m	1.841E+06	Te-133m	8.990E+07
Te-129	3.017E+07	Te-134	1.700E+08
Te-129m	5.821E+06	I-130	3.945E+06
Te-131m	2.119E+07	Xe-131m	1.385E+06
Te-132	1.374E+08	Xe-133m	6.099E+06
I-131	9.814E+07	Xe-135m	4.335E+07
I-132	1.420E+08	Xe-138	1.627E+08

Nuclide	Activity (Curies)	Nuclide	Activity (Curies)
I-133	1.916E+08	Cs-134m	5.865E+06
I-134	2.148E+08	Cs-138	1.776E+08
I-135	1.832E+08	Ba-141	1.522E+08
Xe-133	1.919E+08	La-143	1.419E+08
Xe-135	5.900E+07	Pm-147	1.944E+07
Cs-134	2.523E+07	Pm-148	1.841E+07
Cs-136	6.388E+06	Pm-148m	4.711E+06
Cs-137	1.325E+07	Pm-149	6.245E+07
Ba-139	1.693E+08	Pm-151	2.177E+07
Ba-140	1.639E+08	Sm-153	6.797E+07
La-140	1.700E+08	Eu-154	9.557E+05
La-141	1.533E+08	Eu-155	4.427E+05
La-142	1.475E+08	Eu-156	4.798E+07
Ce-141	1.548E+08	Np-238	5.165E+07
Ce-143	1.430E+08	Pu-243	1.153E+08
Ce-144	1.296E+08	Am-242	1.148E+07

Table 2.2-3: Fuel Handling Accident Source Term

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Co-58	7.595E+03	Pr-143	1.195E+06
Co-60	5.810E+03	Nd-147	5.282E+05
Kr-85	2.189E+04	Np-239	2.230E+07
Kr-85m	2.021E+05	Pu-238	3.531E+03
Kr-87	3.985E+05	Pu-239	3.186E+02
Kr-88	5.319E+05	Pu-240	5.674E+02
Rb-86	1.942E+03	Pu-241	1.370E+05
Sr-89	7.418E+05	Am-241	1.459E+02
Sr-90	8.566E+04	Cm-242	6.341E+04
Sr-91	9.404E+05	Cm-244	1.571E+04
Sr-92	1.012E+06	Kr-83m	9.567E+04
Y-90	8.874E+04	Br-82	3.396E+03
Y-91	9.763E+05	Br-83	9.455E+04
Y-92	1.023E+06	Br-84	1.718E+05

Nuclide	Activity (Ci)	Nuclide	Activity (Ci)
Y-93	1.161E+06	Rb-89	7.098E+05
Zr-95	1.339E+06	Y-91m	5.458E+05
Zr-97	1.356E+06	Y-95	1.279E+06
Nb-95	1.349E+06	Nb-95m	1.535E+04
Mo-99	1.489E+06	Nb-97	1.364E+06
Tc-99m	1.322E+06	Rh-103m	1.581E+06
Ru-103	1.582E+06	Pd-109	4.915E+05
Ru-105	1.275E+06	Sb-124	1.226E+03
Ru-106	8.105E+05	Sb-125	1.052E+04
Rh-105	1.119E+06	Sb-126	5.021E+02
Sb-127	9.122E+04	Te-125m	2.330E+03
Sb-129	2.749E+05	Te-131	6.988E+05
Te-127	9.011E+04	Te-133	8.754E+05
Te-127m	1.574E+04	Te-133m	7.686E+05
Te-129	2.579E+05	Te-134	1.453E+06
Te-129m	4.977E+04	I-130	3.373E+04
Te-131m	1.812E+05	Xe-131m	1.184E+04
Te-132	1.175E+06	Xe-133m	5.214E+04
I-131	1.342E+06	Xe-135m	3.706E+05
I-132	1.214E+06	Xe-138	1.391E+06
I-133	1.638E+06	Cs-134m	5.014E+04
I-134	1.836E+06	Cs-138	1.518E+06
I-135	1.566E+06	Ba-141	1.301E+06
Xe-133	1.641E+06	La-143	1.213E+06
Xe-135	5.044E+05	Pm-147	1.662E+05
Cs-134	2.157E+05	Pm-148	1.574E+05
Cs-136	5.461E+04	Pm-148m	4.028E+04
Cs-137	1.133E+05	Pm-149	5.339E+05
Ba-139	1.447E+06	Pm-151	1.861E+05
Ba-140	1.401E+06	Sm-153	5.811E+05
La-140	1.453E+06	Eu-154	8.170E+03
La-141	1.311E+06	Eu-155	3.785E+03
La-142	1.261E+06	Eu-156	4.102E+05
Ce-141	1.323E+06	Np-238	4.416E+05
Ce-143	1.223E+06	Pu-243	9.857E+05
Ce-144	1.108E+06	Am-242	9.815E+04

Table 2.2-4: RCS Source Term

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
Co-58	0.000E+00	Pr-143	6.713E-03
Co-60	0.000E+00	Nd-147	0.000E+00
Kr-85	2.385E+01	Np-239	0.000E+00
Kr-85m	5.204E-01	Pu-238	0.000E+00
Kr-87	3.299E-01	Pu-239	0.000E+00
Kr-88	9.148E-01	Pu-240	0.000E+00
Rb-86	8.797E-02	Pu-241	0.000E+00
Sr-89	1.335E-03	Am-241	0.000E+00
Sr-90	1.237E-04	Cm-242	0.000E+00
Sr-91	5.681E-04	Cm-244	0.000E+00
Sr-92	2.488E-04	Kr-83m	1.350E-01
Y-90	2.152E-04	Br-82	4.641E-03
Y-91	1.692E-02	Br-83	2.720E-02
Y-92	3.067E-04	Br-84	1.244E-02
Y-93	2.010E-04	Rb-89	2.530E-02
Zr-95	2.409E-02	Y-91m	3.314E-04
Zr-97	3.920E-04	Y-95	0.000E+00
Nb-95	3.478E-02	Nb-95m	1.867E-04
Mo-99	2.070E+00	Nb-97	4.900E-05
Tc-99m	1.980E+00	Rh-103m	1.988E-02
Ru-103	1.991E-02	Pd-109	0.000E+00
Ru-105	9.723E-05	Sb-124	0.000E+00
Ru-106	3.340E-02	Sb-125	0.000E+00
Rh-105	7.689E-04	Sb-126	0.000E+00
Sb-127	0.000E+00	Te-125m	2.449E-02
Sb-129	0.000E+00	Te-131	1.599E-02
Te-127	2.489E-01	Te-133	0.000E+00
Te-127m	2.465E-01	Te-133m	7.643E-03
Te-129	2.281E-01	Te-134	1.092E-02
Te-129m	3.463E-01	I-130	0.000E+00
Te-131m	5.787E-02	Xe-131m	1.600E+00
Te-132	9.639E-01	Xe-133m	1.423E+00
I-131	8.087E-01	Xe-135m	2.138E-01
I-132	6.411E-01	Xe-138	2.292E-01

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
I-133	1.0304E+00	Cs-134m	2.031E-02
I-134	1.231E-01	Cs-138	3.420E-01
I-135	5.365E-01	Ba-141	4.233E-05
Xe-133	1.037E+02	La-143	0.000E+00
Xe-135	3.361E+00	Pm-147	0.000E+00
Cs-134	3.327E+01	Pm-148	0.000E+00
Cs-136	2.188E+00	Pm-148m	0.000E+00
Cs-137	1.852E+01	Pm-149	0.000E+00
Ba-139	1.975E-04	Pm-151	0.000E+00
Ba-140	1.940E-03	Sm-153	0.000E+00
La-140	2.878E-03	Eu-154	0.000E+00
La-141	1.301E-04	Eu-155	0.000E+00
La-142	3.346E-05	Eu-156	0.000E+00
Ce-141	1.445E-02	Np-238	0.000E+00
Ce-143	6.911E-04	Pu-243	0.000E+00
Ce-144	4.229E-02	Am-242	0.000E+00

Table 2.2-5: SG Secondary Source Term

Nuclide	Activity (µCi/g)
I-131	8.087E-02
I-132	6.411E-02
I-133	1.0304E-01
I-134	1.231E-02
I-135	5.365E-02

2.2.5 Fuel Rod Gap Fractions and High Burnup Rods

The fraction of the core fission product inventory located within the fuel rod gap for non-LOCA events is specified in Table 3 of Reference [4.1] and shown below. These values apply to fuel rods that are damaged as a result of the event, and the entire contents of the fuel rod gap are instantaneously released from the fuel. Note that separate gap inventory fractions of 10% for both noble gases and iodines are applied in the Control Rod Ejection event as required by Position 1 of Appendix H to Reference [4.1].

Table 2.2-6: Non-LOCA Fuel Rod Gap Inventory Fraction

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Footnote 11 of Reference [4.1] states that these fuel rod gap inventories are acceptable for fuel rods with a peak burnup of up to 62,000 MWD/MTU provided that the linear heat generation rate does not exceed 6.3 kw/ft for rods with burnups greater than 54 GWD/MTU. For this analysis, a total of 150 rods in two fuel assemblies are assumed to exceed the burnup limits of Footnote 11 to allow for future core design margin. Similar high burnup concerns were raised in alternative source term submittals by Fort Calhoun, Byron/Braidwood, and St. Lucie stations. For these plants, the issue was addressed by doubling the gap fractions for 100% of the rods in the affected assemblies and applying the maximum radial peaking factor. This approach is discussed in Section 2.2.4 of References [4.18], Section 3.3 of Reference [4.19], and Section 2.9.2.2.2.1 of References [4.20] and [4.21]. This same methodology for addressing high burnup fuel is applied to this analysis.

2.3 Atmospheric Dispersion Factors

The dose analysis addresses releases from either unit and must consider the dose impact on all receptor locations applicable to both units. As such, atmospheric dispersion factors are developed for all possible release-receptor pairs, and the values applied in the analysis reflect the most limiting combination without regard to the unit in which the event occurs.

2.3.1 Onsite *X/Q* Determination

New *X/Q* factors for onsite release-receptor combinations are developed using the ARCON96 computer code (NUREG/CR-6331, Reference [4.7]). Reg. Guide 1.194 (Reference [4.26]) contains new guidance that supersedes the NUREG/CR-6331 recommendations for using certain default parameters as input. Therefore, the following changes from the default values are made:

- For surface roughness length, m, a value of 0.2 is used in lieu of the default value of 0.1, and
- For averaging sector width constant, a value of 4.3 is used in lieu of the default value of 4.0.

A number of various release-receptor combinations are considered for the onsite control room atmospheric dispersion factors. These different cases are considered to determine the limiting release-receptor combination for the events.

Figure 2.3-2 provides a sketch of the general layout of the Cook plant that has been annotated to highlight the onsite release and receptor point locations. All releases are taken as ground level releases per the guidance of Position 3.2.1 of Reg. Guide 1.194.

Table 2.3-2 provides information related to the relative elevations of the release-receptor combinations, the straight-line horizontal distance between the release point and the receptor location, and the direction (azimuth) from the receptor location to the release point. Angles are calculated based on trigonometric layout of release and receptor points in relation to the North-South and East-West axes. Plant North is offset 16.39 degrees clockwise from True North (17° 55' 40.5" - 1° 32' 32" = 16.39°) as shown in Figure 2.3-1. This offset is taken into account in development of the release-receptor pair angles.

A building wake term is only applied to releases close to or on the containment surface, where it is clear that the effect of the containment building wake will influence the result. Such releases include the Containment Vent, Containment Surface, Western SG PORVS/MSSVs, West Main Steam Enclosure, and Containment Diffuse Area locations for Units 1 and 2. The building area used for this wake term is 1,690 m^2.

Table 2.3-3 provides the Control Room *X/Q* factors for the release-receptor combinations. These factors are not corrected for occupancy. This table summarizes the *X/Q* factors for the control room intakes used in the various accident scenarios for onsite control room dose consequence analyses. Values are presented for the normal and emergency intakes for each unit.

Table 2.3-6 identifies the Release-Receptor pair and associated Control Room *X/Q* factors from Table 2.3-3 that are used in the event analyses during each of the modes of control room ventilation.

2.3.2 Offsite *X/Q* Determination

For offsite receptor locations, the new atmospheric dispersion (*X/Q*) factors are developed using the PAVAN computer code (NUREG/CR-2858, Reference [4.8]). Table 2.3-4 provides the minimum distance to the EAB and LPZ in each direction for each release location. The offsite maximum *X/Q* factors for the EAB and LPZ are presented in Table 2.3-5. In accordance with Regulatory Position 4 of Reg. Guide 1.145 (Reference [4.27], the maximum value from all downwind sectors for each time period are compared with the 5% overall site *X/Q* values for each boundary, and the larger of the values are used in evaluations.

Offsite release-receptor pair locations are illustrated in Figure 2.3-3. The application of the X/Q values from Table 2.3-5 in the respective event analyses is identified in Table 2.3-6. All of the releases are considered to be ground level releases based upon Position 1.3.2 of Reg. Guide 1.145. As such, the release height in PAVAN is set equal to 10.0 meters as required by Table 3.1 of NUREG/CR-2858. The building area used for the building wake term is the same as for some of the ARCON96 onsite *X/Q* cases, which is 1,690 m^2. The building height entered into PAVAN is the top elevation of the cylindrical portion of the containment building (49.5 m).

2.3.3 Meteorological Data

Meteorological data from a five-year data set (2002, 2004, 2005, 2007, 2010) is used in the development of the new onsite and offsite *X/Q* factors used in the analysis. These years were selected since they were the five

most recent years with full periods of high quality data available. The meteorological data is converted from the raw format into the proper formatting required to create the meteorological data files for use with ARCON96 and PAVAN. The Cook Nuclear Plant records meteorological data from a primary, backup, and shoreline tower. The data from the shoreline tower is considered to most accurately represent the meteorological conditions on-site based on its vicinity to the plant. However, the shoreline tower only records data at a height of 10 meters. As a result, a hybrid meteorological data set is created using the 10 meter shoreline data for the lower level measurements and data from the 60 meter level on the primary tower for the upper level measurements. The stability classes are calculated based on the temperature difference between the 10 m and 60 m levels on the primary tower. The years of data provided are based on the 5 most recent years that have valid data for both the primary and shoreline towers. Five years worth of meteorological data is used which meets the guidance set forth in Regulatory Position 3.1 of Reg. Guide 1.194. The raw data for the years listed above was manipulated within a spreadsheet for appropriate formatting for use with ARCON96 and PAVAN.

The raw data was examined to identify and flag bad or missing data to ensure that the meteorological data used in the atmospheric dispersion factor determination were of high quality. No regulatory guidance is provided in Reg. Guide 1.194 and NUREG/CR-6331 on the valid meteorological data recovery rate required for use in determining onsite *X/Q* values. However, Regulatory Position 5 of Reg. Guide 1.23 (Reference [4.28]) specifies a 90% data recovery threshold for measuring and capturing meteorological data. The 90% data recovery rate applies to the composite of all variables needed to model atmospheric dispersion for each potential release pathway. For the 2002, 2004, 2005, 2007, and 2010 data base, the meteorological data recovery rates are listed in Table 2.3-1.

Table 2.3-1: Meteorological Data Recovery Rate

Parameter	2002	2004	2005	2007	2010
	%	%	%	%	%
Wind Speed 60m	99.9	99.4	99.8	94.3	89.5
Wind Speed 10m Shore	100	96	99.8	99.9	99.2
Wind Direction 60m	99.9	99.4	99.8	99.6	91.8
Wind Direction 10m Shore	100	96	99.8	100	98.6
Delta Temperature 60-10m	99.8	99.3	99.6	99.4	98.1

While the 60 m wind speed for 2010 has a recovery of 89.5%, it is considered acceptable as the cumulative recovery rate for all years of 60 m wind speeds and all parameters for 2010 are well above 90%. With a total of five years worth of data, the contents of the meteorological data file are representative of the long-term meteorological trends at the Cook site.

The raw meteorological data was also processed into annual joint frequency distribution format for 2002, 2004, 2005, 2007, and 2010 for the offsite atmospheric dispersion factor analysis. The joint frequency distribution

file requires the annual meteorological data to be sorted into three classifications that include wind direction, wind speed, and atmospheric stability class. The format for the file conforms to the format provided in Table 3 of Reg. Guide 1.23. The data for all years was sorted into wind speed bins using the guidance provided in RIS 2006-04.

The PAVAN code requires that the maximum speed for each wind speed category be input. The guidance provided in RIS 2006-04 gives a maximum wind speed category of 10 m/s. However, in order to be consistent with Cook meteorological data evaluations, an upper limit of 14 m/s is chosen in order to capture the average of all winds greater than 10 m/s (approximately 12 m/s).

Figure 2.3-1: Site Orientation

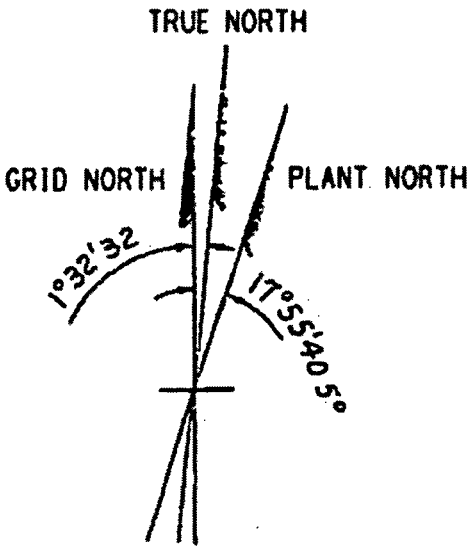
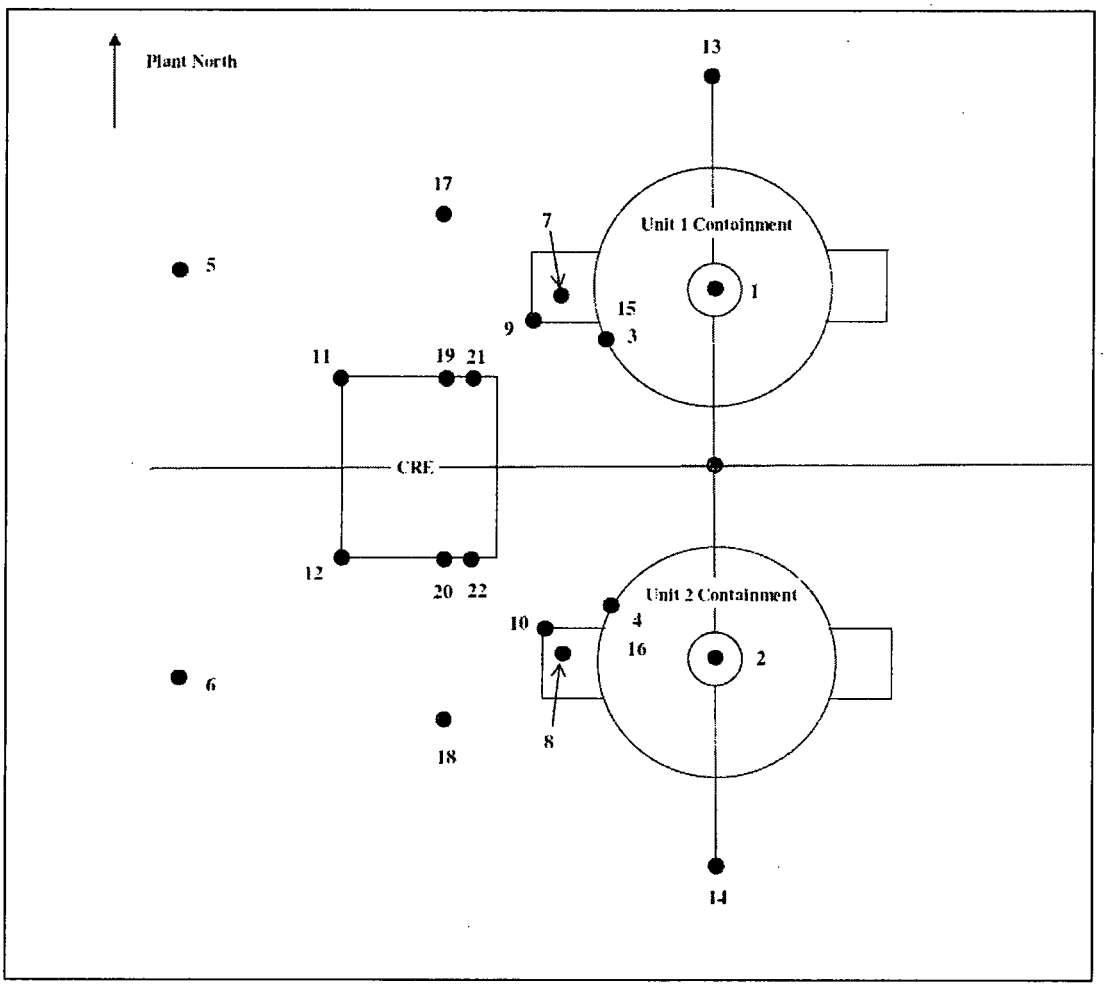


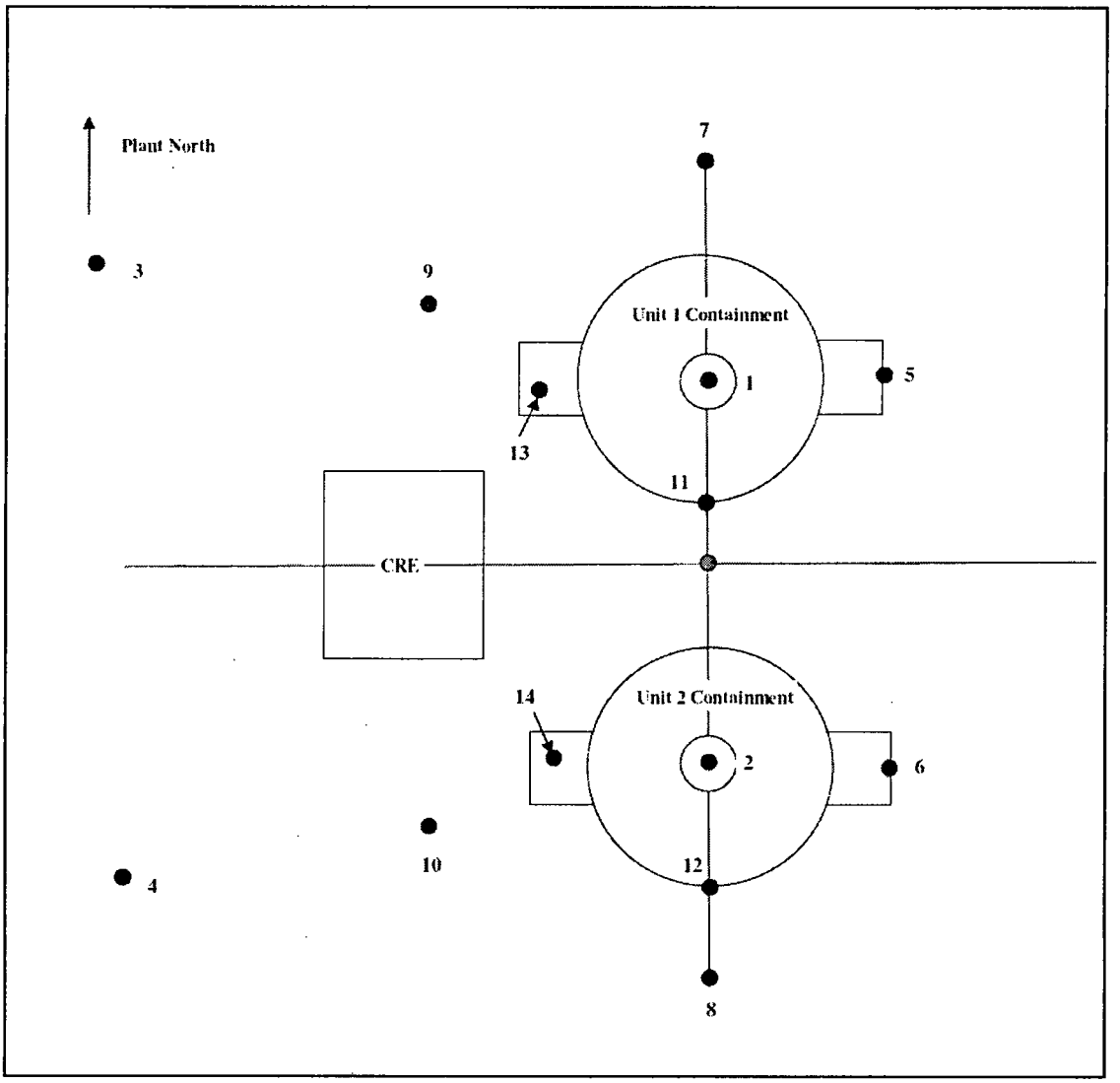
Figure 2.3-2: Onsite Release-Receptor Locations



* Figure is not to scale

1 - Unit 1 Containment Vent
2 - Unit 2 Containment Vent
3 - Closest Point on Unit 1 Containment
4 - Closest Point on Unit 2 Containment
5 - Unit 1 Steam Jet Air Injectors
6 - Unit 2 Steam Jet Air Injections
7 - Unit 1 Western SG PORVs/MSSVs
8 - Unit 2 Western SG PORVs/MSSVs
9 - Unit 1 West Main Steam Enclosure
10 - Unit 2 West Main Steam Enclosure
11 - Unit 1East Turbine Building
12 - Unit 2 East Turbine Building
13 - Unit 1 RWST
14 - Unit 2 RWST
15 - Unit 1 Containment Diffuse Area
16 - Unit 2 Containment Diffuse Area
17 - North Auxiliary Building Supply
18 - South Auxiliary Building Supply
19 - Unit 1 Normal Control Room Intake
20 - Unit 2 Normal Control Room Intake
21 - Unit 1 Emergency Control Room Intake
22 - Unit 2 Emergency Control Room Intake

Figure 2.3-3: Offsite Release-Receptor Locations



* Figure is not to scale

1 - Unit 1 Containment Vent
2 - Unit 2 Containment Vent
3 - Unit 1Turbine Building – NW Corner
4 - Unit 2 Turbine Building – SW Corner
5 - Unit 1 East Main Steam Enclosure
6 - Unit 2 East Main Steam Enclosure
7 - Unit 1 RWST
8 - Unit 2 RWST
9 - North Auxiliary Building Supply
10 - South Auxiliary Building Supply
11 - Unit 1 Containment Surface
12 - Unit 2 Containment Surface
13 - Unit 1 West Main Steam Enclosure
14 - Unit 2 West Main Steam Enclosure

Table 2.3-2: Onsite Release-Receptor Combination Parameters

Release-Receptor Pair	Release Location	Receptor Location (CR Intake)	Release Height (m)	Receptor Height (m)	Distance (m)	Direction (deg)	Building Area (m^2)
A	U1 Plant Vent	U1 Normal	49.5	15.3	44.4	79	1690
B	U1 Plant Vent	U1 Emergency	49.5	14.7	42.0	77	1690
C	U2 Plant Vent	U2 Normal	49.5	14.4	44.4	133	1690
D	U2 Plant Vent	U2 Emergency	49.5	14.7	42.0	135	1690
E	U2 Containment Closest Pt.	U2 Normal	14.4	14.4	25.8	133	1690
F	U2 Containment Closest Pt.	U2 Emergency	14.7	14.7	23.4	136	1690
G	U2 PORV/MSSV	U2 Normal	24.8	14.4	23.2	150	1690
H	U2 PORV/MSSV	U2 Emergency	24.8	14.7	21.4	155	1690
I	U2 Turbine Bldg	U2 Normal	14.4	14.4	9.8	286	0.01
J	U2 Turbine Bldg	U2 Emergency	14.7	14.7	12.5	286	0.01
K	U2 RWST	U2 Normal	12.9	14.4	68.6	161	0.01
L	U2 RWST	U2 Emergency	12.9	14.7	67.1	163	0.01
M	U2 Containment Surface (Diffuse)	U2 Normal	14.4	14.4	25.8	133	1690
N	U2 Containment Surface (Diffuse)	U2 Emergency	14.7	14.7	23.4	136	1690
O	U1 SJAE	U1 Normal	1.5	15.3	87.4	336	0.01
P	South Auxiliary Building Intake	U2 Normal	10.9	14.4	27.2	188	0.01

Table 2.3-3: Onsite Atmospheric Dispersion Factors

Release-Receptor Pair ID	Release Point	Receptor Point	0-2 hour *X/Q*	2-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
A	U1 Plant Vent	U1 Normal	2.03E-03	1.37E-03	4.89E-04	3.84E-04	2.62E-04
B	U1 Plant Vent	U1 Emergency	2.17E-03	1.42E-03	5.18E-04	3.99E-04	2.72E-04
C	U2 Plant Vent	U2 Normal	2.10E-03	1.48E-03	5.52E-04	3.98E-04	3.17E-04
D	U2 Plant Vent	U2 Emergency	2.28E-03	1.59E-03	5.95E-04	4.46E-04	3.49E-04
E	U2 Containment Closest Pt.	U2 Normal	1.02E-02	8.41E-03	2.74E-03	2.66E-03	2.34E-03
F	U2 Containment Closest Pt.	U2 Emergency	1.24E-02	1.02E-02	3.32E-03	3.24E-03	2.84E-03
G	U2 PORV/MSSV	U2 Normal	1.09E-02	8.61E-03	2.87E-03	2.78E-03	2.50E-03
H	U2 PORV/MSSV	U2 Emergency	1.26E-02	9.72E-03	3.26E-03	3.17E-03	2.80E-03
I	U2 Turbine Bldg	U2 Normal	4.57E-02	3.14E-02	1.27E-02	8.30E-03	6.73E-03
J	U2 Turbine Bldg	U2 Emergency	2.91E-02	2.02E-02	8.14E-03	5.34E-03	4.32E-03
K	U2 RWST	U2 Normal	1.74E-03	1.44E-03	5.24E-04	4.42E-04	3.86E-04
L	U2 RWST	U2 Emergency	1.81E-03	1.45E-03	5.43E-04	4.43E-04	3.86E-04
M	U2 Containment Surface (Diffuse)	U2 Normal	2.51E-03	1.94E-03	6.66E-04	6.44E-04	5.61E-04
N	U2 Containment Surface (Diffuse)	U2 Emergency	2.73E-03*	2.05E-03	7.04E-04	6.89E-04	5.95E-04
O	U1 SJAE	U1 Normal	8.50E-04				
P	South Auxiliary Building Intake	U2 Normal	7.91E-03	5.93E-03	2.12E-03	1.50E-03	1.01E-03

* 0-2 hour value is from Unit 1 containment to Unit 1 emergency intake

Table 2.3-4: Offsite Release-Receptor Distances

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Containment Vent	N	542	3,149
	NNE	604	3,149
	NE	702	3,149
	ENE	823	3,158
	E	1,525	3,176
	ESE	1,292	3,201
	SE	982	3,219
	SSE	695	3,219
	S	683	3,219
	SSW	610	3,219
	SW	610	3,219
	WSW	610	3,219
	W	599	3,211
	WNW	575	3,185
	NW	554	3,166
	NNW	542	3,152
Unit 2 Containment Vent	N	609	3,218
	NNE	609	3,218
	NE	769	3,218
	ENE	946	3,218
	E	1,573	3,211
	ESE	1,267	3,185
	SE	790	3,166
	SSE	632	3,152
	S	613	3,149
	SSW	542	3,149
	SW	542	3,149
	WSW	548	3,158
	W	565	3,176
	WNW	589	3,201
	NW	609	3,219
	NNW	609	3,218

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Turbine Building-NW Corner	N	434	3,043
	NNE	442	3,054
	NE	580	3,086
	ENE	770	3,141
	E	1,563	3,207
	ESE	1,413	3,275
	SE	965	3,329
	SSE	786	3,317
	S	666	3,281
	SSW	615	3,235
	SW	566	3,186
	WSW	527	3,142
	W	483	3,103
	WNW	449	3,061
	NW	434	3,044
	NNW	434	3,043
Unit 2 Turbine Building-SW Corner	N	591	3,211
	NNE	644	3,259
	NE	793	3,301
	ENE	990	3,333
	E	1,695	3,305
	ESE	1,048	3,241
	SE	674	3,172
	SSE	563	3,110
	S	519	3,065
	SSW	435	3,045
	SW	434	3,043
	WSW	434	3,043
	W	439	3,051
	WNW	465	3,080
	NW	506	3,128
	NNW	548	3,164

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 East Main Steam Enclosure	N	551	3,146
	NNE	638	3,143
	NE	695	3,143
	ENE	810	3,144
	E	1,523	3,154
	ESE	1,267	3,174
	SE	965	3,192
	SSE	685	3,195
	S	680	3,202
	SSW	617	3,212
	SW	617	3,222
	WSW	623	3,232
	W	626	3,238
	WNW	597	3,209
	NW	569	3,181
	NNW	551	3,161
Unit 2 East Main Steam Enclosure	N	617	3,216
	NNE	684	3,206
	NE	764	3,198
	ENE	933	3,193
	E	1,548	3,184
	ESE	1,242	3,163
	SE	798	3,148
	SSE	620	3,144
	S	610	3,144
	SSW	550	3,144
	SW	550	3,153
	WSW	560	3,172
	W	583	3,198
	WNW	612	3,227
	NW	627	3,236
	NNW	617	3,227

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 RWST	N	508	3,112
	NNE	603	3,112
	NE	667	3,113
	ENE	783	3,126
	E	1,503	3,154
	ESE	1,305	3,187
	SE	1,010	3,223
	SSE	728	3,237
	S	719	3,247
	SSW	641	3,251
	SW	633	3,242
	WSW	620	3,230
	W	592	3,207
	WNW	552	3,168
	NW	524	3,136
	NNW	508	3,118
Unit 2 RWST	N	637	3,247
	NNE	644	3,251
	NE	805	3,242
	ENE	978	3,230
	E	1,587	3,207
	ESE	1,256	3,168
	SE	763	3,136
	SSE	600	3,118
	S	577	3,113
	SSW	508	3,113
	SW	508	3,113
	WSW	515	3,127
	W	539	3,155
	WNW	572	3,188
	NW	613	3,223
	NNW	627	3,237

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
North Auxiliary Building Supply	N	525	3,134
	NNE	526	3,135
	NE	705	3,147
	ENE	816	3,170
	E	1,549	3,200
	ESE	1,327	3,234
	SE	898	3,247
	SSE	716	3,237
	S	694	3,223
	SSW	599	3,209
	SW	587	3,196
	WSW	577	3,187
	W	563	3,175
	WNW	540	3,152
	NW	527	3,137
	NNW	525	3,134
South Auxiliary Building Supply	N	592	3,202
	NNE	606	3,216
	NE	786	3,230
	ENE	908	3,242
	E	1,609	3,246
	ESE	1,299	3,213
	SE	808	3,181
	SSE	642	3,155
	S	611	3,139
	SSW	525	3,134
	SW	525	3,134
	WSW	525	3,134
	W	534	3,144
	WNW	553	3,165
	NW	575	3,184
	NNW	581	3,191

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 Containment Surface	N	523	3,130
	NNE	585	3,130
	NE	683	3,130
	ENE	804	3,139
	E	1,506	3,157
	ESE	1,273	3,182
	SE	963	3,200
	SSE	676	3,200
	S	664	3,200
	SSW	591	3,200
	SW	591	3,200
	WSW	591	3,200
	W	580	3,192
	WNW	556	3,166
	NW	535	3,147
	NNW	523	3,133
Unit 2 Containment Surface	N	590	3,199
	NNE	590	3,199
	NE	750	3,199
	ENE	927	3,199
	E	1,554	3,192
	ESE	1,248	3,166
	SE	771	3,147
	SSE	613	3,133
	S	594	3,130
	SSW	523	3,130
	SW	523	3,130
	WSW	529	3,139
	W	546	3,157
	WNW	570	3,182
	NW	590	3,200
	NNW	590	3,199

Release	Direction	Minimum EAB Distance (m)	Minimum LPZ Distance (m)
Unit 1 West Main Steam Enclosure	N	541	3,150
	NNE	541	3,150
	NE	715	3,158
	ENE	833	3,176
	E	1,549	3,200
	ESE	1,313	3,227
	SE	881	3,231
	SSE	700	3,221
	S	680	3,211
	SSW	592	3,202
	SW	586	3,195
	WSW	583	3,192
	W	577	3,186
	WNW	555	3,167
	NW	544	3,153
	NNW	541	3,150
Unit 2 West Main Steam Enclosure	N	589	3,198
	NNE	596	3,206
	NE	770	3,215
	ENE	891	3,226
	E	1,595	3,235
	ESE	1,294	3,210
	SE	812	3,185
	SSE	649	3,166
	S	623	3,153
	SSW	541	3,150
	SW	541	3,150
	WSW	542	3,151
	W	550	3,160
	WNW	566	3,178
	NW	583	3,192
	NNW	584	3,193

Table 2.3-5: Offsite Atmospheric Dispersion Factors

Release-Receptor Pair ID	Release Point	Receptor Point	0-2 hour *X/Q*	0-8 hour *X/Q*	8-24 hour *X/Q*	1-4 days *X/Q*	4-30 days *X/Q*
Q	U1 Plant Vent	EAB	6.21E-04				
R	U1 Plant Vent	LPZ	7.50E-05	4.17E-05	3.11E-05	1.65E-05	6.60E-06
S	U1 Containment Surface	EAB	6.57E-04				
T	U1 Containment Surface	LPZ	7.54E-05	4.20E-05	3.13E-05	1.66E-05	6.66E-06
U	U1 Turbine Bldg	EAB	9.40E-04				
V	U1 Turbine Bldg	LPZ	7.75E-05	4.34E-05	3.25E-05	1.73E-05	7.04E-06
W	U1 West Main Steam Enclosure	EAB	6.59E-04				
X	U1 West Main Steam Enclosure	LPZ	7.53E-05	4.19E-05	3.13E-05	1.66E-05	6.66E-06
Y	U1 RWST	EAB	6.65E-04				
Z	U1 RWST	LPZ	7.53E-05	4.19E-05	3.13E-05	1.66E-05	6.66E-06
AA	North Auxiliary Building Supply	EAB	6.90E-04				
BB	North Auxiliary Building Supply	LPZ	7.56E-05	4.21E-05	3.15E-05	1.67E-05	6.71E-06

Table 2.3-6: Release-Receptor Pairs Application to the Event Analyses

Event	Normal Intake (1)	Emergency Intake (1)	EAB	LPZ
LOCA:				
- Containment Purge	C	D	Q	R
- Containment Leakage	M	N	S	T
- ECCS Leakage	C	D	Q	R
- RWST Backleakage	K	L	Y	Z
FHA				
- Containment Release	E	F	S	T
- Auxiliary Bldg. Release	A	B	Q	R
MSLB:				
- Break Release	I	J	U	V
- Intact SG Release	G	H	U	V
SGTR	O, G (2)	H	U, W (3)	V, X (3)
Locked Rotor	G	H	W	X
Control Rod Ejection:				
- Containment Leakage	M	N	S	T
- Secondary Side Release	G	H	W	X
WGDT Rupture	P	n/a	AA	BB
VCT Rupture	P	n/a	AA	BB

(1) Control room makeup flow enters through the normal intake prior to realignment of the control room ventilation system and enters through the emergency intake after control room isolation. Unfiltered inleakage enters the control room envelope through the normal intake for the duration of the event.

(2) Prior to reactor trip, the release receptor pair is from the SJAE to the normal intake. The release point changes to the PORV/MSSV immediately after the trip, and the receptor point shifts to the emergency intake following control room isolation.

(3) Prior to reactor trip, the release is from the turbine building. The release point changes to the west main steam enclosure following the trip.

The breathing rates at EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7.

Table 2.3-7: Offsite Breathing Rates

Time (hours)	EAB/LPZ (m^3/sec)
0.0	3.5×10^{-4}
8.0	1.8×10^{-4}
24.0	2.3×10^{-4}
720.0	2.3×10^{-4}

2.4 Direct Shine Dose

In addition to the dose from contamination of the control room atmosphere by intake or infiltration, the total control room dose also requires consideration of direct shine dose contributions from control room filters, from the external radiation plume, and from radioactive material in the containment building. The filter shine dose is calculated by first determining the maximum activity loading on the control room ventilation system filters during the LOCA event. This is done by considering the control room ventilation maximum fan capacity flow rate along with filter efficiencies of 100%. The activities from the recirculation filter edit of the RADTRAD output files are then input into a MicroShield 8.03 model that reflects the geometry of the control room filter housing and the recirculation air handler unit position with respect to the control room. Credit is taken for shielding by structural materials and attenuation in air. An integrated 30-day dose is calculated for control room personnel.

The control room dose due to direct radiation streaming through the equipment hatch following a LOCA conservatively assumes that the control room receptor is positioned directly in front of the equipment hatch such that the exposure from containment is due to a direct line of sight through the entire area of the hatch. The analysis also under-estimates the total thickness of structural walls in the Auxiliary Building. The result is a conservative 30-day dose to an individual in the control room due to direct shine through the containment equipment hatch.

The dose contribution from the external cloud is assessed qualitatively using the guidelines of Reference [4.22] which states that 18 inches of concrete is generally adequate to attenuate the external DBA radiation to negligible levels. A review of site drawings revealed that while the minimum thickness of the control room walls is 18 inches, the control room is surrounded by 3 ft thick Auxiliary Building and 3'-6" Turbine Building walls. Similarly, the thickness of the concrete ceiling immediately above the control room is 18"; however, the thickness of the concrete roof on the elevation above the control room is 2'-6". As such, the shielding against external radiation sources well exceeds the amount identified as adequate in the guidance. Therefore, the additional shine dose contribution to control room personnel from the radioactive plume would be negligible.

The total LOCA shine dose is presented in Table 2.4-1. The filter shine dose following the LOCA event is conservatively applied to all other events.

Table 2.4-1: LOCA Direct Shine Dose

Source	Dose (rem)
Containment	0.246
Control Room Filters	0.139
External Cloud	Negligible
Total	0.385

3 Event Analyses

3.1 Loss of Coolant Accident

Control room and offsite doses are calculated for the LOCA event using the methodology outlined in Appendix A of Reference [4.1]. The dose contribution from the following four different radionuclide release pathways are determined separately and then combined to obtain the total dose for the event:

- Containment Purge
- Containment Leakage
- ESF Leakage to the Auxiliary Building
- ESF Leakage to the Refueling Water Storage Tank (RWST)

For all four cases, the control room ventilation system is automatically placed into the pressurization mode upon receipt of a safety injection signal. Parameters used in modeling the control room ventilation system are shown in Table 2.1-1.

3.1.1 Containment Purge

This containment purge release pathway represents releases through the Containment Purge Supply and Exhaust System prior to containment isolation. Since the purge system is isolated within 15 seconds following the initiation of the event, the release is secured well before the onset of the gap release at 30 seconds as defined in Table 4 of Reference [4.1]. Therefore, only those isotopes initially contained in the RCS fluid (Table 2.2-4) are available for release from containment, which are assumed to be instantaneously and homogeneously mixed throughout the containment atmosphere at the initiation of the event. The containment is modeled as a single compartment without credit for isotope removal by sprays or deposition. Radionuclides are released from containment directly to the environment without mitigation until the containment purge system is isolated. In addition, there is no radionuclide reduction by the containment purge ventilation system, which exhausts to the plant vent.

3.1.2 Containment Leakage

For the containment leakage case, 100% of the core becomes damaged and a phased release of the core fission product inventory (Table 2.2-2) occurs. The fraction of the nuclides in the core that become deposited into containment and the timing of this release are shown in Table 3.1-1 and Table 3.1-2.

Table 3.1-1: Core Inventory Fraction Release into Containment

Group Number	Group Name	Elements	Gap Release Phase	Early In-Vessel Phase
1	Noble Gases	Xe, Kr	0.05	0.95
2	Halogens	I, Br	0.05	0.35
3	Cesium	Cs, Rb	0.05	0.25
4	Tellurium	Te, Sb, Se	0.0	0.05
5	Strontium	Sr	0.0	0.02
6	Barium	Ba	0.0	0.02
7	Ruthenium	Ru, Rh, Pd, Mo, Tc, Co	0.0	0.0025
8	Cerium	Ce, Pu, Np	0.0	0.0005
9	Lanthanum	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0.0	0.0002

Table 3.1-2: LOCA Release Phase Timing

Phase	Onset	Duration
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

The released nuclides are assumed to be instantaneously and homogeneously mixed throughout the containment. The D. C. Cook containment building is modeled as seven separate regions. The entire containment building is served by the safety related Containment Ventilation (CEQ) System. The CEQ System does not have any filtration capability; however, it does provide some additional level of mixing between the regions. Three of the regions are capable of being sprayed by the Containment Spray (CTS) System. Spray induced mixing is also credited between adjacent sprayed and unsprayed regions based on two turnovers of the unsprayed volume per hour. The CTS System is assumed to be secured after 24 hours.

Iodine is released into the containment with a chemical composition of 95% particulate, 4.85% elemental, and 0.15% organic. The containment sump pH is maintained greater than 7.0 following the onset of containment sprays; therefore, re-evolution of particulate iodine into elemental iodine is not considered. Spray removal of elemental and aerosol iodine is credited using the guidance of Reference [4.23]. Removal of elemental iodine in each of the sprayed regions is terminated when the elemental decontamination factor in the region reaches a value of 200. Similarly, the removal rate of the aerosol iodine is reduced by a factor of 10 when the aerosol decontamination factor reaches 50. Natural deposition of only the aerosol iodine is considered in the analysis, and deposition is credited only in unsprayed regions and in sprayed regions after the CTS system has been secured.

Unfiltered leakage from the containment to the environment is assumed to occur uniformly from all seven regions at an initial rate of 0.18%/day. This leakage rate is reduced by 50% to 0.09%/day after 24 hours. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse source from the containment surface.

3.1.3 ESF Leakage into the Auxiliary Building

ESF leakage outside of containment results from operation of ECCS systems which take suction from the containment sump and allow system fluid to be released into the Auxiliary Building through pump seals, valve packing glands, and flanged connections. For this case, a portion of the core source term from Table 2.2-2 is deposited into liquid in the containment sump according to the release fraction and timing shown in Table 3.1-1 and Table 3.1-2. With the exception of noble gases, the fission products released from the fuel are assumed to instantaneously and homogeneously mix in the containment sump water. Leakage from the ECCS systems begins at the onset of switchover to recirculation, and the leak rate into the building is taken as two times the allowable limit established by the leak rate monitoring program. Once the sump fluid exits the system, particulate nuclides are assumed to be retained in the liquid phase, which limits the release to iodine isotopes only. Ten percent of the iodines in the sump fluid are then assumed to become airborne and are released directly to the environment. No credit is taken for holdup or dilution in the Auxiliary Building, or for filtration removal by the ESF Ventilation system. All releases from the Auxiliary Building occur from the plant vent.

3.1.4 ESF Leakage into the RWST

The evaluation of ESF leakage through valves that isolate interfacing systems from the RWST is performed in a separate case. The sump activity for this case is identical to that applied in the case of ESF leakage to the Auxiliary Building described in Section 3.1.3. For this case, flow from the sump into the RWST is assumed to begin immediately upon switchover to recirculation at a rate of 1 gpm (two times 0.5 gpm). The modeling of the release of volatile iodine from the tank to the atmosphere is based upon the guidance of NUREG/CR-5950 (Reference [4.25]). Based upon sump pH controls, the iodine in the sump is considered to be nonvolatile. However, when introduced into the acidic solution of the RWST, a portion of the particulate iodine is converted to elemental iodine. The methodology of NUREG/CR-5950 accounts for two iodine transport/release mechanisms. The first is the fraction of the total iodine in the tank that is released in the form elemental iodine, and the second is the partitioning of elemental iodine between the liquid and vapor phases of the tank.

The fraction of the total iodine that becomes elemental is a function of both the RWST pH and the total iodine concentration in the tank. The analysis determines the time-dependent RWST pH profile as the sump fluid, with a constant pH of 7.0, mixes with the remaining inventory in the tank following switchover to recirculation. Similarly, the total RWST iodine concentration is calculated as sump iodine is transported into the tank from the sump. Calculation of the time-dependent iodine concentration in the RWST liquid for purposes of determining the elemental iodine release fraction conservatively neglects the reduction in concentration due to the release of iodine into the vapor phase. These two parameters combine to produce the elemental iodine fraction, which increases from a value of 0.0 at the beginning of the event to a maximum of 0.1914.

The ratio of the elemental iodine concentrations between the liquid and vapor phases of the tank is determined by a partition coefficient that is a function of the RWST liquid temperature. The analysis calculates a conservatively high RWST temperature profile using GOTHIC by introducing hot sump fluid into the tank without credit for heat removal in the piping between the sump and the tank or heat losses through the tank walls. This results in a time-dependent partition coefficient that decreases from 45.41 at the beginning of the event to 31.92 after 30 days. The elemental iodine fraction and partition coefficient are applied to the leakage flow rate from the sump to the RWST to obtain an adjusted elemental iodine release rate from the tank. A similar approach is taken with the organic iodine, using a release fraction of 0.0015 taken from Position 2 of Appendix A to Reference [4.1] and assuming a conservative partition coefficient of 1.0. The release location for this event is from the RWST vent.

Values of key inputs and assumptions important to the LOCA analysis are provided in Table 3.1-3. The dose consequences for this event are presented in Table 3.1-10.

Table 3.1-3: LOCA Inputs and Assumptions

Input/Assumption	Value
Containment Purge	
Source Term	Initial RCS Activity (Table 2.2-4)
Iodine Chemical Form	95% aerosol, 4.85% elemental, 0.15% organic
Containment Volume	1,066,352 ft^3 (minimum)
Containment Purge Flow Rate	36,300 cfm
Containment Purge Isolation time	15 seconds
Containment Purge Filtration	0%
Removal by Wall Deposition	None
Removal by Sprays	None
Containment Leakage	
Source Term	Core Inventory (Table 2.2-2)
Iodine Chemical Form	95% aerosol, 4.85% elemental, 0.15% organic
Containment Sump pH	>7.0
Compartment Volumes (max)	
Upper Containment (Sprayed)	621,968 ft^3
Lower Containment (Sprayed)	103,770 ft^3
Fan Rooms (Sprayed)	48,913 ft^3
Upper Containment (Unsprayed)	122,600 ft^3
Ice Condenser (Unsprayed)	105,577 ft^3
Lower Containment (Unsprayed)	66,188 ft^3
Dead-End (Unsprayed)	18,663 ft^3

Input/Assumption	Value
Containment Ventilation Start Time	300 seconds
Containment Ventilation Flow Rate Fan Rooms to Lower Containment (Unsprayed) Fan Rooms to Lower Containment (Sprayed) Lower Containment (Unsprayed) to Dead –End Dead-End to Fan Rooms Lower Containment (Unsprayed) to Fan Rooms Lower Containment (Unsprayed) to Ice Condenser Lower Containment (Sprayed) to Ice Condenser Ice Condenser to Upper Containment (Sprayed) Ice Condenser to Upper Containment (Unsprayed) Upper Containment (Sprayed) to Fan Rooms Upper Containment (Unsprayed) to Fan Rooms Lower Containment – Sprayed to/from Unsprayed Upper Containment – Sprayed to/from Unsprayed	 14,580.5 cfm 22,859.5 cfm 90 cfm 90 cfm 1,350 cfm 13,140.5 cfm 22,859.5 cfm 30,072.3 cfm 5,927.7 cfm 30,072.3 cfm 5,927.7 cfm 2206.3 cfm (spray induced circulation) 4086.7 cfm (spray induced circulation)
Sprayed/Unsprayed Volume Induced Mixing Flow Rate	2 Turnovers of Unsprayed Compartment/hour
Containment Spray Start Time	300 seconds
Containment Spray Stop Time	0.319-0.426 hours and after 24 hours
Containment Spray Flow Rate Upper Containment Lower Containment Fan Rooms	 1466 gpm 660 gpm 201 gpm
Containment Spray Drop Fall Height Upper Containment Lower Containment Fan Rooms	 58.6 ft 28.5 ft 20.1 ft
Containment Spray Mean Drop Diameter Upper Containment Lower Containment Fan Rooms	 609 microns 671 microns 671 microns
Elemental Iodine Spray Removal Coefficient	20 hr^{-1}, with a total decontamination factor of 200
Time that Total Elemental DF reaches 200	2 hours

Input/Assumption	Value
Aerosol Spray Removal Coefficient Upper Containment Lower Containment Fan Rooms	 5.06 hr^{-1} 6.65 hr^{-1} 3.03 hr^{-1}
Time that Total Aerosol DF reaches 50	2.32 hours
Organic Iodine Spray Removal	None
Natural Deposition	Elemental, Organic Iodine – None Aerosols - 0.1 hr^{-1} in unsprayed regions only
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.18 %/day 0.09 %/day
Containment Leakage Filtration	0%
ESF Leakage to the Auxiliary Building	
Source Term	Core Inventory (Table 2.2-2)
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Containment Sump Volume	50,955 ft^3
ECCS Recirculation Start Time	1388.4 seconds
ESF Leakage Flow Rate	0.2 gpm (two times the allowable value)
ESF Leakage Flashing Fraction	10%
Auxiliary Building Ventilation Filtration	0%
ESF Leakage to the RWST	
Source Term	Core Inventory (Table 2.2-2)
Containment Sump Volume	50,955 ft^3
ECCS Recirculation Start Time	1388.4 seconds
ESF Leakage Flow Rate	1.0 gpm (two times the allowable value)
Total iodine mass released into the sump	12,035.5 grams
Sump iodine concentration	6.573×10^{-05} g-atom/liter
Sump pH	7.0
Total RWST Volume	420,000 gallons
Initial RWST Liquid Volume	53,637.5 gallons (minimum at time of switchover)
RWST Liquid Iodine Concentration	0.0 - 2.931×10^{-05} g-atom/liter (Table 3.1-4)

Input/Assumption	Value
RWST pH	4.479 – 4.734 (Table 3.1-5)
Elemental Iodine Release Fraction	0.0 - 0.1914 (Table 3.1-6)
Organic Iodine Fraction	0.0015
RWST Liquid Temperature	100 – 118.5 °F (Table 3.1-7)
RWST Liquid/Vapor Iodine Partition Coefficient Elemental Organic	 31.92 - 45.41 (Table 3.1-8) 1.0
Adjusted RWST Iodine Release Rate	Table 3.1-9
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec
Control Room Model Inputs	Table 2.1-1
Control Room Isolation Time	70 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.1-4: RWST Liquid Iodine Concentration

Time (hr)	Iodine Concentration (g-atom/liter)
0.0	0.000E+00
0.386	0.000E+00
0.50	8.381E-09
1.0	4.512E-08
5.0	3.375E-07
10.0	6.994E-07
15.0	1.057E-06
25.0	1.761E-06
50.0	3.456E-06
75.0	5.064E-06
100.0	6.590E-06
125.0	8.042E-06
150.0	9.424E-06
200.0	1.200E-05
250.0	1.435E-05
300.0	1.650E-05
350.0	1.848E-05
400.0	2.031E-05
450.0	2.200E-05
500.0	2.357E-05
550.0	2.503E-05
600.0	2.639E-05
650.0	2.766E-05
700.0	2.886E-05
720.0	2.931E-05

Table 3.1-5: RWST pH

Time (hr)	pH
0.0	4.479
0.386	4.479
0.50	4.479
1.0	4.479
5.0	4.481
10.0	4.484
15.0	4.486
25.0	4.491
50.0	4.502
75.0	4.514
100.0	4.525
125.0	4.535
150.0	4.546
200.0	4.566
250.0	4.586
300.0	4.604
350.0	4.622
400.0	4.639
450.0	4.655
500.0	4.671
550.0	4.686
600.0	4.701
650.0	4.715
700.0	4.729
720.0	4.734

Table 3.1-6: Elemental Iodine Release Fraction

Time (hr)	I_2 Fraction
0.0	0.0000
0.386	0.0000
0.50	0.0002
1.0	0.0011
5.0	0.0080
10.0	0.0161
15.0	0.0238
25.0	0.0379
50.0	0.0673
75.0	0.0902
100.0	0.1085
125.0	0.1234
150.0	0.1356
200.0	0.1541
250.0	0.1670
300.0	0.1761
350.0	0.1824
400.0	0.1866
450.0	0.1893
500.0	0.1908
550.0	0.1914
600.0	0.1914
650.0	0.1907
700.0	0.1896
720.0	0.1890

Table 3.1-7: RWST Temperature Profile

Time (hr)	Temperature °F
0.0	100.0
0.386	100.0
0.50	100.0
1.0	100.1
5.0	100.3
10.0	100.5
15.0	100.8
25.0	101.3
50.0	102.4
75.0	103.5
100.0	104.5
125.0	105.5
150.0	106.4
200.0	108.1
250.0	109.7
300.0	111.1
350.0	112.2
400.0	113.3
450.0	114.3
500.0	115.2
550.0	116.0
600.0	116.8
650.0	117.5
700.0	118.2
720.0	118.5

Table 3.1-8: RWST I_2 Partition Coefficient

Time (hr)	Partition Coefficient
0.0	45.41
0.386	45.41
0.50	45.41
1.0	45.33
5.0	45.15
10.0	44.98
15.0	44.73
25.0	44.30
50.0	43.38
75.0	42.48
100.0	41.68
125.0	40.89
150.0	40.20
200.0	38.92
250.0	37.75
300.0	36.75
350.0	35.99
400.0	35.24
450.0	34.58
500.0	33.99
550.0	33.48
600.0	32.97
650.0	32.53
700.0	32.10
720.0	31.92

Table 3.1-9: Adjusted RWST Iodine Release Rate

Time (hr)	Release Rate (cfm)
0.0	0.0
0.386	6.790E-07
10.0	2.969E-06
25.0	1.300E-05
75.0	3.318E-05
125.0	6.398E-05
200.0	1.118E-04
300.0	1.800E-04
450.0	2.560E-04
600.0	3.167E-04
720.0	3.167E-04

Table 3.1-10: LOCA TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Containment Purge	1.1570E+00	1.3973E-01	7.2674E-01
Containment Leakage	1.9549E+01	3.2836E+00	1.7181E+00
ESF Leakage	2.6402E+00	3.2179E+00	1.6848E+00
RWST Backleakage	2.2441E-02	6.5497E-02	3.9434E-02
Control Room Shine			0.385
Total	**23.37**	**6.71**	**4.56**
Acceptance Limit	25	25	5

3.2 Fuel Handling Accident

The Fuel Handling Accident is evaluated as a drop of a single fuel assembly in which 100% of the rods in the dropped assembly are assumed to fail. The analysis considers both a drop in the containment building without established containment integrity, and a drop in the Auxiliary Building with the Fuel Handling Area Exhaust Ventilation (FHAEV) in service. The source term for this event is described in Section 2.2.2; however, the nuclides available for release are limited to those isotopes present in the fuel rod gap listed in Table 2.2-6. As discussed in Section 2.2.5, the gap inventories shown in Table 2.2-6 are doubled to account for the presence of high burnup fuel rods in the dropped assembly.

The analysis of the Fuel Handling Accident in both locations is modeled as a fuel assembly drop which occurs 120 hours after reactor shutdown and results in the activity which escapes from the pools being released to the environment over a 2-hour period. The water in the pool is credited with a decontamination factor of 285 for elemental iodine and 1.0 for organic iodine as discussed in Item 8 of Reference [4.2]. The pool water is assumed to retain 100% of the alkali metals. It is also assumed that the pool water will have no impact on the noble gases. No credit is taken for mixing or holdup in either the Auxiliary Building or the containment. However, iodine removal by the FHAEV system filters is modeled. The release location for the FHA in containment for the control room dose is assumed to be a point on the external containment surface closest to the control room intakes. For the drop in the Auxiliary Building, the FHAEV system discharges to the plant vent. The control room is assumed to be manually placed into the pressurization mode 20 minutes after the start of the event by the plant operators. Control room ventilation parameters are listed in Table 2.1-1.

Major inputs and assumptions important to this event are provided in Table 3.2-1. The dose consequences are presented in Table 3.2-2 and Table 3.2-3. Note that the direct shine dose contribution from the control room filters conservatively reflects values from the LOCA event.

Table 3.2-1: Fuel Handling Inputs and Assumptions

Input/Assumption	Value
Source Term	FHA Inventory (Table 2.2-3)
Iodine Chemical Form	0% aerosol, 99.85% elemental, 0.15% organic
Number of Fuel Assemblies Damaged	1
Percentage of Fuel Rods Failed	100%
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150
High burnup multiplier applied to gap fractions	2.0
Water Level Above Damaged Fuel	23 feet
Pool Decontamination Factors	Elemental – 285 Organic – 1.0
Delay Before Fuel Movement	120 hours
Containment Release Filtration	0%

Input/Assumption	Value
Fuel Handling Area Exhaust Ventilation Filtration	Aerosol - 98.01% Elemental - 89.1% Organic - 89.1%
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	20 minutes (Manual)
Control Room Occupancy Factor	Table 2.1-1

Table 3.2-2: Fuel Handling Accident – Containment Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Containment Release	3.8807E+00	4.4536E-01	4.3492E+00
Control Room Shine			0.139
Total	**3.89**	**0.45**	**4.49**
Acceptance Limit	6.3	6.3	5

Table 3.2-3: Fuel Handling Accident – Auxiliary Building Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Auxiliary Building Release	5.6965E-01	6.8888E-02	9.6336E-02
Control Room Shine			0.139
Total	**0.57**	**0.07**	**0.24**
Acceptance Limit	6.3	6.3	5

3.3 Main Steam Line Break

This event consists of a break in one main steam line outside of containment in which the faulted steam generator (SG) completely depressurizes and instantly releases the initial contents of the steam generator secondary side to the environment. The plant cooldown continues by dumping steam with the intact steam generators. In addition to the release of nuclides that are initially present in the steam generator secondary side, leakage of primary coolant into the steam generator secondary side occurs at a rate equal to the proposed Tech. Spec. program limit of 0.25 gpm/SG.

This event does not result in fuel damage. Consequently, two iodine spike cases are considered. In the first (pre-accident spike) case, a reactor transient is assumed to occur prior to the MSLB in which the primary coolant iodine concentration has increased to the maximum Tech. Spec. value of 60 μCi/gm. In the second (concurrent spike) case, the iodine release rate into the primary coolant increases to a value that is 500 times greater than the 'normal' release rate that corresponds to the Tech. Spec. specific iodine limit of 1 μCi/gm. This concurrent iodine spike is assumed to have a duration of 8 hours. Inputs used in the development of the concurrent iodine spike appearance rate are shown in Table 3.3-1 and the 8-hour total iodine activity released into the reactor coolant is provided in Table 3.3-2. In both cases, the remaining non-iodine isotopes in the RCS listed in Table 2.2-4 are also available for release.

Leakage from the RCS into all of the steam generators, and steam release from the intact steam generators, continues until the RCS is cooled to 212 °F after 24 hours. During this period, all of the noble gases and all of the nuclides which leak into the faulted steam generator are released directly to the environment without mitigation. Leakage into the intact steam generators mixes with the bulk fluid where a portion of the activity is released based upon the steaming rate and a partition coefficient. A partition coefficient of 100 is applied to the iodine nuclides, and the particulate release is limited by the moisture carryover. It is recognized that early in the transient, the water level in the intact steam generator secondary may be below the top of the tube bundle and the bulk water partitioning may not apply. In this case, a flashing fraction is calculated based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary-to-secondary leakage which flashes to vapor is assumed to be released directly to the environment without mixing. The iodine and particulate partition coefficients are applied to the unflashed portion. The tube bundles in the intact steam generators are assumed to be fully covered after 40 minutes.

The release locations from the faulted steam generator are selected to maximize the control room and offsite doses without regard to the location of the break with respect to the main steam isolation valves (MSIV) and

without credit for MSIV closure. Releases from the intact steam generators occur from the PORVs/MSSVs. The control room is automatically realigned into the pressurization mode (Table 2.1-1) upon receipt of a safety injection signal.

Parameters important to the analysis of the MSLB event are shown in Table 3.3-3, and the analysis results are provided in Table 3.3-4 and Table 3.3-5.

Table 3.3-1: Main Steam Line Break Iodine Appearance Rate Inputs and Assumptions

Input/Assumption	Value
Letdown Flow Rate	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	607,290.6 lbm (maximum)
I-131 Decay Constant	0.000060 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013176 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 3.3-2: Main Steam Line Break 500x Iodine Appearance

Isotope	Appearance Rate (Ci/min)	8-Hour Production (Ci)
I-131	223.45	107,256
I-132	615.35	295,368
I-133	354.95	170,376
I-134	256.40	123,096
I-135	272.95	131,016

Table 3.3-3: Main Steam Line Break Inputs and Assumptions

Input/Assumption	Value
Source Term	Initial RCS Activity (Table 2.2-4)
Maximum Pre-Accident Iodine Spike Concentration	60 μCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	500x Equilibrium (Table 3.3-2)
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
Percentage of Fuel Rods Failed	0%
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Intact Steam Generator Steam Release	0 - 2 hours: 456,000 lbm 2 - 8 hours: 1,186,000 lbm 8 - 24 hours: 1,347,000 lbm
Primary-Secondary Leak Rate	0.25 gpm to each steam generator
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Duration of Intact SG Tube Uncovery After Reactor Trip	40 minutes
Tube Leakage Flashing Fraction During Uncovery	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%
Time to Cool RCS to 212 °F	24 hours
Intact Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Intact Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12

Input/Assumption	Value
Breathing Rates	
EAB/LPZ	The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7.
Control Room	3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	70 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.3-4: Main Steam Line Break Pre-Accident Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.4950E-04	1.5386E-04	1.7202E-03
Pre-Accident Iodine Spike	1.6791E-01	5.9447E-02	4.6052E-01
Initial SG Secondary Iodine	8.7439E-02	7.5715E-03	7.5738E-01
Control Room Shine			0.139
Total	**0.26**	**0.07**	**1.36**
Acceptance Limit	25	25	5

Table 3.3-5: Main Steam Line Break Concurrent Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	4.4950E-04	1.5386E-04	1.7202E-03
Iodine Release	7.3730E-01	1.9621E-01	1.8791E+00
RCS Activity Release	8.4470E-02	2.9910E-02	2.0115E-01
Initial SG Secondary Iodine	8.7439E-02	7.5715E-03	7.5738E-01
Control Room Shine			0.139
Total	**0.91**	**0.24**	**2.98**
Acceptance Limit	2.5	2.5	5

3.4 Steam Generator Tube Rupture

The SGTR event represents an instantaneous rupture of a steam generator tube that releases primary coolant into the lower pressure secondary system. In addition to the break flow rate, primary-to-secondary leakage occurs at a rate equal to the proposed Tech. Spec. program limit of 0.25 gpm/SG to the ruptured and to each of the intact steam generators. All leakage flow into the ruptured steam generator is secured after 30 minutes. Leakage into the intact steam generators continues until the RCS is cooled to 212 °F after 24 hours. A portion of the break and leakage flow to the ruptured steam generator flashes to vapor based upon the thermodynamic conditions in the reactor and secondary coolant. The portion of the primary coolant that does flash in the steam generator secondary is released directly to the environment without mitigation. The unflashed break and leakage flow mixes with the bulk water in the steam generator where the activity is released based upon the steaming rate and a partition coefficient. A steam generator partition coefficient of 100 is applied to the iodine nuclides, and the particulate release is limited by the moisture carryover. Prior to the reactor trip, an additional condenser partition coefficient of 100 is applied to the released activity. This same approach is applied to the primary-to-secondary leakage into the intact steam generators at the start of the event when the SG tube bundles are assumed to become uncovered following the reactor trip. After 40 minutes, flashing of the leakage flow is no longer applicable because the tube bundles have become fully submerged.

This event results in no fuel damage, and as such, two iodine spike cases are considered. In the pre-accident iodine spike case, a reactor transient is assumed to occur prior to the event in which the primary coolant iodine concentration has increased to the maximum Tech. Spec. value of 60 μCi/gm. In the concurrent iodine spike case, the iodine release rate into the primary coolant increases to a value that is 335 times greater than the 'normal' release rate that corresponds to the Tech. Spec. specific iodine limit of 1 μCi/gm. This concurrent iodine spike is assumed to have a duration of 8 hours. Inputs used in the development of the concurrent iodine spike appearance rate are shown in Table 3.4-1, and the 8-hour total iodine activity released into the reactor coolant is provided in Table 3.4-2. In both cases, the remaining non-iodine isotopes in the RCS listed in Table 2.2-4 are also available for release. In addition to the activity transported into the steam generators from the primary coolant, the analysis considers the release of the iodine activity initially present in the steam generator secondary inventory.

Prior to the reactor trip, the activity is assumed to be released from the Steam Jet Air Ejector in the Turbine Building. Following the trip, the release location shifts to the PORVs/MSSVs. The control room is automatically realigned into the pressurization mode (Table 2.1-1) upon receipt of a safety injection signal.

Key inputs and assumptions applied in the analysis of the SGTR event are shown in Table 3.4-3, and the analysis results are provided in Table 3.4-4 and Table 3.4-5.

Table 3.4-1: SGTR Iodine Appearance Rate Inputs and Assumptions

Input/Assumption	Value
Letdown Flow Rate	132 gpm
Identified RCS Leakage	10 gpm
Unidentified RCS Leakage	1 gpm
RCS Mass	607,290.6 lbm (maximum)
I-131 Decay Constant	0.000060 min^{-1}
I-132 Decay Constant	0.005023 min^{-1}
I-133 Decay Constant	0.000555 min^{-1}
I-134 Decay Constant	0.013176 min^{-1}
I-135 Decay Constant	0.001748 min^{-1}

Table 3.4-2: SGTR 335x Iodine Appearance

Isotope	Appearance Rate (Ci/min)	8-Hour Production (Ci)
I-131	149.71	71,861
I-132	412.28	197,894
I-133	237.82	114,154
I-134	171.79	82,459
I-135	182.88	87,782

Table 3.4-3: Steam Generator Tube Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Initial RCS Activity (Table 2.2-4)
Maximum Pre-Accident Iodine Spike Concentration	60 μCi/gm Dose Equivalent I-131
Concurrent Iodine Spike Appearance Rate	335x Equilibrium (Table 3.4-2)
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic

Input/Assumption	Value
Percentage of Fuel Rods Failed	0%
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Intact Steam Generator Steam Release	0 - 30 min. - 198,515 lbm 30 min. - 2 hours: 314,432 lbm 2 - 8 hours: 1,367,475 lbm 8 - 24 hours: 1,347,000 lbm
Ruptured Steam Generator Steam Release	0 - 30 min. – 66,171 lbm
Pre-Trip Total Steam Flow Rate Through Condenser	17,153,800 lbm/hr
Time of Reactor Trip	101 seconds
Primary-Secondary Leak Rate	0.25 gpm to each steam generator
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Ruptured Tube Break Flow	146,704 lbm
Duration of Ruptured Tube Break Flow	30 minutes
Break Flow Flashing Fraction	Pre- Trip - 16% Post-Trip: 0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-30 min: 4%
Duration of Intact SG Tube Uncovery After Reactor Trip	40 minutes
Intact Tube Leakage Flashing Fraction During Uncovery	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%
Time to Cool RCS to 212 °F	24 hours
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Condenser Partition Coefficient	100
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	

Input/Assumption	Value
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	394.74 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.4-4: Steam Generator Tube Rupture Pre-Accident Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	3.1725E-04	1.4862E-04	7.3352E-04
Pre-Accident Iodine Spike	3.7617E+00	4.0743E-01	3.6161E+00
Initial SG Secondary Iodine	2.1118E-03	6.2739E-04	2.6023E-03
Control Room Shine			0.139
Total	**3.77**	**0.41**	**3.76**
Acceptance Limit	25	25	5

Table 3.4-5: Steam Generator Tube Rupture Concurrent Spike TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas	3.1725E-04	1.4862E-04	7.3352E-04
Iodine Release	3.6754E-01	5.1149E-02	2.1296E-01
RCS Activity Release	1.8590E+00	1.9948E-01	1.7749E+00
Initial SG Secondary Iodine	2.1118E-03	6.2739E-04	2.6023E-03
Control Room Shine			0.139
Total	**2.23**	**0.26**	**2.13**
Acceptance Limit	2.5	2.5	5

3.5 Locked Rotor

The Locked Rotor dose analysis is defined by the 11% of the fuel rods which become damaged by the event. Radionuclides released from the fuel are instantaneously and homogeneously distributed throughout the primary coolant. Noble gases are released directly to the environment, and the remaining isotopes are transported to the steam generators at a rate of 1 gpm. The core source term from Table 2.2-2 is applicable to this event, and the fraction of these activities available for release into the coolant are based upon the gap inventory fractions shown in Table 2.2-6 and the assembly radial peaking factor of 1.65. To account for fuel rods contained in two of the fuel assemblies which exceed the burnup limits of Footnote 11 of Reference [4.1], the gap inventory of all of the rods in these two assemblies are assumed to be twice those listed in Table 2.2-6 as discussed in Section 2.2.5. As such, with 193 fuel assemblies in the core, the effective core-wide multiplier on the gap inventory fractions is 1 + (2/193) = 1.0104. Since the fuel failure fraction is applied to the entire core source term, 11% of the rods in both the standard and high burnup assemblies are assumed to fail, and the assembly peaking factor is conservatively applied to all of the failed rods in the core.

During the first 40 minutes of the event, the water level on the secondary side of the steam generators is assumed to be below the top of the tube bundles. During this time, a portion of the primary-to-secondary leakage flashes to vapor based upon the thermodynamic conditions of the reactor and secondary coolant. Nuclides contained in the flashed tube leakage are released to the environment without mitigation. The unflashed leakage mixes with the bulk water in the steam generators and is released as a function of the steaming rate and the partition coefficients. After 40 minutes, all of the leakage is treated as unflashed, which continues until 24 hours when the RCS temperature is cooled to 212 °F.

Since the quantity of the fission products released from the failed fuel dominates the RCS activity during the event, the initial nuclide concentration in the RCS prior to the event is not considered. However, the analysis does include the dose contribution from the release of iodine initially present in the steam generator secondary side. All releases occur from the PORVs/MSSVs, which are located in the Main Steam Enclosures. For this event, the control room ventilation system remains in the normal alignment without filtration or recirculation until manually placed into the pressurization mode after 20 minutes.

Major inputs and assumptions applicable to the Locked Rotor event are listed in Table 3.5-1. The analysis results are presented in Table 3.5-2.

Table 3.5-1: Locked Rotor Inputs and Assumptions

Input/Assumption	Value
Source Term	Core Inventory (Table 2.2-2)
Fuel Rod Gap Fractions	I-131 - 0.08 Kr-85 - 0.10 Other Noble Gases - 0.05 Other Halogens - 0.05 Alkali Metals - 0.12
Percentage of Fuel Rods Failed	11%

Input/Assumption	Value
Fuel Rod Peaking Factor	1.65
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150 rods in two assemblies
High burnup multiplier applied to gap fractions	1.0104
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Primary-Secondary Leak Rate	1.0 gpm to all steam generators
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Secondary Steam Release	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm
Time to Cool RCS to 212 °F	24 hours
Duration of SG Tube Uncovery Following Reactor Trip	40 minutes
Intact Tube Leakage Flashing Fraction During Uncovery	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec

Input/Assumption	Value
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	20 minutes (Manual)
Control Room Occupancy Factor	Table 2.1-1

Table 3.5-2: Locked Rotor TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas Dose	5.9364E-01	1.1156E-01	4.4546E-01
Non-Noble Gas Dose - Iodine	1.0324E+00	3.9490E-01	2.7935E+00
Non-Noble Gas Dose – Alkali Metals	3.5664E-01	8.3519E-02	1.0328E+00
Initial SG Secondary Iodine	1.6639E-03	5.5715E-04	3.1707E-03
Control Room Shine			0.139
Total	**1.99**	**0.60**	**4.42**
Acceptance Limit	2.5	2.5	5

3.6 Control Rod Ejection

The Control Rod Ejection event involves a reactivity insertion that produces a short, rapid core power level increase which results in fuel rod damage and localized melting. For this event, a larger fraction of the core inventory is released from the damaged fuel than that identified in Table 2.2-6. Two separate release pathways are evaluated:, a release from containment and a release from the secondary system. In both cases, 10% of the noble gases and 10% of the iodines in the core (Table 2.2-2) are available for release from the fuel gap of the damaged fuel rods. In addition, 12% of the alkali metals are also assumed to be located in the fuel rod gap.

For releases from containment, 10% of the fuel rods in the core are breached and 0.25% of the fuel experiences melting. The activity in the fuel rod gap of the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere. In addition, the gap inventory fractions are increased by a factor of 1.0104 to account for high burnup fuel as discussed in Sections 2.2.5 and 3.5. Moreover, 100% of the noble gases and 25% of the iodines in the melted fuel are also added to the fission product inventory in containment. No credit is taken for removal by containment sprays or for deposition of elemental iodine on containment surfaces. Natural deposition of aerosols in containment is assumed to occur beginning 24 hours after the start of the event. Activity is released from containment at the proposed Tech. Spec. leak rate. The release from the containment is based upon an atmospheric dispersion factor assuming a diffuse release from the containment surface. For conservatism, the release of iodine initially present in the steam generator secondary side is also considered to address any supplemental cooldown by the steam generators for this event.

For releases from the secondary system, 10% of the fuel rods in the core are breached and 0.25% of the fuel experiences melting. Activity released from the fuel is completely dissolved in the primary coolant and is available for release to the secondary system. The gap activity is increased by a factor of 1.0104 to address fuel rods with burnups that exceed the values of Footnote 11 of Reference [4.1]. In this case, 100% of the noble gases and 50% of the iodines in the melted fuel is also released into the reactor coolant. The noble gases are assumed to be released directly to the environment, and the remaining fission products are transported to the steam generators at the Tech. Spec. steam generator program leakage limit of 1 gpm. At the beginning of the event, a portion of the primary-to-secondary leakage is assumed to flash to vapor based upon the thermodynamic conditions of the reactor and secondary coolant, and the flashed leakage is released directly to the environment without mitigation. The unflashed portion of the tube leakage mixes with the bulk fluid in the steam generator secondary and becomes vapor at a rate that is a function of the steaming rate and the partition coefficients. After 40 minutes, the water level in the steam generator is assumed to fully cover the tube bundles, and all of the primary-to-secondary leakage is treated as unflashed. The leakage continues until steam releases are terminated when the RCS temperature is cooled to 212 °F at 24 hours. With the large amount of fission products introduced into the reactor coolant by failed fuel, the initial activity of the RCS prior to the event is not considered. However, the dose contribution from the iodine activity initially present in the steam generator secondary is included in the analysis. All releases from the secondary system occur from the PORVs/MSSVs.

The control room ventilation system is automatically realigned into the pressurization mode following receipt of a safety injection signal. Key control room ventilation parameters applied in the analysis are shown in Table 2.1-1. Other inputs and assumptions important to the Control Rod Ejection event are provided in Table 3.6-1. The dose consequences for this event are summarized in Table 3.6-2 and Table 3.6-3.

Table 3.6-1: Control Rod Ejection Inputs and Assumptions

Input/Assumption	Value
Source Term	Core Inventory (Table 2.2-2)
Fuel Rod Gap Fractions	Noble Gases - 0.10 Other Halogens - 0.10 Alkali Metals - 0.12
Percentage of Fuel Rods Failed	10%
Percentage of fuel that experience fuel melting	0.25%
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	150 rods in two assemblies
High Burnup multiplier applied to gap fractions	1.0104
Fuel Rod Peaking Factor	1.65
Initial Steam Generator Iodine Source Term	0.1 μCi/gm Dose Equivalent I-131
Iodine Chemical Form - Secondary Release	0% aerosol, 97% elemental, 3% organic
Iodine Chemical Form - Containment Release	95% aerosol, 4.85% elemental, 0.15% organic
Containment Volume	1,066,352 ft^3
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.18 %/day 0.09 %/day
Containment Leakage Filtration	0%
Natural Deposition in Containment	Elemental Iodine – None Aerosols - 0.1 hr^{-1} after 24 hours
Iodine/Particulate Removal by Containment Sprays	None
RCS Mass	466,141.5 lbm (minimum)
Steam Generator Secondary Liquid Mass	97,515.7 lbm/SG (minimum) 161,000 lbm/SG (maximum)
Primary-Secondary Leak Rate	1.0 gpm to all steam generators
Density Used for Leakage Volume-to-Mass Conversion	62.3 lbm/ft^3
Secondary Steam Release	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm
Time to Cool RCS to 212 °F	24 hours
Duration of SG Tube Uncovery Following Reactor Trip	40 minutes

Input/Assumption	Value
Tube Leakage Flashing Fraction During Uncovery	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0
Steam Generator Moisture Carryover Fraction	0.2% (Particulate Partition Coefficient = 500)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	120 seconds (Safety Injection)
Control Room Occupancy Factor	Table 2.1-1

Table 3.6-2: Control Rod Ejection Secondary Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Noble Gas, Cladding Failure	1.0795E+00	2.0283E-01	8.0333E-01
Noble Gas, Fuel Melt	2.6712E-01	5.0193E-02	1.9879E-01
Non-Noble Gas, Cladding Failure, Iodine	1.3048E+00	5.0037E-01	1.8690E+00
Non-Noble Gas, Cladding Failure, Alkali	3.2421E-01	7.5925E-02	2.2304E-01
Non-Noble Gas, Fuel Melt	1.6312E-01	6.2554E-02	2.3365E-01
Initial SG Secondary Iodine	1.6639E-03	5.5715E-04	1.9539E-03
Control Room Shine			0.139
Total	**3.14**	**0.90**	**3.47**
Acceptance Limit	6.3	6.3	5

Table 3.6-3: Control Rod Ejection Containment Release TEDE Dose Results

Release	EAB (rem)	LPZ (rem)	Control Room (rem)
Cladding Failure	4.2230E+00	2.0589E+00	1.5378E+00
Fuel Melt	2.0137E-01	9.1727E-02	6.8727E-02
Initial SG Secondary Iodine	1.6639E-03	5.5715E-04	1.9539E-03
Control Room Shine			0.139
Total	**4.43**	**2.16**	**1.75**
Acceptance Limit	6.3	6.3	5

3.7 Waste Gas Decay Tank Rupture

This event involves a major rupture of one of the Waste Gas Decay Tanks that causes the entire contents of the tank to be released directly to the environment. Reg. Guide 1.183 does not provide any guidance relative to the Waste Gas Decay Tank Rupture event. Guidelines for the WGDT analyses are given in Branch Technical Position 11-5 (BTP 11-5) of the Standard Review Plan (Reference [4.24]), with additional instruction available from Regulatory Issue Summary 2006-04 (Reference [4.2]). The activity in the tank is assumed to be equal to the noble gas content of the reactor coolant system during normal operation. This source term is derived from plant operation with 1% fuel defects which has operated for a sufficient length of time to achieve equilibrium radioactive concentrations in the RCS. The equilibrium RCS concentrations are then adjusted to 100/$\bar{E}$ as discussed in Section 2.2.3. For the WGDT rupture event, the entire noble gas inventory of the RCS is then assumed to be stripped and placed into a single tank. This inventory is conservatively calculated by taking the

noble gas specific activities in the RCS from Table 2.2-4 and multiplying by the maximum RCS mass as shown in Table 3.7-1. The total activity released is determined to be equal to 59,256.4 Curies Dose Equivalent Xe-133, which exceeds the single WGDT licensing limit of 43,800 Curies. The WGDT failure simulates a major tank rupture in which the entire contents of the tank are instantaneously released directly to the environment. No credit is taken for hold-up, dilution, or decay in the Auxiliary Building.

Section B.1.C of Reference [4.24] requires that the release to the environment occur through a pathway not normally used for planned releases and will require a reasonable time to detect and take remedial action to terminate the release. This condition is satisfied in the analysis by assuming that normal Auxiliary Building ventilation is not in service. As such, discharges to the plant vent are not ensured, and gases escaping from the ruptured tank are permitted to be released through building openings with the highest atmospheric dispersion factors. For the offsite dose, the most limiting atmospheric dispersion factor is from the north Auxiliary Building normal ventilation intake, and for the control room dose, the limiting release point is the south normal ventilation intake. The control room ventilation system is assumed to remain in the normal alignment since a safety injection signal, which is required to automatically place the system in the pressurization mode, is not received for this event.

Neither BTP 11-5 nor RIS 2006-04 require dose evaluations at the LPZ or for the control room for a WGDT failure; however, both evaluations are completed in this effort for consistency and completeness. Item 11 of RIS 2006-04 allows the use of existing current licensing basis acceptance criterion of 500 mrem whole body at the EAB when this event is not submitted as part of the AST implementation. Consequently, the results of the WGDT failure are evaluated against the existing 500 mrem acceptance criterion for both the EAB and LPZ. Reg. Guide 1.183 does identify that the criterion for the control room dose is provided in 10 CFR 50.67, which establishes the dose limit for the control room as 5 rem TEDE. This value is applied here.

A summary of the inputs and assumptions to the WGDT rupture analysis are given in Table 3.7-2, and the dose consequences are shown in Table 3.7-3.

Table 3.7-1: WGDT Source Term

Nuclide	RCS Activity (μCi/g)	Total Activity (Ci)
Kr-85m	5.204E-01	1.433E+02
Kr-85	2.385E+01	6.570E+03
Kr-87	3.299E-01	9.087E+01
Kr-88	9.148E-01	2.520E+02
Xe-131m	1.600E+00	4.407E+02
Xe-133m	1.423E+00	3.920E+02
Xe-133	1.037E+02	2.857E+04
Xe-135m	2.138E-01	5.889E+01
Xe-135	3.361E+00	9.258E+02
Xe-138	2.292E-01	6.314E+01

Table 3.7-2: WGDT Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Table 3.7-1. Equal to 59,256.4 D.E. Xe-133
RCS Mass	607,290.6 lbm (275,460,950 gm) (maximum)
Tank Volume	500 ft^3 (arbitrary)
Tank Release Rate	1,000,000 cfm (conservatively high to simulate an instantaneous release)
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ Control Room	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7. 3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	Not Isolated
Control Room Occupancy Factor	Table 2.1-1

Table 3.7-3: WGDT Rupture Dose Results

Release	EAB (rem whole body)	LPZ (rem whole body)	Control Room (rem TEDE)
WGDT Rupture	0.24	0.03	0.09
Acceptance Limit	0.5	0.5	5

3.8 Volume Control Tank Rupture

The analysis of the VCT rupture conservatively assumes a failure of the VCT just prior to venting which releases the accumulated noble gases in the liquid and vapor phases of the tank. In addition, the noble gases within the fluid of the letdown line entering the tank continue to be released for an additional 15 minutes following the tank rupture. The tank and letdown line activities are calculated from RCS equilibrium noble gas concentrations based upon 1% failed fuel with no adjustment for 100/Ē. This conservative source term is consistent with Section B.1.B of Branch Technical Position 11-5. Accumulated gases in the VCT vapor space are calculated using gas stripping fractions, and the entire amount of dissolved gases in the VCT liquid space and the continued letdown flow are available for release after the tank ruptures. The resulting source term is shown in Table 3.8-1.

Further guidance for the analysis of this event is taken from BTP 11-5 since Reg. Guide 1.183 does not address waste system failures. Pathways for the noble gases to escape from the VCT to the environment include those which are not normally used for planned releases. For this event, the release location is the south normal ventilation intake for the control room dose, and is the north normal ventilation intake for the offsite doses based upon the most limiting atmospheric dispersion factors. Releases from the normal supply vents are possible when the Auxiliary Building ventilation system is not in service. This analytical approach is consistent with the guidance of Section B.1.C of BTP 11-5. The analysis assumes that this release is made directly to the environment, without taking credit for hold-up, dilution, or decay in the Auxiliary Building. Similarly, while the control room ventilation system filters would have no impact on this event, the control room ventilation system remains in the normal system alignment described in Section 2.1. Significant inputs and assumptions applied in the analysis of this event are listed in Table 3.8-2.

The acceptance criteria for this event is based upon the guidance of Item 11 of RIS 2006-04, which permits the use of the existing current licensing basis acceptance criterion of 500 mrem whole body at the EAB when AST is not implemented for this event. Without specific guidance for the LPZ, the 500 mrem whole body limit is also applied to the LPZ dose for this event. The acceptance limit of 5 rem TEDE to operators in the control room from 10 CFR 50.67 is applied based upon Section 4.4 of Reg. Guide 1.183. The dose consequences for the VCT rupture event are provided in Table 3.8-3.

Table 3.8-1: VCT Source Term

Nuclide	Equilibrium RCS (μCi/g)	VCT Gas Phase Activity (Ci)	VCT Liquid Phase Activity (Ci)	Letdown Flow Activity (Ci)	VCT Total Release Activity (Ci)
Kr-85m	1.366E+00	1.596E+02	1.021E+01	1.012E+01	1.799E+02
Kr-85	6.261E+01	1.835E+04	4.679E+02	4.639E+02	1.928E+04
Kr-87	8.660E-01	3.955E+01	6.472E+00	6.416E+00	5.244E+01
Kr-88	2.401E+00	2.070E+02	1.794E+01	1.779E+01	2.427E+02
Xe-131m	4.199E+00	8.716E+02	3.138E+01	3.111E+01	9.341E+02
Xe-133m	3.734E+00	7.125E+02	2.791E+01	2.766E+01	7.681E+02
Xe-133	2.723E+02	5.421E+04	2.035E+03	2.017E+03	5.826E+04
Xe-135m	5.612E-01	5.806E+00	4.194E+00	4.158E+00	1.416E+01
Xe-135	8.821E+00	1.200E+03	6.593E+01	6.535E+01	1.331E+03
Xe-138	6.017E-01	5.770E+00	4.497E+00	4.458E+00	1.473E+01

Table 3.8-2: VCT Rupture Inputs and Assumptions

Input/Assumption	Value
Source Term	Table 3.8-1
Tank Volume Liquid Volume	267 ft^3
Tank Volume Vapor Volume	500 ft^3 (arbitrary)
VCT Release Rate	1,000,000 cfm (conservatively high to simulate an instantaneous release)
Letdown Flow Rate	132 gpm
Letdown Isolation Time	15 minutes
Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	 Table 2.3-5 and Table 2.3-6 Table 2.3-3 and Table 2.3-6
Dose Conversion Factors	FGR 11 & FGR 12
Breathing Rates EAB/LPZ	 The breathing rates at the EAB and LPZ are taken from Position 4.1.3 of Reference [4.1] and shown in Table 2.3-7.

Input/Assumption	Value
Control Room	3.5×10^{-4} m^3/sec
Control Room Ventilation System Parameters	Table 2.1-1
Control Room Isolation Time	Not Isolated
Control Room Occupancy Factor	Table 2.1-1

Table 3.8-3: VCT Rupture Dose Results

Release	EAB (rem whole body)	LPZ (rem whole body)	Control Room (rem TEDE)
VCT Rupture	0.36	0.04	0.13
Acceptance Limit	0.5	0.5	5

3.9 Results Summary

The results of the D. C. Cook radiological analyses using the AST methodology are summarized in Table 3.9-1.

Table 3.9-1: Dose Results Summary

Event	EAB Dose (rem TEDE)	LPZ Dose (rem TEDE)	Control Room (rem TEDE)
LOCA	23.37	6.71	4.56
MSLB Pre-accident Iodine Spike	0.26	0.07	1.36
SGTR Pre-accident Iodine Spike	3.77	0.41	3.76
Acceptance Criteria	**25**	**25**	**5**
FHA – Containment Release	3.89	0.45	4.49
FHA – Auxiliary Building Release	0.57	0.07	0.24
Control Rod Ejection – Containment	4.43	2.16	1.75
Control Rod Ejection – Secondary	3.14	0.90	3.47
Acceptance Criteria	**6.3**	**6.3**	**5**
MSLB Concurrent Iodine Spike	0.91	0.24	2.98
SGTR Concurrent Iodine Spike	2.23	0.26	2.13
Locked Rotor	1.99	0.60	4.42
Acceptance Criteria	**2.5**	**2.5**	**5**
WGDT Rupture	0.24(1)	0.03(1)	0.09
VCT Rupture	0.36(1)	0.04(1)	0.13
Acceptance Criteria	**0.5(1)**	**0.5(1)**	**5**

(1) EAB and LPZ results and acceptance criteria for the WGDT and VCT events are whole body doses

4 References

4.1 USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.

4.2 NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms", March 7, 2006.

4.3 ALION-UGM-RADTRRAD-2408-02, Alion RADTRAD 3.10 User's Manual, Revision 0.

4.4 NUREG/CR-6604, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, December 1997.

4.5 NUREG/CR-6604, Supplement 1, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, June 8, 1999.

4.6 NUREG/CR-6604, Supplement 2, RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, August 2002.

4.7 ARCON96 Computer Code "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Rev.1, May 1997, RSICC Computer Code Collection No. CCC-664 and July 1997 errata.

4.8 "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accident Releases of Radioactive Material from Nuclear Power Stations," NUREG/CR-2858, November 1982, (RSICC Computer Code Collection No. CCC-445).

4.9 ORNL/TM-2005/39, Scale: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, Version 6.1, Section D.1 - ORIGEN-ARP: Automatic Rapid Processing for Spent Fuel Depletion, Decay, and Source Term Analysis, June 2011.

4.10 MicroShield User's Manual, Grove Software, Inc., 2009.

4.11 GOTHIC Containment Analysis Package User Manual, Version 7.2a (QA), EPRI, January 2006.

4.12 GOTHIC Containment Analysis Package Technical Manual, Version 7.2a(QA), EPRI, January 2006.

4.13 GOTHIC Containment Analysis Package Qualification Report, Version 7.2a(QA), EPRI, January 2006.

4.14 NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, February 1995.

4.15 Virginia Electric and Power Company letter to the USNRC, Surry Power Station Units 1 & 2, Proposed Technical Specifications and Bases Change – Alternate Source Term Implementation, April 11, 2000 (ML003704270).

4.16 EPA-520/1-88-020, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report No. 11, September 1988.

4.17 EPA-402-R-93-081, External Exposure to Radionuclides in Air, Water, and Soil, Federal Guidance Report No. 12, September 1993.

4.18 USNRC Safety Evaluation Report, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 201 to Facility Operating License No. DPR-40, Omaha Public Power District, Fort Calhoun Station, Unit No. 1, Docket No. 50-285, December 5, 2001 (ML013030027).

4.19 USNRC Safety Evaluation Report, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 147 to Facility Operating License No. NPF-37, Amendment No. 147 to Facility Operating License No. NPF-66, Amendment No. 140 to Facility Operating License No. NPF-72, and Amendment No. 140 to Facility Operating License No. NPF-77, Exelon Generation Company, LLC, Byron Station Unit Nos. 1 and 2, Braidwood Station Unit Nos. 1 and 2, Docket Nos. STN-50-454, STN-50-455, STN-50-456, STN-50-457, September 8, 2006 (ML062340420).

4.20 USNRC Safety Evaluation Report, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 213 to Facility Operating License No. DPR-67, Florida Power and Light Company, St. Lucie Plant, Unit No. 1, Docket No. 50-335, July 9, 2012 (ML12181A019).

4.21 USNRC Safety Evaluation Report, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 163 to Facility Operating License No. NPF-16, Florida Power and Light Company, St. Lucie Plant, Unit No. 2, Docket No. 50-389, September 24, 2012 (ML12235A463).

4.22 NUREG-0800, Standard Review Plan, Section 6.4 – Control Room Habitability System, Revision 3, March 2007.

4.23 NUREG-0800, Standard Review Plan, Section 6.5.2 – Containment Spray as a Fission Product Cleanup System, Revision 4, March 2007.

4.24 NUREG-0800, Standard Review Plan, Branch Technical Position 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure", Revision 3, March 2007.

4.25 NUREG/CR-5950, "Iodine Evolution and pH Control," December 1992.

4.26 USNRC Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, Rev. 0, June 2003.

4.27 NRC Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1, February 1983.

4.28 USNRC Regulatory Guide 1.23, Meteorological Monitoring Programs For Nuclear Power Plants, Rev. 1, March 2007.

4.29 NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," July 1982.

4.30 TSTF-05-16, Technical Specifications Task Force, Subject TSTF-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec", September 13, 2005.

Enclosure 10 to AEP-NRC-2014-65

D. C. Cook AST Regulatory Guide 1.183 Compliance Matrix

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
3.0	**Accident Source Term**		
3.1	**Fission Product Inventory** The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the	Conforms	The core source term is calculated based upon the Unit 2 licensed core power level of 3468 MWt plus the 10CFR 50 Appendix K thermal power uncertainty of 0.34%. Fission product activities are evaluated over a range of fuel enrichments from 1.5% to 5%, which exceeds the licensed limit of 4.955%. The source term is developed by irradiating the fuel to a conservative end-of-cycle core burnup up 43,000 MWD/MTU. This duration is sufficiently long to allow the activity of the radionuclides to reach equilibrium values. The source term is determined using ORIGEN-ARP. With the exception of the Fuel Handling Accident (FHA), the analyses of events which involve fuel damage assume that the entire core is affected with a source

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods. No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.		term based upon full power, core average conditions. The FHA source term is derived from the core source term, the number of damaged fuel rods, and a conservative assembly peaking factor which corresponds to the maximum fuel rod peaking factor permitted by the Technical Specifications The analysis of the FHA considers radioactive decay between the time of core shutdown and the beginning of fuel movement.
3.2	**Release Fractions** The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core.	Conforms	For the LOCA event, the core inventory release fractions for each radionuclide group for both the gap release and early-in-vessel phases from Table 2 of Reg. Guide 1.183 are applied. For the non-LOCA events with fuel damage, the fraction of the core inventory in the gap for the selected isotopes and radionuclide groups are based upon Table

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	Footnote 11: The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load.		3 of Reg. Guide 1.183, with the exception of the Control Rod Ejection (CRE) event. The CRE event is analyzed with 10% of the core inventory of noble gases and 10% of the core inventory of iodines in the fuel rod gap consistent with Position 1 of Appendix H to Reg. Guide 1.183. The gap fractions in Table 3 are applicable to fuel with a maximum rod burnup of 62,000 MWD/MTU and a maximum linear heat generation rate of 6.3 kw/ft for rods exceeding burnups of 54 GWD/MTU per Footnote 11. The dose analyses are performed based upon a small number of rods which exceed the burnup limits of Footnote 11 to provide margin for future core designs. To address these rods, the gap inventory is set to two times the values shown in Table 3 of Reg. Guide 1.183 for all rods in each fuel assembly which contain high burnup rods.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
3.3	**Timing of Release Phases** Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. For non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.	Conforms	For the LOCA event, the onset and release duration of the activity release for each phase conforms to Table 4 of Reg. Guide 1.183. The release occurs in a linear manner over the phase duration. For the non-LOCA events, all releases occur instantaneously at the time of fuel damage.
3.4	**Radionuclide Composition** Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses.	Conforms	The source term in the design basis analysis represents the 100 most dose significant isotopes from the elements listed in Table 5 of Reg. Guide 1.183.
3.5	**Chemical Form** Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from	Conforms	The chemical composition of the iodine released from the RCS to containment in the LOCA event is 95% aerosol, 4.85% elemental, and 0.15% organic. All non iodine and non-noble gas fission products are assumed to be in particulate form. The chemical composition of iodines in

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.		the non-LOCA events are based upon the guidance in the respective appendices of Reg. Guide 1.183.
3.6	**Fuel Damage in Non-LOCA DBAs** The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	The amount of fuel damage in the Locked Rotor event is based upon the fraction of the core which experiences DNB as reported in the Updated Final Safety Analysis Report (UFSAR). The fraction of the fuel rods assumed to melt in the CRE event is conservatively based upon the portion of the fuel centerline that is calculated to exceed the melting temperature as documented in the UFSAR.
4.0	**Dose Calculation Methodology**		
4.1	**Offsite Dose Consequences**		
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides, that are significant with regard to dose consequences and the released radioactivity.	Conforms	The dose calculations determine the TEDE (sum of CEDE and DDE) and considers all dose significant isotopes from elements listed in Table 5 of Reg. Guide 1.183, including progeny from decay of parent radionuclides.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Dose Conversion Factors for inhalation in this analysis are taken from Table 2.1 of Federal Guidance Report 11.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	Offsite breathing rates used in the analysis are consistent with the values specified in Section 4.1.3 of Reg. Guide 1.183.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" , provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Dose Conversion Factors for air submergence are taken from the Table III.1 of Federal Guidance Report 12.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of	Conforms	The TEDE was determined for the most limiting person at the EAB. The maximum two-hour TEDE was determined by calculating the postulated dose for a series of small time increments

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release.		and performing a 'sliding' sum over the increments for successive two-hour periods.
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	The TEDE is determined for the most limiting person at the LPZ.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No correction is made for deposition of the effluent plume by deposition on the ground.
4.2	**Control Room Dose Consequences**		
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The control room TEDE analysis considers all significant sources of radiation that will cause exposure to personnel, including intake from the radioactive plume, shine from the external plume, direct shine from containment, and shine from control room ventilation system filters.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The control room doses are determined using the same source term, transport, and release assumptions used in the calculation of the EAB and LPZ TEDE values.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	The models used to transport radioactive material into and through the control room and the shielding models used to determine radiation dose rates from external sources are developed to provide suitably conservative estimates of the exposure to control room personnel.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", for guidance.	Conforms	Credit is taken for automatic realignment of the control room ventilation system into the pressurization mode upon receipt of a safety injection signal, and filtration of the control room atmosphere by the recirculation filters.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Credit was not taken for the use of personal protective equipment or prophylactic drugs.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40%	Conforms	Control room occupancy and breathing rates used in the analysis are consistent with the values specified in Section 4.2.6

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5 x 10^{-4} cubic meters per second.		of Reg. Guide 1.183.
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	Conforms	Control room doses are calculated using dose conversion factors identified in Position 4.1 above. Equation 1 from Reg. Guide 1.183 is used for finite cloud correction when calculating the DDE immersion doses due to airborne activity inside the control room.
4.3	**Other Dose Consequences** The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Exception	The current TID-14844 accident source term will remain the licensing basis for equipment qualification and NUREG-0737 evaluations other than control room habitability.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
4.4	**Acceptance Criteria** The radiological criteria for the EAB, the outer boundary of the LPZ, and for the control room are in 10 CFR 50.67. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. The control room criterion applies to all accidents. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 6.	Conforms	The EAB and LPZ acceptance criteria from Table 6 of Reg. Guide 1.183 are applied. The control room acceptance criteria of 5 rem TEDE is taken from 10 CFR 50.67(b)(2)(iii).
5.0	**Analysis Assumptions and Methodology**		
5.1	**General Considerations**		
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident	Conforms	The analyses have been prepared, reviewed, and will be maintained in accordance with quality assurance programs that comply with 10 CFR 50, Appendix B.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based upon data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence -- the proposed deviation may not be conservative for other accident sequences.		
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Only safety-related Engineered Safety Features are credited in the analysis with an assumed single active failure that results in the greatest impact on the radiological consequences. Assumptions regarding the occurrence and timing of a loss of offsite power are made with the objective of maximizing the impact on dose.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.	Conforms	Numerical values are selected and biased for each application in a conservative direction with the objective of maximizing the dose consequences. Parameters which are controlled by Technical Specifications are used as direct inputs in the analyses.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
5.1.4	The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. In order to issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.	Conforms	The analysis assumptions and methods are compatible with the AST and TEDE criteria.
5.2	**Accident-Specific Assumptions** The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are required by 10 CFR 50.67. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only. The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST. The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives.	Conforms	The LOCA, FHA, Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Locked Rotor, and CRE events are all analyzed using the guidance provided in the appendices to Reg. Guide 1.183 and evaluated against the AST acceptance criteria specified in Position 4.4. The Waste Gas Decay Tank (WGDT) rupture and Volume Control Tank (VCT) rupture events are analyzed and the results compared to the 500 mrem EAB limit provided in Item 11 of NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternative Source Terms."

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
5.3	**Meteorological Assumptions** Atmospheric dispersion values (*X/Q*) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19"(Refs. 6, 7, 22, and 28). References 22 and 28 should be used if the FSAR *X/Q* values are to be revised or if values are to be determined for new release points or receptor distances. Fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the worst 2-hour exposure period. The NRC computer code PAVAN (Ref. 29) implements Regulatory Guide 1.145 (Ref. 28) and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 (Ref. 26) is generally acceptable to the NRC staff for use in determining control room *X/Q* values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/ Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 30). All changes in *X/Q* analysis methodology should be reviewed by the NRC staff.	Conforms	All *X/Q* values have been recalculated for the AST analysis. Offsite *X/Q*s are calculated using PAVAN based upon the guidance provided in Reg. Guide 1.145. New control room atmospheric dispersion factors are developed based upon Reg. Guide 1.194 and calculated using ARCON96. The meteorological data used as input to the *X/Q* development conforms to Reg. Guide 1.23 and follows the guidance of Item 4 of NRC Regulatory Issue Summary 2006-04.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
6.0	**Assumptions for Evaluating the Radiation Doses for Equipment Qualification** The assumptions in Appendix I to this guide are acceptable to the NRC staff for performing radiological assessments associated with equipment qualification. The assumptions in Appendix I will supersede Regulatory Positions 2.c(1) and 2.c(2) and Appendix D of Revision 1 of Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 11), for operating reactors that have amended their licensing basis to use an alternative source term. Except as stated in Appendix I, all other assumptions, methods, and provisions of Revision 1 of Regulatory Guide 1.89 remain effective. The NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or the TID14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs TID14844) on EQ doses pending the outcome of the evaluation of the generic issue.	Conforms	An AST assessment was not performed for equipment qualification. The TID-14844 assumptions will continue to be used as the radiation dose basis for equipment qualification, radiation zone maps, and shielding calculations.
Appendix A	**Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of–Coolant Accident**		
Appendix A	**Source Term Assumptions**		
Appendix A 1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	Assumptions regarding core inventory and the release of radionuclides from the fuel are consistent with Position 3 of Reg. Guide 1.183.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix A 2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The sump pH is controlled at values of 7 or greater. The chemical form of radioiodine released to the containment is 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodine. With the exception of elemental and organic iodine and noble gases, all fission products are assumed to be in particulate form.
Appendix A 3.0	**Assumptions on Transport in Primary Containment**		
Appendix A 3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment as it is released. There is no adjustment to the distribution due to limited ventilation exchange.
Appendix A 3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2 of the Standard Review Plan (SRP), NUREG-0800 and in NUREG/CR-6189. The prior practice of deterministically assuming that a 50% plateout of iodine is released from the fuel is no longer acceptable to the NRC staff as it is inconsistent with the characteristics of the revised source terms.	Conforms	Natural deposition of elemental iodine is conservatively ignored. Plateout of aerosols is credited in unsprayed regions of containment and in sprayed regions following termination of containment sprays with a removal coefficient of 0.1/hr.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix A 3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966. This simplified model is incorporated into the analysis code RADTRAD. The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays.	Conforms	Elemental and aerosol removal coefficients are calculated for the sprayed regions of the containment using the guidelines of Chapter 6.5.2 of the Standard Review Plan. The elemental iodine removal coefficients are limited to a maximum value of 20/hr, and are set to zero when the elemental iodine decontamination factor (DF) reaches a value of 200. The aerosol removal coefficients are reduced by a factor of 10 when the aerosol DF reaches 50. The mixing rate between the sprayed and unsprayed compartments is based upon two turnovers of the unsprayed volume per hour and is modeled only during the times that the containment spray system is in service.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix A 3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Conforms	Modeling of air mixing by the safety related containment ventilation (CEQ) system. However, this system does not contain filters and no radionuclide removal is credited.
Appendix A 3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool. Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	N/A	Regulatory Position 3.5 applies to BWRs only.
Appendix A 3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP.	N/A	No credit is taken for retention in containment by engineering safety features.
Appendix A 3.7	The primary containment should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate.	Conforms	The initial containment leak rate applied in the analysis is equal to the proposed Tech. Spec. value of 0.18% per day. After 24 hours, the leak rate is reduced to 0.09% per day.
Appendix A 3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity.	Conforms	The dose contribution from containment purge is considered in the analysis. 100% of the radionuclide activity in the reactor coolant system (RCS) is assumed to be instantly and homogeneously released to containment at the beginning of the event. The RCS source term is based upon Tech.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable		Spec. limits on specific activity in the coolant without iodine spiking. The containment purge system is automatically isolated before the onset of the gap release phase.
Appendix A 4.0	**Assumptions on Dual Containments**	N/A	Regulatory Positions 4.1 through 4.6 pertain to facilities with dual containments and are not applicable to D. C. Cook.
Appendix A 5.0	**Assumptions on ESF System Leakage**		
Appendix A 5.1	With the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used.	Conforms	With the exception of noble gases, all of the fission products released from the fuel as defined by Table 2 of Reg. Guide 1.183 are instantly and homogeneously mixed in the containment sump water. The release rate from the fuel is assumed to occur consistent with the phase timing listed in Table 4 of the Reg. Guide.
Appendix A 5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737, would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through	Conforms	The D. C. Cook Tech. Specs do not provide a specific limit for operational leakage from ECCS systems. However, administrative limits ensure that operational leakage is adequately controlled. In the analysis, leakage from ECCS systems is taken as two times the

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
	valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.		programmatic limit of 0.1 gpm for leakage of sump water outside of containment into the Auxiliary Building. In addition, two times the allowable leak rate of 0.5 gpm past valves that isolate return flow to the Refueling Water Storage Tank (RWST) is evaluated separately. The leakage is assumed to start at the earliest time that recirculation occurs in the ECCS systems and continues for the 30-day duration of the event.
Appendix A 5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in the recirculating liquid is assumed to be retained in the liquid phase.
Appendix A 5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment. $$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	N/A	The temperature of the ESF leakage remains below 212°F.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix A 5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	A flashing fraction of 10% is applied in the analysis for ESF leakage into the Auxiliary Building. The amount of the radioiodine in the sump fluid available for release from the RWST as elemental iodine is calculated from the total iodine concentration and pH history of the tank using the guidance of NUREG/CR-5950 as discussed in Item 5 of Regulatory Issue Summary 2006-04.
Appendix A 5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 and Generic Letter 99-02.	Conforms	The radioiodine that is released to the environment through the Auxiliary Building is assumed to be 97% elemental and 3% organic. No credit is taken for holdup or filtration within the Auxiliary Building. 100% of the iodine that is converted to volatile iodine in the RWST using the guidance of NURG/CR-5950 is assumed to elemental. The 0.15% organic iodide in the sump fluid specified in Position 2 of Appendix A is also assumed to be available for release. Holdup and dilution within the tank is credited as allowed by Position 5.6.
Appendix A 6.0	**Assumptions on Main Steam Isolation Valve Leakage in BWRs**	N/A	Regulatory Positions 6.1 through 6.5 pertain to BWRs are not applicable to D. C. Cook.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix A 7.0	**Assumption on Containment Purging** The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated.	Conforms	Containment purging for combustible gas control is not credited in any design basis analysis and the radiological consequences of post-isolation purging are not considered.
Appendix B	**Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident**		
Appendix B 1	**Source Term**		
Appendix B 1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	100% of the fuel rods in the dropped assembly are conservatively assumed to fail. An assessment was made of the failure of rods in multiple assemblies being moved into the storage cask. Damage to 100% of the rods in a freshly discharged assembly was determined to be more limiting.
Appendix B 1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	The fission products released from the damaged fuel rods are based upon Position 3.2 of Reg. Guide 1.183 and the number of fuel rods breached. Radionuclides considered include noble gases, halogens, and alkali metal. It should be noted that per Position 3 of Appendix B to the Reg. Guide, particulates are retained by the water in

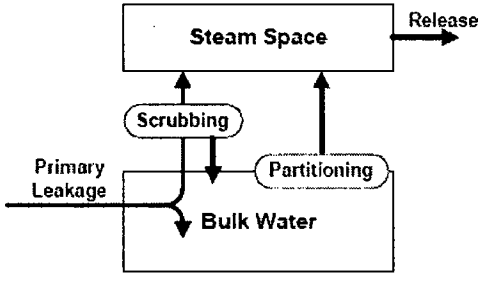
RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
			the pool. Therefore, the cesiums and rubidiums released from the fuel have no further impact on the analysis.
Appendix B 1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously.	Conforms	The chemical form of the iodine released from the fuel is 95% cesium iodide, 4.85% elemental, and 0.15% organic. The particulate iodine is assumed to instantaneously re-evolve as elemental iodine in the pool, resulting a pool iodine composition of 99.85% elemental and 0.15% organic.
Appendix B 2	**Water Depth** If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species.	Conforms	The analysis considers iodine decontamination by at least 23 feet of water above the damaged fuel. Using supplemental guidance from Item 8 of Regulatory Issue Summary 2006-04, elemental and organic decontamination factors of 285 and 1, respectively, are applied to obtain an overall effective pool decontamination factor of 200.
Appendix B 3	**Noble Gases** The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	Noble gases are released immediately to the atmosphere without mitigation by the pool. All particulates which escape from

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
			the damaged fuel are retained by the water in the pool.
Appendix B 4	**Fuel Handling Accidents Within the Fuel Building**		
Appendix B 4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	The D. C. Cook fuel handling analysis considers a release in the Fuel Handling Area of the Auxiliary Building. The radioactive material that escapes from the fuel pool to the Aux. Building is assumed to be released to the environment over a 2-hour time period.
Appendix B 4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	A reduction in the amount of radioactive material released from the Auxiliary Building is credited by use of the Fuel Handling Area Exhaust Ventilation (FHAEV) system. This system meets the requirements of Reg. Guide 1.52 and is required to be in service prior to the movement of irradiated fuel in the building.
Appendix B 4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building.	Conforms	There is no credit taken for mixing or dilution in the Auxiliary Building.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix B 5	**Fuel Handling Accidents Within Containment**		
Appendix B 5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Conforms	Containment isolation is not required during fuel movement and the fuel handling accident within containment is considered.
Appendix B 5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Conforms	No automatic closure of containment is credited in the analysis.
Appendix B 5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	The containment is permitted to be open during fuel handling operations. The radioactive material that escapes from the cavity to containment is assumed to be released to the environment over a 2-hour time period.
Appendix B 5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02. Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Conforms	There is no credit taken for reduction in the amount of radioactive material released from containment by filtration systems.
Appendix B 5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis.	Conforms	There is no credit taken for mixing or dilution in the containment.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix E	**Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident**		
Appendix E	**Source Term**		
Appendix E 1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.	Conforms	No fuel damage is postulated to occur during the MSLB event at D. C. Cook.
Appendix E 2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.	Conforms	There is no fuel damage for the D. C. Cook MSLB event. Two iodine spike cases are evaluated.
Appendix E 2.1	A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value permitted by the technical specifications (i.e., a pre-accident iodine spike case).	Conforms	A pre-iodine spike case is considered in which the RCS activity increases to the maximum value permitted by the Tech. Specs., which is 60 μCi/gm DE I-131.
Appendix E 2.2	The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.	Conforms	A concurrent iodine spike case is considered in which the reactor coolant iodine production rate increases to 500 times the iodine appearance rate that produces the maximum equilibrium value allowed by Tech. Specs. The concurrent spike duration is assumed to continue for 8 hours.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix E 3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the reactor coolant.
Appendix E 4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	The chemical form of the iodine released from the steam generators to the environment is 97% elemental and 3% organic.
Appendix E 5	**Transport**		
Appendix E 5.1	For facilities that have not implemented alternative repair criteria, the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Primary-to-secondary leakage is 0.25 gpm per SG and 1 gpm to all steam generators. These values are consistent with the proposed change to the leakage performance criteria of the steam generator program described in the Tech. Specs.
Appendix E 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used to convert volumetric leak rates to mass leak rates corresponds to a temperature of 70°F and a pressure of 14.7 psia as directed by plant leak rate monitoring procedures.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix E 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Primary-to-secondary leakage and releases from intact steam generators is assumed to continue for 24 hours. This value conservatively bounds the time required to cool the RCS to 212°F and to place shutdown cooling in service.
Appendix E 5.4	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gases are released to the environment without reduction or mitigation.
Appendix E 5.5	The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below. **Figure E-1** **Transport Model**  	Conforms	The transport model described in Section 5.5 of Reg. Guide 1.1983 and shown in Figure E-1 is used for iodine and particulate release from the steam generators.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix E 5.5.1	A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. • During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation. • With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence	Conforms	For the faulted steam generator, which dries out completely, all of the primary-to-secondary leakage flashes to vapor and is released to the environment without mitigation. During periods of tube uncovery in the intact steam generators, a portion of the leakage flashes to vapor based upon the thermodynamic conditions of in the primary and secondary coolant. Once the SG tubes are fully covered, the leakage mixes with the water in the SG secondary without flashing.
Appendix E 5.5.2	The leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident", during periods of total submergence of the tubes.	Conforms	The leakage which flashes to vapor is conservatively assumed to rise through the bulk water of the steam generator and enters the steam space without any mitigation by scrubbing.
Appendix E 5.5.3	The leakage that does not immediately flash is assumed to mix with the bulk water.	Conforms	All leakage that does not immediately flash to vapor mixes with the bulk water in the steam generator secondary side.
Appendix E 5.5.4	The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient for iodine of 100 may be assumed. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.	Conforms	The radioactivity in the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is applied to the iodine nuclides, and the partition coefficient applied to particulate isotopes originating from the equilibrium RCS gross activity is

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
			set to the steam generator moisture carryover fraction.
Appendix E 5.6	Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip. The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.	Conforms	Tube bundle uncovery is postulated to occur in the intact steam generators following a reactor trip. During the time of uncovery, flashing of the tube leakage is modeled in accordance with the guidelines of Position 5.5.1. The tube bundles are assumed to be fully covered by operation of the Auxiliary Feedwater system within 40 minutes based upon plant operating experience.
Appendix F	**Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident**		
Appendix F	**Source Term**		
Appendix F 1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	No fuel damage is postulated to occur during the SGTR event at D. C. Cook.
Appendix F 2	If no or minimal fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specification. Two cases of iodine spiking should be assumed.	Conforms	There is no fuel damage for the D. C. Cook SGTR event. Two iodine spike cases are evaluated.
Appendix F 2.1	A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value permitted by the technical specifications (i.e., a pre-accident iodine spike case).	Conforms	A pre-iodine spike case is considered in which the RCS activity increases to the maximum value permitted by the Tech. Specs., which is 60 μCi/gm DE I-131.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix F 2.2	The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours.	Conforms	A concurrent iodine spike case is considered in which the reactor coolant iodine production rate increases to 335 times the iodine appearance rate that produces the maximum equilibrium value allowed by Tech. Specs. The concurrent spike duration is assumed to continue for 8 hours.
Appendix F 3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the reactor coolant.
Appendix F 4	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	The chemical form of the iodine released from the steam generators to the environment is 97% elemental and 3% organic.
Appendix F 5	**Transport**		
Appendix F 5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Primary-to-secondary leakage is 0.25 gpm per SG and 1 gpm to all steam generators. These values are consistent with the proposed change to the leakage performance criteria of the steam generator program described in the Tech. Specs.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix F 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density used to convert volumetric leak rates to mass leak rates corresponds to a temperature of 70°F and a pressure of 14.7 psia as directed by plant leak rate monitoring procedures.
Appendix F 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Primary-to-secondary leakage and releases from intact steam generators is assumed to continue for 24 hours. This value conservatively bounds the time required to cool the RCS to 212°F and to place shutdown cooling in service.
Appendix F 5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	The release of fission products from the secondary system is evaluated assuming a coincident loss of offsite power.
Appendix F 5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gases are released to the environment without reduction or mitigation.
Appendix F 5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators. For the SGTR event, a partition coefficient of 100 is provided by the condenser in addition to partitioning by the water in the steam generators between the initiation of the event and the time of reactor trip.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix G	**Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident**		
Appendix G	**Source Term**		
Appendix G 1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	The fission product inventory conforms to the guidelines of Position 3 of Reg. Guide 1.183. The release of radionuclides from breached fuel is based upon the fraction of the core inventory within the fuel rod gap from Table 3 and the amount of damaged fuel.
Appendix G 2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.	Conforms	The D. C. Cook Locked Rotor analysis is performed with damage to 11% of the fuel in the core.
Appendix G 3	The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.	Conforms	The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
Appendix G 4	The chemical form of radioiodine released from the fuel should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.	Conforms	The chemical form of the iodine released from the steam generators to the environment is 97% elemental and 3% organic.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix G 5	**Transport**		
Appendix G 5.1	The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.	Conforms	Primary-to-secondary leakage is 1 gpm to all steam generators. This value is consistent with the leakage performance criteria of the steam generator program described in the Tech. Specs.
Appendix G 5.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).	Conforms	The density used to convert volumetric leak rates to mass leak rates corresponds to a temperature of 70°F and a pressure of 14.7 psia as directed by plant leak rate monitoring procedures.
Appendix G 5.3	The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Primary-to-secondary leakage and releases from intact steam generators is assumed to continue for 24 hours. This value conservatively bounds the time required to cool the RCS to 212°F and to place shutdown cooling in service.
Appendix G 5.4	The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.	Conforms	The release of fission products from the secondary system is evaluated assuming a coincident loss of offsite power.
Appendix G 5.5	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gases are released to the environment without reduction or mitigation.
Appendix G 5.6	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix H	**Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident**		
Appendix H	**Source Term**		
Appendix H 1	Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.	Conforms	The fission product release is based upon the amount of damaged fuel and the assumption that 10% of the core inventory of noble gases and iodines are in the fuel rod gap. In addition, 12% of the alkali metals are assumed to be in the fuel gap based upon Table 3 of Reg. Guide 1.183. For releases from containment which involve fuel melting, 100% of the noble gases and 25 % of the iodines contained in the portion of the fuel which melts is available for release from containment. For releases from the secondary system, 100% of the noble gases and 50% f the iodines in the fraction of the core that melts is released into the RCS.
Appendix H 2	If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA), main steam line break, and steam generator tube rupture.	Conforms	The D. C. Cook Locked Rotor analysis is performed with damage to 10% of the fuel in the core due to DNB and 0.25% of the fuel with centerline melting.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix H 3	Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.	Conforms	Two release cases are considered. In the release from containment, 100% of the activity available for released from the fuel is instantaneously and homogeneously distributed through the containment atmosphere. In the case with the release from the secondary system, 100% of the activity released from the fuel is completely dissolved in the RCS and is available for release from the steam generators.
Appendix H 4	The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The chemical form of radioiodine released to the containment atmosphere is assumed to be 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide. Since containment sprays will not necessarily be activated in this event, which prevents assurances that the pH will be controlled at values of 7 or greater, the control room recirculation filter efficiency for particulates is conservatively reduced to the same value as that for elemental and organic iodine.
Appendix H 5	Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.	Conforms	The chemical form of the iodine released from the steam generators to the environment is 97% elemental and 3% organic.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix H	**Transport from Containment**		
Appendix H 6.1	A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.	Conforms	Radioactive material removal from the containment atmosphere by sprays and other engineered safety features is not credited. Natural deposition of elemental iodine is conservatively ignored. Plateout of aerosols is credited with a removal coefficient of 0.1/hr.
Appendix H 6.2	The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.	Conforms	The initial containment leak rate applied in the analysis is equal to the proposed Tech. Spec. value of 0.18% per day. After 24 hours, the leak rate is reduced to 0.09% per day.
Appendix H	**Transport from Secondary System**		
Appendix H 7.1	A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.	Conforms	Primary-to-secondary leakage is 1 gpm to all steam generators. This value is consistent with the leakage performance criteria of the steam generator program described in the Tech. Specs. Primary-to-secondary leakage and releases from intact steam generators are assumed to continue for 24 hours. This value conservatively bounds the time required to cool the RCS to 212°F and to place shutdown cooling in service.

RG 1.183 Section	Regulatory Guide 1.183 Position	Compliance	Basis of Compliance
Appendix H 7.2	The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft3).	Conforms	The density used to convert volumetric leak rates to mass leak rates corresponds to a temperature of 70°F and a pressure of 14.7 psia as directed by plant leak rate monitoring procedures.
Appendix H 7.3	All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.	Conforms	All noble gases are released to the environment without reduction or mitigation.
Appendix H 7.4	The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.	Conforms	The transport model described in Position 5.5 and 5.6 of Appendix E is applied to releases from the steam generators.

Enclosure 11 to AEP-NRC-2014-65

D. C. Cook AST Regulatory Issue Summary 2006-04 Compliance Matrix

RIS 2006-04 Issue	Licensee Comments
1.) Level of Detail Contained in LARs An AST amendment request should describe the licensee's analyses of the radiological and non-radiological impacts and provide a justification for the proposed modification in sufficient detail to support review by the NRC staff. For example, the AST amendment request should (1) provide justification for each individual proposed change to the technical specifications (TS), (2) identify and justify each change to the licensing basis accident analyses, and (3) contain enough details (e.g., assumptions, computer analyses input and output) to allow the NRC staff to confirm the dose analyses results in independent calculations. The provision of sufficient detail is necessary for the NRC staff to be able to conclude, with reasonable assurance, whether the licensee's analyses and changes are acceptable. For a previous NRC staff discussion on the level of detail necessary for review, see RIS 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests". In response to RAIs, some licensees have made changes to originally proposed LARs and their supporting analyses. In some cases, these changes were extensive or involved multiple re-analyses and supplements. Because of the depth and scope of many AST submittals, multiple changes to the original submittal (particularly those with multiple supplements that revise portions of previous supplements) can increase the chance of NRC staff using information that has been superseded during the review. For these cases, NRC staff recommends that licensees identify the most current analyses, assumptions, and TS changes in their submittal and supplements to the submittal.	The license amendment request identifies and provides justification for each of the proposed changes to the Technical specifications. In addition, the technical report included with the submittal provides sufficiently detailed assumptions, analysis inputs, and calculation outputs to allow the NRC staff to independently confirm the dose analysis results. This analysis is based upon the most current set of licensing basis accident analyses and proposed Tech. Spec. changes.

RIS 2006-04 Issue	Licensee Comments
2.) Main Steam Isolation Valve (MSIV) Leakage and Fission Product Deposition in Piping For calculation of aerosol settling velocity in the main steam line (MSL) piping of boiling water reactors, some LARs reference Accident Evaluation Report (AEB) 98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term". This is acceptable. However, it is important to note that the report was written based on the parameters of a particular plant and, therefore, the removal rate constant is specific to that plant. Any licensee who chooses to reference these AEB 98-03 assumptions should provide appropriate justification that the assumptions are applicable to their particular design.	This item is pertains to BWRs only and is not applicable to D. C. Cook.
3.) Control Room Habitability When implementing an AST, some licensees have proposed that certain engineered safety features (ESF) ventilation systems not be credited as a mitigation feature in response to an accident. In some cases, the licensee's revised design basis analysis introduced the assumption that normal (non-ESF) ventilation systems are operating during all or part of an accident scenario. Such an assumption is inappropriate unless the non-ESF system meets certain qualities, attributes, and performance criteria as described in RG 1.183, Regulatory Positions 4.2.4 and 5.1.2. For example, credit for the operation of non-ESF ventilation systems should not be assumed unless they have a source of emergency power. In addition, the operation of ventilation systems establishes certain building or area pressures based upon their flow rates. These pressures affect leakage and infiltration rates which ultimately affect operator dose. Therefore, to credit the use of these systems, licensees should incorporate the systems into the ventilation filter testing program in Section 5 of the TS. In summary, use of non-ESF ventilation systems during a DBA should not be assumed unless the systems have emergency power and are part of the ventilation filter testing program in Section 5 of the TS.	No non-ESF ventilation systems have been credited in the radiological consequence analysis. The AST analysis uses an unfiltered inleakage rate into the control room envelope of 40 cfm, which provides margin to the actual tracer gas test results.

RIS 2006-04 Issue	Licensee Comments
Generic Letter (GL) 2003-01, "Control Room Habitability" requested licensees to confirm the ability of their facility's control room to meet applicable habitability regulatory requirements. In addition, licensees were requested to confirm that control room habitability systems were designed, constructed, configured, operated and maintained in accordance with the facility's design and licensing bases. The GL placed emphasis on licensees confirming that the most limiting unfiltered inleakage into the control room envelope (CRE) was not greater than the value assumed in the DBA analyses. Some AST amendment requests proposed operating schemes for the control room and other ventilation systems which affect areas adjacent to the CRE and are different from the manner of operation and performance described in the response to the GL without providing sufficient justification for the proposed changes in the operating scheme.	
4.) Atmospheric Dispersion Licensees may continue to use atmospheric relative concentration (X/Q) values and methodologies from their existing licensing-basis analyses when appropriate. Licensees also have the option to adopt the generally less conservative (more realistic) updated NRC staff guidance on determining X/Q values in support of design basis control room radiological habitability assessments provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants". Regulatory positions on X/Q values for offsite (i.e., exclusion area boundary and low population zone) accident radiological consequence assessments are provided in RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants".	The license amendment request includes revised atmospheric dispersion factors that were developed based upon Reg. Guide 1.194 using ARCON96 for the control room receptor locations; and calculated with PAVAN using the guidance of Reg. Guide 1.145 for offsite dose locations. The submittal includes a site plan sketch which shows the site orientation with respect to true north and includes the positions of all release and receptor points. In addition, the application provides a basis for the use of the normal control room ventilation intakes

RIS 2006-04 Issue	Licensee Comments
Based on submittal reviews, the NRC staff identified the following areas of improvement for licensee submittals that propose revision of the design basis atmospheric dispersion analyses for implementing AST. They should include the following information: • A site plan showing true North and indicating locations of all potential accident release pathways and control room intake and unfiltered inleakage pathways (whether assumed or identified during inleakage testing). • Justification for using control room intake X/Q values for modeling the unfiltered inleakage, if applicable. • A copy of the meteorological data inputs and program outputs along with a discussion of assumptions and potential deviations from staff guidelines. Meteorological data input files should be checked to ensure quality (e.g., compared against historical or other data and against the raw data to ensure that the electronic file has been properly formatted, any unit conversions are correct, and invalid data are properly identified). When running the control room atmospheric dispersion model ARCON96, two or more files of meteorological data representative of each potential release height should be used if X/Q values are being calculated for both ground-level and elevated releases (see RG 1.23, "Onsite Meteorological Programs," Regulatory Position 2 and Table A-2 in Appendix A to RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"). In addition, licensees should be aware that (1) two levels of wind speed and direction data should always be provided as input to each data file, (2) fields of "nines" (e.g., 9999) should be used to indicate invalid or missing data, and (3) valid wind direction data should range from 1° to 360°. Licensees should also provide detailed engineering information when	as the location for unfiltered inleakage into the control room envelope. Atmospheric dispersion factor program inputs are listed in the submittal, and meteorological data used by the programs is available electronically. The data has been verified to ensure quality and properly formatted for use by ARCON96 and PAVAN. Wind speed categories established for the joint frequency data include 0.22, 0.50, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0, and 10.0 meters/second. All releases are treated as ground level releases, and the plume rise model discussed in Reg. Guide 1.194 has not been applied.

RIS 2006-04 Issue	Licensee Comments
applying the default plume rise adjustment cited in RG 1.194 to control room X/Q values to account for buoyancy or mechanical jets of high energy releases. This information should demonstrate that the minimum effluent velocity during any time of the release over which the adjustment is being applied is greater than the 95th percentile wind speed at the height of release. When running the offsite atmospheric dispersion model PAVAN, two or more files of meteorological data representative of each potential release height should be used if X/Q values are being calculated for pathways with significantly different release heights (e.g., ground level versus elevated stack). The joint frequency distributions of wind speed, wind direction, and atmospheric stability data used as input to PAVAN should have a large number of wind speed categories at the lower wind speeds in order to produce the best results (e.g., Section 4.6 of NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations", suggests wind speed categories of calm, 0.5, 0.75, 1.0, 1.25, 1.5, 2.0, 3.0, 4.0 5.0, 6.0, 8.0 and 10.0 meters per second).	
5.) Modeling ESF Leakage ESF systems that recirculate sump water outside the primary containment may leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (e.g., refueling water storage tank). Appendix A to RG 1.183, Regulatory Position 5, states that "the radiological consequences from the postulated [ESF] leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the [loss-of-coolant accident] LOCA."	The dose contribution due to ESF leakage from ECCS systems during post-LOCA recirculation operation is analyzed and combined with the consequences of other fission product release pathways. Two different ESF leakage pathways are considered. The first pathway involves leakage through valve packing glands, pump shaft seals, and flanged connections in the Auxiliary Building. The leakage rate is assumed to be twice the administrative limit established by plant

RIS 2006-04 Issue	Licensee Comments
The allowable ESF leakage is typically contained in the plant's TS or procedures. The ESF leakage at accident conditions may differ from the ESF leakage at normal operating conditions. Licensees should account for ESF leakage at accident conditions in their dose analyses so as not to underestimate the release rate. In Appendix A to RG 1.183, Regulatory Position 5.5, the NRC staff provided a conservative value of 10 percent as the assumed amount of iodine that may become airborne from ESF leakage that is less than 212 °F. The NRC staff structured this regulatory position to be deterministic and conservative. The 10 percent value also compensates for the lack of research concerning iodine speciation beyond the containment and the uncertainties of applying laboratory data to the post-accident environment of the plant. Regulatory Position 5.5 states that a smaller flash fraction could be justified. Some licensees have referenced NUREG/CR-5950, "Iodine Evolution and pH Control" to justify a smaller flash fraction. However, NUREG/CR-5950 was developed for very specific laboratory conditions and the results have a degree of uncertainty. The mechanism for release of the fluid is also uncertain. Leaked fluid may spray onto surfaces and evaporate, or be sprayed in fine droplets into the air. A value of less than 10 percent can be justified by including considerations for plant-specific variables, including the post-accident environment (e.g., impurities in the water or the presence of organic substances) and the uncertainties in the application of research situations to plant environments. Figure 3.1 in NUREG/CR-5950 can be used to quantify the amount of elemental iodine as a function of the sump water pH and the concentration of iodine in the solution. In some cases, however, licensees have misapplied this figure. Rather than using the total concentration of iodine (i.e., stable and radioactive), licensees based their assessment on only the radioactive iodine in the sump water. By using only the radioactive iodine, licensees have underestimated how much iodine evolves during post-accident conditions.	procedures, and 10% of the iodine in the leaked fluid is assumed to become airborne. The second leakage pathway considers seat leakage past valves which isolate interfacing systems that circulate sump ESF fluid to the RWST. A separate administrative leakage limit is established for this pathway. The amount of elemental iodine in the RWST that becomes volatile and is available for release is based upon NUREG/CR-5950 and is a function of both the pH of the tank liquid and the iodine concentration. Both the amount of radioactive and stable iodine is considered in the calculation of the total RWST iodine concentration.

RIS 2006-04 Issue	Licensee Comments
6.) Release Pathways Changes to the plant configuration associated with an LAR (e.g., an "open" containment during refueling) may require a re-analysis of the design basis dose calculations. A request for TS modifications allowing containment penetrations (i.e., personnel air lock, equipment hatch) to be open during refueling cannot rely on the current dose analysis if this analysis has not already considered these release pathways. RG 1.194, Regulatory Position 3.2.4.2 supports review of penetration pathways, by stating that "leakage is more likely to occur at a penetration, [and that the] analysts must consider the potential impact of leakage from building penetrations exposed to the environment." Therefore, releases from personnel air locks and equipment hatches exposed to the environment and containment purge releases prior to containment isolation need to be addressed. Some licensees have identified unique release pathways that had not been previously considered. For example, a recent submittal noted that containment hatches and containment plugs may be removed during refueling. The removal of these barriers creates new release pathways. Licensees are responsible for identifying all release pathways and for considering these pathways in their AST analyses, consistent with any proposed modification.	No changes to the plant configuration which impact release pathways are being proposed. The onside atmospheric dispersion factors developed to support the submittal considers all possible release locations from containment, secondary systems, and plant systems located within the Auxiliary Building. Since the D. C. Cook containment is permitted to be open during refueling, the release location from containment during the Fuel Handling Accident is assumed to be a location on the containment surface closest to the control room intakes.
7.) Primary to Secondary Leakage Some analysis parameters can be affected by density changes that occur in the process steam. The NRC staff continues to find errors in LAR submittals concerning the modeling of primary to secondary leakage during a postulated accident. This issue is discussed in Information Notice (IN) 88-31, "Steam Generator Tube Rupture Analysis Deficiency," and Item 3.f in RIS 2001-19. An acceptable methodology for modeling this leakage is provided in Appendix F to RG 1.183, Regulatory Position 5.2.	The density used in converting volumetric leak rates to mass leak rates corresponds to the fluid temperature specified in surveillance tests used to satisfy the requirements of the reactor coolant leak rate monitoring program. This approach is consistent with Position 5.2 of Appendix F to Reg. Guide 1.183.

RIS 2006-04 Issue	Licensee Comments
8.) Elemental Iodine Decontamination Factor (DF) Appendix B to RG 1.183, provides assumptions for evaluating the radiological consequences of a fuel handling accident. If the water depth above the damaged fuel is 23 feet or greater, Regulatory Position 2 states that "the decontamination factors for the elemental and organic [iodine] species are 500 and 1, respectively, giving an overall effective decontamination factor of 200." However, an overall DF of 200 is achieved when the DF for elemental iodine is 285, not 500.	An elemental iodine decontamination factor of 285 is applied in the analysis of the Fuel Handling Accident. This value, in combination with an organic iodine decontamination factor of 1.0, results in an overall DF 0f 200.
9.) Isotopes Used in Dose Assessments For some accidents (e.g., main steam line break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose assessments for AST implementation.	The analysis source term is based upon elements listed in Table 5 of Reg. Guide 1.183. Noble gases and alkali metals are included in this source term, and are considered both in releases of RCS activity to the environment and in releases of fuel gap activities to the RCS during events which involve fuel failures.
10.) Definition of Dose Equivalent I-131 In the conversion to an AST, licensees have proposed a modification to the TS definition of dose equivalent I-131. Some have modified the definition to base it upon the thyroid dose conversion factors of International Commission on Radiation Protection (ICRP) Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation" or ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers". Others have proposed a definition which is a combination of different iodine dose conversion factors, (e.g., RG 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR [Part] 50, Appendix I", ICRP Publication 2, Federal Guidance Report	The proposed Definition of Dose Equivalent I-131 in the license submittal specifies the use of thyroid dose conversion factors (DCF) from Federal Guidance Report 11. These same DCFs are used in the analysis to establish the equilibrium iodine activities in the RCS source term, and to determine the iodine appearance rates in the Main Steam Line Break and Steam Generator Tube Rupture events.

RIS 2006-04 Issue	Licensee Comments
11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion". Although different references are available for dose conversion factors, the TS definition should be based on the same dose conversion factors that are used in the determination of the reactor coolant dose equivalent iodine curie content for the main steam line break and steam generator tube rupture accident analyses.	
11.) Acceptance Criteria for Off-Gas or Waste Gas System Release As part of full AST implementation, some licensees have included an accident involving a release from their off-gas or waste gas system. For this accident, they have proposed acceptance criteria of 500 millirem (mrem) total effective dose equivalent (TEDE). The acceptance criteria for this event is that associated with the dose to an individual member of the public as described in 10 CFR Part 20, "Standards for Protection Against Radiation." When the NRC revised 10 CFR Part 20 to incorporate a TEDE dose, the offsite dose to an individual member of the public was changed from 500 mrem whole body to 100 mrem TEDE. Therefore, any licensee who chooses to implement AST for an off-gas or waste gas system release should base its acceptance criteria on 100 mrem TEDE. Licensees may also choose not to implement AST for this accident and continue with their existing analysis and acceptance criteria of 500 mrem whole body.	The analysis of releases from ruptures of the Waste Gas Decay Tank (WGDT) and the Volume Control Tank (VCT) are not included in the license submittal. The EAB acceptance criteria for these events continues to be the 500 mrem limit in the existing offsite analysis. This value is also applied to the LPZ. The 5 rem TEDE dose limit from 10CFR50.67 is applied in the control room habitability analysis.
12.) Containment Spray Mixing Some plants with mechanical means for mixing containment air have assumed that the containment fans intake air solely from a sprayed area and discharge it solely to an unsprayed region or vice versa. Without additional analysis, test measurements or further justification, it should be assumed that the intake of air by containment ventilation systems is supplied proportionally to the sprayed and unsprayed volumes in containment.	The flow rates between compartments in the D. C. Cook containment model that result from operation of the Containment Ventilation (CEQ) system are based upon the actual design configuration of the system. The CEQ fans take suction from the

RIS 2006-04 Issue	Licensee Comments
	compartments throughout the building and discharge into the lower containment. The distribution of the discharge flow into the sprayed and unsprayed portions of the lower containment in the model is proportional to the respective sprayed and unsprayed volumes.

Enclosure 12 to AEP-NRC-2014-65

D. C. Cook AST Accident Analyses Input Values Comparison Tables

D. C. Cook AST Accident Analyses Input Values Comparison Tables

During a public meeting (Reference 1) regarding this submittal, I&M agreed to include summary tables for each accident being analyzed including a comparison between current licensing basis (CLB) input parameters and the values utilized in the new alternative source term (AST) accident analyses. The tables are provided within this enclosure for the following accident scenarios:

Table 2 - Loss of Coolant Accident (LOCA)

Table 3 - Fuel Handling Accident (FHA)

Table 4 - Main Steam Line Break (MSLB)

Table 5 - Steam Generator Tube Rupture (SGTR)

Table 6 - Locked Rotor Accident (LRA)

Table 7 - Control Rod Ejection (CRE)

Table 8 - Waste Gas Decay Tank (WGDT) Rupture

Table 9 - Volume Control Tank (VCT) Rupture

Additionally, Table 1, "Control Room Parameters," is provided to show the parameters that are shared among all accident analyses for control room habitability (CRH).

The current CNP licensing basis for offsite radiological consequence analyses is based on methodologies prescribed in Regulatory Guide (RG) 1.195. The current CNP licensing basis for CRH radiological consequence analyses is based upon AST methodology from RG 1.183. Therefore, with the exception of the Table 1, "Control Room Parameters," each table includes a column for "CLB Offsite Value", "CLB CRH AST Value", and a column representing the "New AST Value for Offsite and CRH," which is obtained from the report provided in Enclosure 9 to this letter. The right-most column in each table provides reasons for instances in which the CLB parameters differ from the new AST values.

Errors were introduced into the CLB from dose consequence re-analyses performed in the 2007-2010 time frame at CNP, subsequent to the previous control room habitability AST LAR. The errors are being managed via D.C. Cook's corrective actions program and have been appropriately assessed for operability. Additionally, offsite dose analyses for the LRA and CRE accident scenarios are not part of the current CNP licensing basis and the corresponding tables have been labeled accordingly.

<u>References</u>:

1. Summary of the August 8, 2014, Pre-Application Meeting to Discuss Pending License Amendment Request Associated with the Implementation of Alternative Source Term And Technical Specification Task Force Traveler (TSTF)-490 (TAC NOS. MF4483 AND MF4484), dated August 27, 2014 (ML14231A07)

Table 1: Control Room Parameters			
Input/Assumption	**CLB CRH AST Value**	**New AST Value**	**Reason for Change**
Control Room Volume	50,000 ft3	50,616 ft3	CLB and new AST values were taken from same source document. The CLB value was arbitrarily rounded down to add conservatism, which is not necessary for the new analysis.
Normal Operation			
Filtered Make-up Flow Rate	0 cfm	0 cfm	
Filtered Recirculation Flow Rate	0 cfm	0 cfm	
Unfiltered Make-up Flow Rate	880 cfm	880 cfm	
Unfiltered Inleakage	80 cfm	40 cfm	The CLB value was arbitrarily doubled to add conservatism, which is not necessary for the new analysis.
Emergency Operation			
Recirculation Mode:			
Filtered Make-up Flow Rate	880 cfm	880 cfm	
Filtered Recirculation Flow Rate	4520 cfm	4520 cfm	
Unfiltered Make-up Flow Rate	0 cfm	0 cfm	
Unfiltered Inleakage	80 cfm	40 cfm	
Filter Efficiencies			
Elemental	94.05%	94.05%	
Organic	94.05%	94.05%	
Particulate	99%	98.01%	New AST value models 1% bypass.
Occupancy			
0-24 hrs 1-4 days 4-30 days	1 0.6 0.4	1 0.6 0.4	
Breathing Rate	3.5 x 10-4 m3/sec	3.5 x 10-4 m3/sec	

Table 2: LOCA Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Containment Purge				
Iodine Chemical Form	5% aerosol, 91% elemental, 4% organic	95% aerosol, 4.85% elemental, 0.15% organic	95% aerosol, 4.85% elemental, 0.15% organic	Difference in methodology between RG 1.195 and RG 1.183.
Containment Volume	1,044,000 ft3	1,044,000 ft3	1,066,352 ft3	The CLB value was arbitrarily rounded down to add conservatism, which is not necessary for the new analysis.
Containment Purge Flow Rate	23,100 cfm	23,100 cfm	36,300 cfm	Values used in previous analyses represent supply from upper containment. New value represents purge fan exhaust.
Containment Purge Isolation Time	30 sec	30 sec	15 seconds	The CLB value was arbitrarily chosen to add conservatism, which is not necessary for the new analysis.
Containment Purge Filtration	0%	0%	0%	
Removal by Wall Deposition	None	None	None	
Removal by Sprays	None	None	None	
Containment Leakage				
Iodine Chemical Form	5% aerosol, 91% elemental, 4% organic	95% aerosol, 4.85% elemental, 0.15% organic	95% aerosol, 4.85% elemental, 0.15% organic	Difference in methodology between RG 1.195 and RG 1.183.
Containment Sump pH	>7.0	>7.0	>7.0	
Compartment Volumes (max)				Differences are due to rounding and application of a 2% conservative bias.
Upper Containment (Sprayed)	609,000 ft3	609,000 ft3	621,968 ft3	
Lower Compartment (Sprayed)	101,000 ft3	101,000 ft3	103,770 ft3	
Fan Rooms (Sprayed)	47,000 ft3	47,000 ft3	48,913 ft3	
Upper Containment (Unsprayed)	120,000 ft3	120,000 ft3	122,600 ft3	
Ice Condenser (Unsprayed)	103,000 ft3	103,000 ft3	105,577 ft3	
Lower Containment (Unsprayed)	64,000 ft3	64,000 ft3	66,188 ft3	
Dead-End (Unsprayed)	18,000 ft3	18,000 ft3	18,663 ft3	

Table 2: LOCA Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Containment Ventilation Start Time	180 seconds	180 seconds	300 seconds	Higher value used to provide flexibility for future plant changes.
Containment Ventilation Flow Rate				Differences are due to the new volumes which are used to ratio the flows (see above).
Fan Rooms to Lower Containment (Unsprayed)	14,530 cfm	14,530 cfm	14,580.5 cfm	
Fan Rooms to Lower Containment (Sprayed)	22,910 cfm	22,910 cfm	22,859.5 cfm	
Lower Containment (Unsprayed) to Dead –End	90 cfm	90 cfm	90 cfm	
Dead-End to Fan Rooms	90 cfm	90 cfm	90 cfm	
Lower Containment (Unsprayed) to Fan Rooms	1,350 cfm	1,350 cfm	1,350 cfm	
Lower Containment (Unsprayed) to Ice Condenser	13,090 cfm	13,090 cfm	13,140.5 cfm	
Lower Containment (Sprayed) to Ice Condenser	22,910 cfm	22,910 cfm	22,859.5 cfm	
Ice Condenser to Upper Containment (Sprayed)	30,070 cfm	30,070 cfm	30,072.3 cfm	
Ice Condenser to Upper Containment (Unsprayed)	5,930 cfm	5,930 cfm	5,927.7 cfm	
Upper Containment (Sprayed) to Fan Rooms	30,070 cfm	30,070 cfm	30,072.3 cfm	
Upper Containment (Unsprayed) to Fan Rooms	5,930 cfm	5,930 cfm	5,927.7 cfm	
Lower Containment – Sprayed to/from Unsprayed	2,133 cfm (spray induced circulation)	2,133 cfm (spray induced circulation)	2206.3 cfm (spray induced circulation)	
Upper Containment – Sprayed to/from Unsprayed	4,000 cfm (spray induced circulation)	4,000 cfm (spray induced circulation)	4086.7 cfm (spray induced circulation)	
Sprayed/Unsprayed Volume Induced Mixing Flow Rate	2 Turnovers of Unsprayed Compartment/hour	2 Turnovers of Unsprayed Compartment/hour	2 Turnovers of Unsprayed Compartment/hour	

Table 2: LOCA Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Containment Spray Start Time	180 seconds	180 seconds	300 seconds	Higher value used to provide flexibility for future plant changes.
Containment Spray Stop Time	0.5167-0.6334 hours and after 24 hours	0.5167-0.6334 hours and after 24 hours	0.319-0.426 hours and after 24 hours	A different source document was utilized for these values. New AST value is conservatively low.
Containment Spray Flow Rate				
Upper Containment	1466 gpm	1466 gpm	1466 gpm	
Lower Containment	660 gpm	660 gpm	660 gpm	
Fan Rooms	201 gpm	201 gpm	201 gpm	
Containment Spray Drop Fall Height				
Upper Containment	58.6 ft	58.6 ft	58.6 ft	
Lower Containment	28.5 ft	28.5 ft	28.5 ft	
Fan Rooms	20.1 ft	20.1 ft	20.1 ft	
Containment Spray Mean Drop Diameter				
Upper Containment	609 microns	609 microns	609 microns	
Lower Containment	671 microns	671 microns	671 microns	
Fan Rooms	671 microns	671 microns	671 microns	
Elemental Iodine Spray Removal Coefficient	10 hr-1, with a total decontamination factor of 200	20 hr-1, with a total decontamination factor of 200	20 hr-1, with a total decontamination factor of 200	Difference in methodology between RG 1.195 and RG 1.183.
Time that Total Elemental DF reaches 200	0.9 hours	2.352 hours	2 hours	Output of computer code, which is case-specific.
Aerosol Spray Removal Coefficient				Differences are due to recalculation. The values are a function of the compartment volume and spray flow rate, which are slightly different as represented above.
Upper Containment	5.17 hr-1	5.17 hr-1	5.06 hr-1	
Lower Containment	6.83 hr-1	6.83 hr-1	6.65 hr-1	
Fan Rooms	3.15 hr-1	3.15 hr-1	3.03 hr-1	

Table 2: LOCA Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Time that Total Aerosol DF reaches 50	1.214 hours	2.615 hours	2.32 hours	Output of computer code, which is case-specific.
Organic Iodine Spray Removal	None	None	None	
Natural Deposition	Elemental, Organic Iodine – None	Elemental, Organic Iodine – None	Elemental, Organic Iodine – None	
	Aerosols - None	Aerosols - None	Aerosols - 0.1 hr-1 in unsprayed regions only	New value is permitted per Appendix A of RG 1.183.
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days	 0.25 %/day 0.125 %/day	 0.25 %/day 0.125 %/day	 0.18 %/day 0.09 %/day	TS 5.5.14.c, which establishes the allowable containment leakage rate, is being modified to provide analytical margin.
Containment Leakage Filtration	0%	0%	0%	
ESF Leakage to the Auxiliary Building				
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	
Containment Sump Volume	50,000 ft3	50,000 ft3	50,955 ft3	New value has not been rounded down.
ECCS Recirculation Start Time	1,860 seconds	1,860 seconds	1388.4 seconds	A different source document was utilized for these values. New AST value is conservatively low.
ESF Leakage Flow Rate	0.2 gpm (two times the allowable value, including RWST)	0.2 gpm (two times the allowable value, including RWST)	0.2 gpm (two times the allowable value)	New AST analyses explicitly models . leakage to the RWST in a separate case. The old analysis did not use this approach.
ESF Leakage Flashing Fraction	10%	10%	10%	
Auxiliary Building Ventilation Filtration	0%	0%	0%	
ESF Leakage to the RWST (Not Explicitly Modeled in CLB)				

Table 3: FHA Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Iodine Chemical Form	0% aerosol, 99.75% elemental, 0.25% organic	0% aerosol, 99.85% elemental, 0.15% organic	0% aerosol, 99.85% elemental, 0.15% organic	Difference in methodology between RG 1.195 and RG 1.183.
Number of Fuel Assemblies Damaged	1	1	1	

Table 3: FHA Inputs and Assumptions				
Percentage of Fuel Rods Failed	100%	100%	100%	
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU	Not Modeled	Not Modeled	150	High burnup rods modeled to allow for flexibility in future fuel cycles.
High burnup multiplier applied to gap fractions	Not Modeled	Not Modeled	2	High burnup rods modeled to allow for flexibility in future fuel cycles.
Water Level Above Damaged Fuel	23 feet	23 feet	23 feet	
Pool Decontamination Factors	Elemental - 326 Organic - 1.0	Elemental - 285 Organic - 1.0	Elemental - 285 Organic - 1.0	Difference in methodology between RG 1.195 and RG 1.183.
Delay Before Fuel Movement	Range (72 hours, 100 hours, 120 hours)	Range (72 hours, 100 hours, 120 hours)	120 hours	New AST value represents current TRO 8.9.2 value for reactor sub-criticality prior to fuel movement.
Containment Release Filtration	0%	0%	0%	
Fuel Handling Area Exhaust Ventilation Filtration	Aerosol - 0% Elemental - 89.1% Organic - 89.1%	Aerosol - 0% Elemental - 89.1% Organic - 89.1%	Aerosol - 98.01% Elemental - 89.1% Organic - 89.1%	Previous analyses arbitrarily set particulate/aerosol filtration to zero.

Table 4: MSLB Inputs and Assumptions				
Input/Assumption	**CLB Offsite Value**	**CLB CRH AST Value**	**New AST Value for Offsite and CRH**	**Reason for Change**
Maximum Pre-Accident Iodine Spike Concentration	60 µCi/gm Dose Equivalent I-131	60 µCi/gm Dose Equivalent I-131	60 µCi/gm Dose Equivalent I-131	
Concurrent Iodine Spike Appearance Rate	500x Equilibrium	500x Equilibrium	500x Equilibrium	
Initial Steam Generator Iodine Source Term	0.1 µCi/gm Dose Equivalent I-131	0.1 µCi/gm Dose Equivalent I-131	0.1 µCi/gm Dose Equivalent I-131	
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	
Percentage of Fuel Rods Failed	0%	0%	0%	
RCS Mass	499,325 lbm	499,325 lbm	466,141.5 lbm	New AST value conservatively biased low.
Steam Generator Secondary Liquid Mass	91,000 lbm/SG 161,000 lbm/SG	91,000 lbm/SG 161,000 lbm/SG	97,515.7 lbm/SG 161,000 lbm/SG	New AST values for min and max mass are taken from same source document. The CLB values used different calculations as source for min and max mass.
Intact Steam Generator Steam Release	0 - 2 hours: 456,000 lbm 2 - 8 hours: 1,186,000 lbm 8 - 24 hours: Not Modeled	0 - 2 hours: 456,000 lbm 2 - 8 hours: 1,186,000 lbm 8 - 24 hours: Not Modeled	0 - 2 hours: 456,000 lbm 2 - 8 hours: 1,186,000 lbm 8 - 24 hours: 1,347,000 lbm	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.

Table 4: MSLB Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Primary-Secondary Leak Rate	150 gpd/SG	150 gpd/SG	0.25 gpm to each steam generator	New AST value utilizes the accident induced leakage rate from the proposed revision to TS 5.5.7.b.2. The CLB values represent the operational leakage from TS 3.4.13. This issue is being managed via D.C. Cook's corrective actions program.
Density Used for Leakage Volume-to-Mass Conversion	62.4 lbm/ft3	62.4 lbm/ft3	62.3 lbm/ft3	New AST value is consistent with the current reactor coolant leak rate monitoring program. Difference is negligible.
Duration of Intact SG Tube Uncovery After Reactor Trip	Not Modeled	Not Modeled	40 minutes	Tube uncovery was not explicitly modeled in previous analysis. New AST value conservatively derived from plant simulator data.
Tube Leakage Flashing Fraction During Uncovery	Not Modeled	Not Modeled	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%	See above.
Time to Cool RCS to 212F	8 hours	8 hours	24 hours	Justification for CLB value not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Intact Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Not Considered	Unflashed Leakage - 100 Not Considered	Unflashed Leakage - 100 Flashed Leakage - 0	
Intact Steam Generator Moisture Carryover Fraction	Not Modeled	Not Modeled	0.2% (Particulate Partition Coefficient = 500)	New AST value conservatively biased high using plant-specific values.

Table 5: SGTR Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Maximum Pre-Accident Iodine Spike Concentration	60 µCi/gm Dose Equivalent I-131	60 µCi/gm Dose Equivalent I-131	60 µCi/gm Dose Equivalent I-131	
Concurrent Iodine Spike Appearance Rate	335x Equilibrium	335x Equilibrium	335x Equilibrium	

Table 5: SGTR Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Initial Steam Generator Iodine Source Term	0.1 µCi/gm Dose Equivalent I-131	0.1 µCi/gm Dose Equivalent I-131	0.1 µCi/gm Dose Equivalent I-131	
Iodine Chemical Form	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	
Percentage of Fuel Rods Failed	0%	0%	0%	
RCS Mass	499,325 lbm	499,325 lbm	466,141.5 lbm	New AST value conservatively biased low.
Steam Generator Secondary Liquid Mass	91,000 lbm/SG	91,000 lbm/SG	97,515.7 lbm/SG	New AST values for min and max mass are taken from same source document. The CLB value used a different calculation as source document for mass.
	Not Modeled	Not Modeled	161,000 lbm/SG	
Intact Steam Generator Steam Release	0 - 2 hours: 565,000 lbm 2 - 8 hours: 1,505,000 lbm 8 - 24 hours: Not Modeled	0 - 2 hours: 565,000 lbm 2 - 8 hours: 1,505,000 lbm 8 - 24 hours: Not Modeled	0 - 30 min: 198,515 lbm 30 min - 2 hours: 314,432 lbm 2 - 8 hours: 1,367,475 lbm 8 - 24 hours: 1,347,000 lbm	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Ruptured Steam Generator Steam Release	0-30 min. - 73,000 lbm	0-30 min. - 73,000 lbm	0-30 min. - 66,171lbm	The CLB value for release mass was arbitrarily increased to add conservatism, which is not necessary for the new analysis.
Pre-Trip Total Steam Flow Rate Through Condenser	17,200,000 lbm/hr	17,200,000 lbm/hr	17,153,800 lbm/hr	New AST value represents 105% of the maximum secondary steam flow rate at 100% rated thermal power.
Time of Reactor Trip	120 seconds	120 seconds	101 seconds	Conservatively low value chosen for New AST value.
Primary-Secondary Leak Rate	150 gpd/SG	150 gpd/SG	0.25 gpm to each steam generator	New AST value utilizes the accident induced leakage rate from the proposed revision to TS 5.5.7.b.2. The CLB values represent the operational leakage from TS 3.4.13. This issue is being managed via D.C. Cook's corrective actions program.
Density Used for Leakage Volume-o-Mass Conversion	62.4 lbm/ft3	62.4 lbm/ft3	62.3 lbm/ft3	New AST value is consistent with the current reactor coolant leak rate monitoring program. Difference is negligible.
Ruptured Tube Break Flow	162,000 lbm	162,000 lbm	146,704 lbm	The CLB value was arbitrarily increased to add conservatism, which is not necessary for the new analysis.
Duration of Ruptured Tube Break Flow	30 minutes	30 minutes	30 minutes	

Table 5: SGTR Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Break Flow Flashing Fraction	Pre-Trip: 0% Post Trip: 0-60 seconds: 12.69% 60-480 seconds: 8.12% 480-28 min: 5.97%	Pre-Trip: 0% Post Trip: 0-60 seconds: 12.69% 60-480 seconds: 8.12% 480-28 min: 5.97%	Pre-Trip: 16% Post Trip: 0-60 seconds: 16% 60-300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-30 min: 4%	New AST value conservatively derived from plant simulator data.
Duration of Intact SG Tube Uncovery After Reactor Trip	Not Modeled	Not Modeled	40 minutes	Tube uncovery was not explicitly modeled in previous analysis. New AST value conservatively derived from plant simulator data.
Tube Leakage Flashing Fraction During Uncovery	Not Modeled	Not Modeled	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%	See above.
Time to Cool RCS to 212F	8 hours	8 hours	24 hours	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Steam Generator Iodine Partition Coefficient	Unflashed Leakage - 100 Flashed Leakage - 0	Unflashed Leakage - 100 Flashed Leakage - 0	Unflashed Leakage - 100 Flashed Leakage - 0	
Condenser Partition Coefficient	100	100	100	
Intact Steam Generator Moisture Carryover Fraction	Not Modeled	Not Modeled	0.2% (Particulate Partition Coefficient = 500)	New AST value conservatively biased high using plant-specific values.

Table 6: LRA Inputs and Assumptions

Input/Assumption	CLB Offsite Value (Not Part of CLB)	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Fuel Rod Gap Fractions		I-131 - 0.08	I-131 - 0.08	
		Kr-85 - 0.10	Kr-85 - 0.10	
		Other Noble Gases - 0.05	Other Noble Gases - 0.05	
		Other Halogens - 0.05	Other Halogens - 0.05	
		Alkali Metals - 0.12	Alkali Metals - 0.12	

Table 6: LRA Inputs and Assumptions

Input/Assumption	CLB Offsite Value (Not Part of CLB)	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Percentage of Fuel Rods Failed		11%	11%	
Fuel Rod Peaking Factor		Not Modeled	1.65	Peaking factor not applied in CLB. This issue is being managed via D.C. Cook's corrective actions program.
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU		Not Modeled	150 rods in two assemblies	High burnup rods modeled to allow for flexibility in future fuel cycles.
High burnup multiplier applied to gap fractions		Not Modeled	1.0104	High burnup rods modeled to allow for flexibility in future fuel cycles.
Initial Steam Generator Iodine Source Term		Not Modeled	0.1 µCi/gm Dose Equivalent I-131	New AST analyses model an initial steam generator iodine release. Prior to the event, the specific iodine activity on the secondary side of the steam generators is modeled at the Technical Specification limit.
Iodine Chemical Form		0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	
RCS Mass		499,325 lbm	466,141.5 lbm	New AST value conservatively biased low.
Steam Generator Secondary Liquid Mass		91,000 lbm/SG	97,515.7 lbm/SG	New AST values for min and max mass are taken from same source document. The CLB value used a different calculation as source document for mass.
		Not Used	161,000 lbm/SG	New AST analyses model an initial steam generator iodine release. This value is utilized to maximize the nuclide inventory available.

Table 6: LRA Inputs and Assumptions

Input/Assumption	CLB Offsite Value (Not Part of CLB)	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Primary-Secondary Leak Rate		150 gpd/SG	0.25 gpm to each steam generator	New AST value utilizes the accident induced leakage rate from the proposed revision to TS 5.5.7.b.2. The CLB values represent the operational leakage from TS 3.4.13. This issue is being managed via D.C. Cook's corrective actions program.
Density Used for Leakage Volume-o-Mass Conversion		62.4 lbm/ft3	62.3 lbm/ft3	New AST value is consistent with the current reactor coolant leak rate monitoring program. Difference is negligible.
Secondary Steam Release		0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: Not Modeled	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Time to Cool RCS to 212F		8 hours	24 hours	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Duration of SG Tube Uncovery Following Reactor Trip		Not Modeled	40 minutes	Tube uncovery was not explicitly modeled in previous analysis. New AST value conservatively derived from plant simulator data.
Intact Tube Leakage Flashing Fraction During Uncovery		Not Modeled	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%	See above.

Table 6: LRA Inputs and Assumptions				
Input/Assumption	**CLB Offsite Value (Not Part of CLB)**	**CLB CRH AST Value**	**New AST Value for Offsite and CRH**	**Reason for Change**
Steam Generator Iodine Partition Coefficient		Unflashed Leakage - 100 Flashed Leakage - 0	Unflashed Leakage - 100 Flashed Leakage - 0	
Steam Generator Moisture Carryover Fraction		Not Modeled	0.2% (Particulate Partition Coefficient = 500)	New AST value conservatively biased high using plant-specific values.

Table 7: CRE Inputs and Assumptions				
Input/Assumption	**CLB Offsite Value (Not Part of CLB)**	**CLB CRH AST Value**	**New AST Value for Offsite and CRH**	**Reason for Change**
Fuel Rod Gap Fractions		Noble Gases - 0.10 Iodines - 0.10 Alkali Metals - 0.12	Noble Gases - 0.10 Other Halogens - 0.10 Alkali Metals - 0.12	Bromine isotopes also considered along with iodines in New AST analysis.
Percentage of Fuel Rods Failed		10%	10%	
Percentage of Fuel That Experiences Melting		0.25%	0.25%	
No. of rods exceeding 6.3 kw/ft above 54 GWD/MTU		Not Modeled	150 rods in two assemblies	High burnup rods modeled to allow for flexibility in future fuel cycles.
High burnup multiplier applied to gap fractions		Not Modeled	1.0104	High burnup rods modeled to allow for flexibility in future fuel cycles.
Fuel Rod Peaking Factor		Not Modeled	1.65	Peaking factor not applied in CLB. This issue is being managed via D.C. Cook's corrective actions program.
Initial Steam Generator Iodine Source Term		Not Modeled	0.1 µCi/gm Dose Equivalent I-131	New AST analyses model an initial steam generator iodine release. Prior to the event, the specific iodine activity on the secondary side of the steam generators is modeled at the Technical Specification limit.
Iodine Chemical Form - Secondary Release		0% aerosol, 97% elemental, 3% organic	0% aerosol, 97% elemental, 3% organic	
Iodine Chemical Form - Containment Release		95% aerosol, 4.85% elemental, 0.15% organic	95% aerosol, 4.85% elemental, 0.15% organic	
Containment Volume		1,062,000 ft3	1,066,352 ft3	Difference is due to rounding.
Containment Leakage Rate 0 to 24 hours 24 hours to 30 days		 0.25 %/day 0.125 %/day	 0.18 %/day 0.09 %/day	TS 5.5.14.c, which establishes the allowable containment leakage rate, is being modified to provide analytical margin.
Containment Leakage Filtration		0%	0%	
Natural Deposition in Containment		Element Iodine - None	Element Iodine - None	
		Aerosols - None	Aerosols - 0.1 hr-1 after 24 hours	New value is permitted per Appendix H of RG 1.183.

Table 7: CRE Inputs and Assumptions

Input/Assumption	CLB Offsite Value (Not Part of CLB)	CLB CRH AST Value	New AST Value for Offsite and CRH	Reason for Change
Iodine/Particulate Removal by Containment Sprays		None	None	
RCS Mass		499,325 lbm	466,141.5 lbm	New AST value conservatively biased low.
Steam Generator Secondary Liquid Mass		91,000 lbm/SG	97,515.7 lbm/SG	New AST values for min and max mass are taken from same source document. The CLB value used a different calculation as source document for mass.
		Not Used	161,000 lbm/SG	New AST analyses model an initial steam generator iodine release. This value is utilized to maximize the nuclide inventory available for release.
Primary-Secondary Leak Rate		150 gpd/SG	0.25 gpm to each steam generator	New AST value utilizes the accident induced leakage rate from the proposed revision to TS 5.5.7.b.2. The CLB values represent the operational leakage from TS 3.4.13. This issue is being managed via D.C. Cook's corrective actions program.
Density Used for Leakage Volume-o-Mass Conversion		62.4 lbm/ft3	62.3 lbm/ft3	New AST value is consistent with the current reactor coolant leak rate monitoring program. Difference is negligible.
Secondary Steam Release		Fraction of 17.2E6 lbm/hr 0 - 2 min: 0.35 2 - 10 min: 0.10 10 - 120 min: 0.05 120 - 480 min: 0.03	0 - 2 hours: 460,000 lbm 2 - 8 hours: 1,256,000 lbm 8 - 24 hours: 1,347,000 lbm	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Time to Cool RCS to 212F		8 hours	24 hours	Justification for termination of steam releases at 8 hours not adequate in CLB analyses. This issue is being managed via D.C. Cook's corrective actions program.
Duration of SG Tube Uncovery Following Reactor Trip		Not Modeled	40 minutes	Tube uncovery was not explicitly modeled in previous analysis. New AST value conservatively derived from plant simulator data.
Tube Leakage Flashing Fraction During Uncovery		Not Modeled	0-60 seconds: 16% 60- 300 seconds: 6% 300-1200 seconds: 5% 1200 seconds-40 min: 4%	See above.
Steam Generator Iodine Partition Coefficient		Unflashed Leakage - 100 Flashed Leakage - 0	Unflashed Leakage - 100 Flashed Leakage - 0	
Steam Generator Moisture Carryover Fraction		Not Modeled	0.2% (Particulate Partition Coefficient = 500)	New AST value conservatively biased high using plant-specific values.

Table 8: WGDT Inputs and Assumptions

Input/Assumption	CLB Offsite Value*	CLB CRH AST Value*	New Analytical Value for Offsite and AST CRH	Reason for Change
RCS Mass	Not Modeled	Not Modeled	275,460,950 gm	New analytical value is a conservatively high RCS mass.
Tank Volume	Not Modeled	Not Modeled	500 ft3	Input value is required for new analysis and was conservatively chosen but is immaterial for analytical conclusion.
Tank Release Rate	Not Modeled	Not Modeled	1,000,000 cfm	New value is conservatively high to simulate an instantaneous release.

*CLB offsite and CRH dose consequence analyses for WGDT rupture used an alternate method, as described in Enclosure 9, section 3.7.

Table 9: VCT Inputs and Assumptions

Input/Assumption	CLB Offsite Value	CLB CRH AST Value	New Analytical Value for Offsite and AST CRH	Reason for Change
RCS Mass	Not Modeled	241,000,000 gm	275,460,950 gm	New analytical value is a conservatively high RCS mass.
Tank Volume Liquid Volume	120 ft3	267ft3	267 ft3	CLB Offsite value utilized out of date information.
Tank Volume Vapor Volume	Not Modeled	133.1 ft3	500 ft3	New input value is required for new analysis and was conservatively chosen but is immaterial for analytical conclusion.
VCT Release Rate	Not Modeled	Not Modeled	1,000,000 cfm	New value is conservatively high to simulate an instantaneous release.
Letdown Flow Rate	75 gpm	120 gpm	132 gpm	New AST analysis uses conservatively high value from the CNP UFSAR.
Letdown Isolation Time	15 minutes	15 minutes	15 minutes	

Enclosure 13 to AEP-NRC-2014-65

D. C. Cook AST Accident Analyses Meteorological Data

(The enclosed compact disc contains the meteorological data used to develop the new on-site and off-site atmospheric dispersion (*X/Q*) factors)