November 14, 2002

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB5318 AND MB5319)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 271 to Facility Operating License No. DPR-58 and Amendment No. 252 to Facility Operating License No. DPR-74 for the Donald C. Cook (D. C. Cook) Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated June 12, 2000, as supplemented by letters dated November 7, 2000, June 19 and August 17, 2001, January 15, June 5, September 20, and November 13, 2002.

The amendments replace the current accident source term used in design-basis radiological analyses for control room habitability with an alternative source term (AST) pursuant to Title 10 of the *Code of Federal Regulations* (CFR) Section 50.67, "Accident Source Term." The licensee for D. C. Cook, Units 1 and 2, Indiana Michigan Power Company has requested a selective implementation of the AST limited to control room habitability assessments. The licensee has elected to use the AST and its associated acceptance criteria in preparing a revised control room dose analysis to show compliance with 10 CFR Part 50, Appendix A Criterion 19 "Control Room."

In addition, the proposed amendments revise the TSs to change the standard by which charcoal used in engineered safeguard features systems is tested. The proposed changes to the TSs are made in accordance with Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear-grade Activated Charcoal." The amendments also revise the format of the TS pages to adopt the format of Technical Specification Task Force (TSTF) Document TSTF -287 "Ventilation System Envelope Outage Time."

By letter dated March 29, 2001, the Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI) to provide the NRC staff with adequate information to make an independent assessment of the proposed license amendment. By letters dated June 19 and August 17, 2001, the licensee responded to the RAI. The NRC staff reviewed the licensee's June 19, 2001, response to the RAI and found that the additional information provided was not sufficient to resolve its concerns. The NRC issued a second RAI on August 16, 2001. The licensee responded to the RAI by letters dated January 15 and June 5, 2002. The licensee's response to the second RAI in the area of Reactor Coolant Pump (RCP) Lock Rotor analysis did not resolve the staff's concerns. The NRC staff visited the licensee's engineering offices on

August 13, 2002, to review information concerning the RCP Lock Rotor analysis. By letter dated September 20, 2002, the licensee documented the information reviewed at the site by the NRC staff.

By letter dated October 24, 2001, the NRC issued Amendments Nos. 257 for Unit 1 and No. 240 for Unit 2. The amendments approved that portion of the original June 12, 2000, proposed license amendment which dealt with GL 99-02. In addition, by letter dated November 13, 2001, the NRC issued Amendment Nos. 258 for Unit 1, and No. 241 for Unit 2, which approved that portion of the June 12, 2000, application which requested use of the AST associated with a fuel-handling accident.

This amendment completes the evaluation of the remainder of the proposed changes from the original June 20, 2000, application, with the exception of the changes associated with the high-efficiency particulate air (HEPA) filter/charcoal adsorber differential pressure. The licensee's September 20, 2002, letter, requested that the changes associated with the HEPA filter/charcoal adsorber differential pressure be issued at a later date under a separate cover.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

***/RA/***

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 271 to DPR-58
2. Amendment No. 252 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

August 13, 2002, to review information concerning the RCP Lock Rotor analysis. By letter dated September 20, 2002, the licensee documented the information reviewed at the site by the NRC staff.

By letter dated October 24, 2001, the NRC issued Amendments Nos. 257 for Unit 1 and No. 240 for Unit 2. The amendments approved that portion of the original June 12, 2000, proposed license amendment which dealt with GL 99-02. In addition, by letter dated November 13, 2001, the NRC issued Amendment Nos. 258 for Unit 1, and No. 241 for Unit 2, which approved that portion of the June 12, 2000, application which requested use of the AST associated with a fuel-handling accident.

This amendment completes the evaluation of the remainder of the proposed changes from the original June 20, 2000, application, with the exception of the changes associated with the high-efficiency particulate air (HEPA) filter/charcoal adsorber differential pressure. The licensee's September 20, 2002, letter, requested that the changes associated with the HEPA filter/charcoal adsorber differential pressure be issued at a later date under a separate cover.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

***/RA/***

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures: 1. Amendment No. 271 to DPR-58
2. Amendment No. 252 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

| | | | | |
|---|---|---|---|---|
| PUBLIC | GHill(2) | WBeckner | MShuaibi | THarris |
| PD 3-1 r/f | OGC | JRavel | SLaVie | |
| AVegel, RIII | ACRS | HWalker | JStang | |

**See previous concurrence
*Provide SE input by memo

ADAMS Accession Number: ML022980619

| OFFICE | PM:PD3-1 | LA:PD3-1 | SC:SPSB* | SC:SPLB* | SC:SRXB** | OGC/NLO** | SC:PDIII-1 |
|---|---|---|---|---|---|---|---|
| NAME | JStang | THarris | MReinhart | EWeiss | FAkstulewicz | BSmith | KMcConnell for LRaghavan |
| DATE | 11/14/02 | 11/15/02 | 09/23/02 | 10/02/02 | 11/12/02 | 11/06/02 | 11/14/02 |

OFFICIAL RECORD COPY

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL  60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI  48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI  49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI  49127

David W. Jenkins, Esquire
Indiana Michigan Power Company
One Cook Place
Bridgman, MI  49106

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI  49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI  48909

Drinking Water and Radiological
Project Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr. Blvd.
P. O. Box 30630, CPH Mailroom
Lansing, MI  48909-8130

Scot A. Greenlee
Director, Nuclear Technical Services
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI  49107

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC  20036-1495

Michael J. Finissi
Plant Manager
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI  49106

Joseph E. Pollock
Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI  49106

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 271
License No. DPR-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated June 12, 2000, as supplemented November 7, 2000, June 19 and August 17, 2001, January 15, June 5, September 20, and November 13, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 271 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA by Keith McConnell for/***

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 14, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 271

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
|---|---|
| VIII | VIII |
| XII | XII |
| 3/4 7-19 | 3/4 7-19 |
| 3/4 7-20 | 3/4 7-20 |
| 3/4 7-22 | 3/4 7-22 |
| | 3/4 7-22a |
| B 3/4 7-5 | B 3/4 7-5 |
| B 3/4 7-5a | B 3/4 7-5a |

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 252
License No. DPR-74

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated June 12, 2000, as supplemented November 7, 2000, June 19 and August 17, 2001, January 15, June 5, September 20, and November 13,2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 252 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA by Keith McMonnell for/***

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 14, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 252

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
| --- | --- |
| VIII | VIII |
| 3/4 7-14 | 3/4 7-14 |
| 3/4 7-15 | 3/4 7-15 |
| 3/4 7-16a | 3/4 7-16a |
| | 3/4 7-16b |
| B 3/4 7-4a | B 3/4 7-4a |
| | B 3/4 7-4b |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 271 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 252 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

## 1.0 INTRODUCTION

By application dated June 12, 2000, as supplemented November 7, 2000, June 19 and August 17, 2001, January 15, June 5, September 20, and November 13, 2002, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook (D. C. Cook) Nuclear Plant, Units 1 and 2. The proposed amendments request approval for the use of the methodology and the alternative source term (AST) in Title 10 of the Code of Federal Regulations (CFR) Section 50.67 “Accident Source Term,” to show compliance with 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 “Control Room,” of 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE).

In addition, the proposed amendments revise the TSs to change the standard by which the charcoal used in engineered safeguard features systems is tested. These proposed TSs changes are made in accordance with Generic Letter (GL) 99-02, “Laboratory Testing of Nuclear-Grade Activated Charcoal.” The amendments also revise the format of the TS pages to adopt the format of Technical Specification Task Force (TSTF) Document TSTF 287 “Ventilation System Envelope Outage Time.”

The supplemental letters provided by the licensee contained clarifying information and did not change the initial no significant hazards consideration and did not expand the scope of the original *Federal Register* notice.

## 2.0 BACKGROUND

The licensee implemented a design-basis reconstitution program in 1998. As a result of these efforts, the licensee discovered that the design and licensing basis documentation supporting control room habitability following a design-basis accident did not support compliance with GDC 19. The licensee determined that it would be necessary to reconstitute the control room habitability calculations. The licensee chose to use the revised analysis assumptions, methodology, and an acceptance criterion contained in 10 CFR 50.67 for the new control room habitability calculation to show compliance with GDC 19. In accordance with 10 CFR 50.67, the licensee in the proposed amendment requested prior approval of the AST.

By letter dated March 29, 2001, the Nuclear Regulatory Commission (NRC) issued a request for additional information (RAI). By letters dated June 19 and August 17, 2001, the licensee responded to the RAI. The NRC staff reviewed the licensee's June 19, 2001, response to the RAI, and found that some of the additional information provided was not adequate. The NRC again issued an RAI on August 16, 2001. The licensee responded to the RAI by letters dated January 15 and June 5, 2002. The licensee's response to the second RAI in the area of Reactor Coolant Pump (RCP) Lock Rotor analysis did not provide adequate information. The NRC staff visited the the licensee's engineering offices on August 13, 2002, to review information concerning the RCP Lock Rotor analysis. By letter dated September 20, 2002, the licensee documented the information reviewed at the site by the NRC staff.

By letter dated October 24, 2001, the NRC issued Amendment Nos. 257 for Unit 1, and No. 240 for Unit 2, approving the proposed TSs dealing with the GL 99-02 portion of the June 12, 2000, proposed license amendment. In addition, by letter dated November 13, 2001, the NRC issued Amendment Nos. 258 for Unit 1 and No. 241 for Unit 2, approving the use of the AST associated with a fuel-handling accident portion of the June 12, 2000, application.

This amendment completes the NRC staff evaluation of the remainder of the proposed changes from the original June 20, 2000, application, with the exception of the changes associated with the high-efficiency particulate air (HEPA) filter/charcoal absorber differential pressure. The licensee's September 20, 2002, letter, requested that the changes associated with the HEPA filter/charcoal absorber differential pressure be issued at a later date under a separate cover.

## 3.0 EVALUATION

### 3.1 Alternate Source Term 10 CFR 50.67

The proposed license amendment will replace the current accident source term used in design-basis radiological analyses for control room habitability with an AST pursuant to 10 CFR 50.67, "Accident Source Term." The licensee has requested a selective implementation of the AST limited to control room habitability assessments. The licensee has elected to use the AST, and its associated acceptance criteria, in preparing a revised control room dose analysis. Specifically, this amendment application requests the following:

> In December 1999, the NRC issued a new regulation, 10 CFR 50.67, which provided a mechanism for licensed power reactors to replace the traditional accident source term used in their design-basis accident analyses with voluntarily ASTs. Regulatory guidance for the implementation of these ASTs is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors." 10 CFR 50.67 requires a licensee seeking to use an alternative source term to apply for a license amendment and requires that the application contain an evaluation of the consequences of design-basis accidents. The present amendment request addresses these requirements in proposing selectively to use an AST in evaluating the post-accident habitability of the D. C. Cook main control rooms. The licensee has stated their intent to provide a separate request addressing the offsite consequences of design-basis accidents.

By letter dated October 28, 1998, the staff requested the licensee to submit revised analyses showing conformance of the control room ventilation systems with GDC-19. The licensee has submitted these re-analyses as a part of the June 20, 2000, amendment request.

The licensee evaluated the habitability of the control room for the following design-basis events:

- Large Break Loss-of-Coolant Accident (LBLOCA)
- Small Break Loss-of-Coolant Accident (SBLOCA)
- Steam Generator Tube Rupture (SGTR)
- Locked Rotor Accident (LRA)
- Rod Ejection Accident (REA)
- Main Steamline Break (MSLB)
- Gas Decay Tank (GDT) Rupture
- Volume Control Tank (VCT) Rupture
- Loss-of-Offsite Power (LOOP)

These re-analyses involved several changes in selected analysis assumptions including revised values for atmospheric dispersion values for the control room outside air intakes. The licensee used information contained in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” and Draft Regulatory Guide 1081, “Alternative Radiological Source Terms for Evaluating the Radiological Consequences of Design-Basis Accidents at Boiling and Pressurized Water Reactors,” in performing these evaluations. The bounding analysis inputs and assumptions for either Unit 1 and Unit 2 at Cook Nuclear Plant (CNP) were selected, eliminating the need for separate set of analyses for each unit.

The licensee performed the radiological analyses supporting this amendment assuming a reactor power equal to 102 percent of 3588 Megawatt Thermal (MWt). This power level exceeds the current licensed reactor power for the CNP units. This is a conservative approach and is acceptable to the staff. However, the staff notes that this amendment does not provide authorization to operate at the higher reactor power. The licensee expects to pursue a power uprate amendment in a separate license amendment application.

The staff reviewed the changes proposed by the licensee as described in their application and the supplemental submittals. The staff performed confirmatory calculations for the spectrum of accidents analyzed by the licensee and did a confirmatory evaluation of the revised atmospheric dispersion parameters used in the control room assessments. The licensee did not submit analysis results for individuals located offsite at the exclusion area boundary or the low population zone. Since only the dose to the control room operators was assessed and reviewed, the CNP implementation of the AST is considered a selective application, applicable only to control room habitability systems and analyses. The following sections of this safety evaluation report provide the results of the staff’s review of the licensee’s analyses. Table 1 tabulates the analysis inputs and assumptions found acceptable to the staff.

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 establishes minimum design, fabrication, construction, testing, and performance requirements for structures, systems, and components, such that the facility can be operated without undue risk to the health and safety of the public. GDC 19, contains the requirements for a control room in which actions are taken to operate the nuclear power unit safely under normal conditions, and to maintain the plant in a safe condition under accident conditions. For facilities implementing an AST, the control room is required to have adequate radiation protection to permit access and occupancy under accident conditions without personnel receiving radiation exposures greater than 5 rem TEDE. This dose criterion is also stated in 10 CFR 50.67. The focus of the control room dose criteria in 10 CFR 50.67 and GDC-19 is providing for continued ability of the control room operator to take required actions to protect the public.

In its application, the licensee proposed to use the AST described in NUREG-1465 and Draft Regulatory Guide 1081. The staff cannot accept NUREG-1465 as a part of the CNP licensing basis since the document contains information that the staff has not found to be acceptable as regulatory guidance for operating reactors. The staff developed guidance for using ASTs that incorporates and expands upon those aspects of NUREG-1465 deemed acceptable for operating reactors. The guidance was published for public comment in December 1999, as DG-1081 and issued as final guidance as Regulatory Guide (RG) 1.183 in July 2000. Some of the DG-1081 assumptions used by the licensee in their analyses were modified in the final RG 1.183. The staff asked the licensee to provide a statement of their intent to use the guidance of RG 1.183 in future AST analyses at CNP, and to identify and justify the use of any alternatives used in analyses supporting this application. In their responses, the licensee addressed the differences between the assumptions identified in the application and those in RG 1.183. The licensee provided sufficient information for the staff to accept some of the differences as acceptable alternatives. For the remaining differences, the licensee provided a commitment in the January 15, 2002, letter, to use the applicable RG 1.183 assumptions in future re-analyses. The staff performed its confirmatory analyses using the accepted assumptions. In all these cases, there was a sufficient margin in the analyses results to absorb the difference in the assumptions. The staff's approval of this amendment is based on the information docketed by the licensee and on the staff's finding that the methods, inputs, and assumptions used in the licensee's analyses as modified by this commitment are acceptable. The staff determined there was reasonable assurance that, in the event of a Design-Basis Accident (DBA), the radiological consequences would continue to comply with applicable acceptance criteria. Table 1 on page 19 of this safety evaluation (SE) tabulates the assumptions found acceptable to the staff.

### 3.1.1 Control Room Modeling

D. C Cook has a separate control room for each unit. Each control room has its own redundant ventilation systems. Each control room has two outside air intakes (OAI). One OAI at each unit supplies both of the air handlers for that unit. The other OAI at each unit is used during emergency conditions for pressurization. Motor operated dampers isolate all four outside air intakes. The CNP control rooms operate in the zone isolation with filtered pressurization mode. Since the operation of the control rooms is identical and since the analyses performed in support of this amendment were based on the limiting parameters for either unit, radiological consequences only need to be assessed for one control room. The operation of the control room ventilation systems is modeled in three phases:

- Before the event, the control room ventilation system works in an unfiltered recirculation mode with a flow rate of 13,400 cubic feet per minute (cfm). Unfiltered outside air is admitted at 1000 cfm as normal makeup and to maintain a slight positive pressure during normal operations.

- On receipt of the isolation signal, the normal OAI shuts, ending the normal makeup flow. The emergency ventilation OAI opens supplying 2000 cfm of air to the emergency ventilation filter bank. The emergency ventilation fans also drive 8800 cfm of recirculation flow through the filter bank, for a total filter flow of 10,800 cfm. At this flow rate, the licensee assumes the filter efficiency to be 80 percent for elemental and organic species and 98 percent for particulates. The licensee assumes a delay of 66 seconds from the time of safety injection signal actuation, which varies from accident to accident. Although control room isolation can be actuated by a radiation monitor alarm, this actuation was not credited in these analyses since the monitor arrangement does not meet single-failure requirements.

  There are two 100 percent flow pressurization fans - each on a separate emergency power bus. Although application of single-failure concepts would normally require that one fan be assumed to fail, the more limiting case would be for the two fans to run simultaneously. Also during this period, it is assumed that 98 cfm of unfiltered inleakage enters the control room. This value was determined by tracer gas leakage testing.

- At two hours following actuation of isolation, it is assumed that operators would manually stop the redundant fan, reducing the flow through the filter to 5400 cfm - 1000 cfm as outside air makeup and 4,400 cfm as recirculation. At the lower design flow rates, the licensee assumes the filter efficiency to be 95 percent for elemental and organic species and 98 percent for particulates. The licensee's assumed that a two-hour delay in stopping the second emergency filtration fan is considered conservative. This configuration is maintained for the duration of the accident.

### 3.1.2 Analyzed Accidents

#### 3.1.2.1 Large-Break Loss-of-Coolant Accident (LBLOCA) Radiological Consequences

A loss-of-coolant accident (LOCA) is a failure of the reactor coolant system that results in the loss of reactor coolant and, if not mitigated, fuel damage possibly including a core melt. Analyses are performed using a spectrum of break sizes to evaluate fuel and emergency core cooling system (ECCS) performance. A LBLOCA is postulated as the failure of the largest pipe in the reactor coolant system (RCS). RG 1.183 establishes the LBLOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. The containment building holds up the majority of the radioactivity released from the core. Evaluation of the effectiveness of plant safety features, such as ECCS, has shown that core melt is unlikely. The objective of this DBA is to evaluate the ability of the plant design to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective. Fission products from the damaged fuel are released into RCS and then into the containment (CNMT). With the LBLOCA, it is anticipated that the initial fission product released to the CNMT will last 30 seconds and will release all of the radioactive materials dissolved or suspended in the RCS liquid. The gap inventory release phase begins with the onset of fuel cladding failure and is assumed to continue for 30 minutes. As the core continues to degrade, the gap inventory release phase ends and the in-vessel release phase will commence. This phase continues for 1.3 hours. Tables 2, 4, and 5 of RG 1.183 define the source term used for these two phases. This data is summarized in Table 3.1.2.1-1 below.

The inventory in each release phase is assumed to be released at a constant rate over the duration of the phase and starting at the onset of the phase.

During the progression of an LBLOCA, some fission products released from the fuel will be carried to the CNMT sump via spillage from the RCS or by the CNMT spray system and natural processes such as deposition and plateout. For the purposes of assessing the consequences of leakage from the ECCS, the licensee assumes that all of the radioiodines released from the fuel are instantaneously moved to the CNMT sump. Noble gases are assumed to remain in the CNMT atmosphere. The remaining radionuclides in Table 3.1.2.1-1 are aerosols or particulates. Since radionuclides of this chemical form will not become airborne on release from the ECCS, they are not included in the ECCS source term. This source term assumption is conservative in that all of the radioiodines released from the fuel are credited in both the CNMT atmosphere and CNMT sump. In a mechanistic treatment, the relocation of the radioiodines in the CNMT atmosphere would occur over time and deplete the radioiodine concentration in the CNMT atmosphere.

**Table 3.1.2.1-1 Release Fractions as a Function of Release Period**

| **RADIONUCLIDE GROUP** | **GAP RELEASE** *(0.5 Hours)* | **EARLY IN-VESSEL** *(1.3 Hours)* |
|---|---|---|
| Noble Gases (Xe, Kr) | 0.05 | 0.95 |
| Halogens (I, Br) | 0.05 | 0.35 |
| Alkaline Metals (Cs, Rb) | 0.05 | 0.25 |
| Tellurium Group (Te, Sb, Se) | 0 | 0.05 |
| Ba, Sr | 0 | 0.02 |
| Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) | 0 | 0.0025 |
| Cerium Group (Ce, Pu, Np) | 0 | 0.0005 |
| Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) | 0 | 0.0002 |

Once dispersed in the CNMT, release to the environment is assumed to occur through the following pathways:

- Release of CNMT atmosphere through purge system before isolation.
- Leakage of CNMT atmosphere (i.e., design leakage).
- Leakage from the ECCS that recirculate CNMT sump water outside CNMT (i.e., design leakage).

Fission products are released from the core into the lower CNMT and transported through the ice condensers into the upper CNMT. Ventilation fans recirculate the fission products between the upper and lower CNMTs. The ice condenser and the CNMT sprays scavenge fission products from the atmosphere and transport them to the CNMT sump. The thermal gradients caused by the action of the ice condenser and the CNMT sprays, and the forced air flow associated with the recirculation fans result in thorough mixing of the CNMT atmosphere. The recirculation fans will start automatically on a high pressure signal and are assumed to be operating after three minutes. The sprays start automatically on high pressure and are assumed to inject water after 126 seconds. This time delay includes an emergency diesel generator sequencing time. When the source of water to the sprays drains to a predetermined set point level, the operators manually realign the system to draw on the CNMT sump (i.e., recirculation mode). Based on evaluation of the drain down times associated with various system operating configurations, the licensee assumes that the switchover process will start at 1.25 hours. During the five minute period it takes to effect the switchover, no credit is taken for spray removal. The analysis assumes that the recirculation spray is effective at 1.33 hours. When the particulate iodine inventory in the CNMT atmosphere is reduced to 2 percent of its original value (i.e., a Decontamination Factor (DF) of 50) the removal coefficients for particulates are reduced by a factor of 10 following standard review plan (SRP) guidance. This reduction occurs at 2.41 hours. The sprays continue to remove particulates at the reduced rate until six hours. When the elemental iodine inventory in the CNMT atmosphere is reduced to 0.5 percent of its original value (i.e., a DF of 200) credit for the removal of elemental iodine is halted following SRP guidance. This occurs at 2.78 hours. The model used by the licensee to determine the spray removal coefficients is that of the SRP Chapter 6.5.2. The staff reviewed the modeling and methodology used by the licensee and found the license treatment to be consistent with that guidance.

Elemental iodine removal by the ice condensers was credited with a removal efficiency of 0.3 per pass. This removal is credited between three minutes and one hour following the event. The three minute delay is based on the delayed starting of the CNMT recirculation fans. The credit for removal ceases when the first ice bed melts-out, projected to occur after one hour. The maximum DF is not reached in this period.

The analysis assumes that the iodine released to the containment includes 95 percent CsI, 4.85 percent elemental iodine, and 0.15 percent organic forms. The assumption of this iodine speciation is predicated on maintaining the CNMT sump water at pH 7.0 or higher. Existing analyses reported in the CNP updated final safety analysis report (UFSAR) show that the final pH at Unit 1 will be 9.3 and the pH at Unit 2 will be between 7.6 and 9.5. These values consider melted ice. Sump pH is controlled by means of Sodium Hydroxide addition.

The licensee conservatively assumes that the CNMT purge dampers are open at the start of the event and are not shut for 10 seconds, creating a pathway for the release of the CNMT atmosphere to the environment. The exhaust flow rate for the purge system is 16,000 cfm and the dampers automatically close in about five seconds (10 seconds assumed in analyses). During this release, the fission product inventory in the CNMT atmosphere is limited to that released from the normal RCS inventory. The licensee assumed that the radioiodine concentration of the RCS was equal to the TS 1.0 µCi/gm value, while the noble gas concentrations were based on 1.0 percent failed fuel.

The CNMT is projected to leak at its design leakage of 0.25 percent of its contents by weight per day for the first 280 hours and then at 0.125 percent for the remainder of the 30-day accident duration.  This assumption differs from staff guidance in RG 1.183 that provides for this reduction at 24 hours.  Site-specific analyses at CNP show that the CNMT pressure at 280 hours will be 25 percent of the design value.  Because of this, the licensee conservatively delays the reduction in leakage until 280 hours.  This alternative to the RG 1.183 guidance is acceptable.  This CNMT leakage is not collected and enters the environment without processing.

The ECCS recirculates the contaminated sump water outside the CNMT where small amounts of system leakage could provide a path for the release of fission products to the environment.  The analysis considers the equivalent of 0.2 gallons per minute (gpm) unfiltered ECCS leakage starting at the onset of the LOCA.  Site administrative controls limits the leakage to 0.1 gpm.  These limits are described in the UFSAR.  The licensee assumes the iodine species in the recirculation water to be 100 percent elemental.  This assumption differs from staff guidance in RG 1.183 that provides 97 percent elemental and 3 percent organic.  However, since the licensee does not credit filtration before release and since the efficiency of the control room ventilation system filter is identical for elemental and organic forms, there can be no impact on the postulated doses.  This alternative to the staff guidance is acceptable.

In the June 12, 2000, submittal, the licensee assumed that only 0.01 percent of the iodine in the ECCS leakage becomes airborne.  This assumption differs from staff guidance in RG 1.183 that provides for a minimum flash fraction of 10 percent.  The licensee stated in the submittal that this value was consistent with the value currently in their licensing basis that had been experimentally established at the time of plant licensing.  While the staff does not question the conclusions of the experiment, the staff does question the applicability of the experiment results (obtained in a laboratory-like setting) to the in-plant conditions in which the leakage occurs.  The staff notes the mass transfer of entrained iodine from the water droplets falling through air or being ejected under pressure and perhaps impinging on nearby surfaces will be different from that determined in the experiment.  In response to the staff concerns, the licensee reperformed the LBLOCA analysis using acceptable assumptions and reported the changes in the June 5, 2002, supplemental letter.

Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE.  The estimated control room dose for the postulated LBLOCA was found to be acceptable.

3.1.2.2  Small-Break Loss-of-Coolant Accident (SBLOCA) Radiological Consequences

A LOCA is a failure of the RCS that results in the loss of reactor coolant and, if not mitigated, fuel damage possibly including a core melt.  Analyses are performed using a  spectrum of break sizes to evaluate fuel and ECCS performance.  RG 1.183 establishes the LBLOCA as the licensing basis LOCA with regards to radiological consequences since this represents the larger challenge to plant safety features designed to mitigate the release of radionuclides to the environment in the unlikely event that ECCS is not effective.  The SBLOCA is not currently part of the licensing basis for CNP with regard to radiological consequences.  The licensee had the SBLOCA analysis performed to address a concern regarding the availability of CNMT sprays, for iodine removal during a SBLOCA.  Although CNMT spray actuation is expected at ice condenser plants during a SBLOCA, there is a possibility that control room operators could stop CNMT spray for SBLOCA events after it had started.

The licensee's analysis assumes that the break is small enough not to trigger CNMT sprays, but large enough to cause fuel cladding damage that releases 100 percent of the fuel gap inventory. The licensee asserted that thermo-hydraulic analyses demonstrate that the clad temperature does not exceed 2200 °F and that the fuel pellet temperature would not be significantly greater than the clad temperature. The staff notes that the 2200 °F threshold is a criterion in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," which applies to the entire spectrum of breaks, including the LBLOCA. Nonetheless, the staff believes that the assumption of substantial fuel melting, required for the LBLOCA analysis (Footnote 1 to 10 CFR 50.67), is not required for the SBLOCA, since the LBLOCA is the licensing basis LOCA with regard to radiological consequences (RG 1.183). In this context, the licensee's assumption of 100 percent gap release is acceptable.

The licensee assumed that 5 percent of the core inventories of noble gases, iodines, and alkali metals are present in the fuel rod gap and that 3 percent would be released between 30 seconds and 90 seconds and the remaining 2 percent released over the next 28.5 minutes. RG 1.183 provides for a linear release ramp from 30 seconds to 30 minutes or for a step increase to 5 percent at 30 seconds. The licensee's assumption is acceptable since it falls between these two extremes. The licensee assumes that the RCS inventory is at 60 µCi/gm dose equivalent I-131 at the start of the event.

The release to the environment is assumed to occur through two pathways:

- Release of CNMT atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through steam generators (SGs)

For the CNMT leakage case, all of the gap inventory released from the fuel gap is assumed to enter the CNMT with an iodine speciation of 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee took credit for the sedimentation removal of particulates. The sediment removal coefficient is conservatively assumed to be 0.1 $hr^{-1}$ with a DF limited to 1000. Sedimentation was credited for only the first 65 hours. The licensee based their assumptions on the containment systems experiments work. RG 1.183 cites NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," as an acceptable method for addressing natural deposition. Table 24 of that report provides time variant removal coefficients ranging from 0.0157 $hr^{-1}$ to 0.223 $hr^{-1}$ for 3000-4000 MWt plants. The licensee provided additional information regarding this assumption in its August 17, 2001, letter. The staff reconsidered this issue and has decided to accept the proposed values for CNP based upon the additional information provided by the licensee.

The CNMT is projected to leak at its design leakage of 0.25 percent of its contents by weight per day for the first 280 hours and then at 0.125 percent for the remainder of the 30-day accident duration. This assumption differs from staff guidance in RG 1.183 that provides for this reduction at 24 hours. Site-specific analyses at CNP show that the CNMT pressure at 280 hours will be 25 percent of the design leakage. Because of this, the licensee conservatively delays the reduction in leakage until 280 hours. This alternative to the staff guidance is acceptable. This leakage is not collected and enters the environment without mitigation.

For the secondary release pathway, all of the gap inventory released from the fuel gap is assumed to remain in the RCS and be available for leakage to the secondary plant and to the environment. The licensee assumes the iodine species in the secondary to be 100 percent elemental. This assumption differs from staff guidance in RG 1.183 that provides 97 percent elemental and 3 percent organic. However, since the licensee does not credit filtration before release and since the efficiency of the control room ventilation system filter are identical for elemental and organic forms, there can be no impact on the postulated doses. This alternative to the staff guidance is acceptable.

The primary-to-secondary leakage of 1.0 gpm and the SG steaming continue until the RCS pressure decreases to less than the SG pressures, thereby reversing the pressure differential and the flow direction. A bounding duration of two hours was assumed since the thermo-hydraulic analyses for an SBLOCA show the pressure drop occurs well before this time. While the staff accepts that the primary-to-secondary leakage will cease at this time, steam will continue to be released from the SGs until a cold shutdown is reached. Since the analysis assumes the holdup of iodine and alkali metals in the SG, a radioactivity release is possible even though the primary-to-secondary leakage has stopped. However, given the relative magnitude of the steaming rate in comparison to the primary-to-secondary leak rate, the majority of the release occurs early. The staff performed a confirmatory calculation with a 30-day steam release and found only a negligible increase in postulated doses. As such, the analysis approach is acceptable. An iodine partition factor of 0.01 was assumed in the SGs. This factor is also conservatively used for the alkali metal inventory in the SGs. The alkali metal retention is expected to be greater.

Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The estimated control room dose for the postulated SBLOCA was found to be acceptable. In the January 15, 2002, letter, the licensee stated that since the radiological consequences of a SBLOCA have been shown to be bounded by those of the LBLOCA, it does not intend to re-perform the SBLOCA analysis in the future. This is acceptable to the staff since the RG 1.183 establishes the LBLOCA as the licensing basis LOCA with regards to radiological consequences.

3.1.2.3 Steam Generator Tube Rupture (SGTR) Radiological Consequences

This DBA postulates a rupture in a tube in one of the four steam generators resulting in the transfer of reactor coolant water to the ruptured steam generator. The primary-to-secondary flow through the ruptured tube (“break flow”) following a SGTR results in a depressurization of the RCS, a reactor trip, and actuation of safety injection. Since a loss of offsite power is assumed to occur when the reactor trips, the main condenser is not available as a heat sink and contaminated steam is released to the environment from the ruptured SG. After safety injection actuates, it is assumed that the RCS pressure will stabilize at a value at which the safety injection and break flows are equal. The break flow is assumed to continue at this equilibrium value until plant operators have taken action to reduce RCS pressure. When RCS pressure is less than the SG pressure, the pressure differential and the flow reverses direction, terminating the break flow. The licensee assumes this occurs within 30 minutes from safety injection actuation. The unaffected SGs are used to cool down the plant by dumping steam to the environment. The released steam maybe contaminated due to leakage of reactor coolant into the SGs.

The licensee identified that the operators may not be able to end break flow within 30 minutes for all postulated SGTR events. The licensee postulates that RCS mass transferred to the secondary side is greater for the constant flow rate for 30 minutes case, than the case assuming time-variant break flows generated with a transient analysis program and actual operator response times. Although this transient analysis program has been approved for other facilities, The licensee opted not to use this program to quantify the amount of break flow since it would represent a change in analysis methodology for CNP. Instead, the licensee proposed retaining the original, more conservative analytical basis. The staff accepts the licensee's rationale in establishing the more conservative approach, as it applies to break flow. However, the staff questioned whether the same conclusion would be valid for other analysis parameters such as, total steam mass released, break flow flashing fraction, and SG liquid mass, and that the resulting doses might not be conservative. Based upon additional information provided by the licensee, the staff finds this approach to be acceptable. After 30 minutes, the licensee assumes that steam is released from the three unaffected SGs to cool the plant down to the conditions at which further cooling could be achieved using the residual heat removal system (RHR), thereby ending the release to the environment. The licensee conservatively assumes that this switchover to RHR will not occur for 30 days.

Break flows, steam releases, and feedwater flows were determined using thermal-hydraulic analyses to bound the operating conditions for the two CNP units. The analysis assumes that the noble gases entrained in the break flow are released to the environment without holdup or decontamination in the SGs. For the ruptured SG, the analysis assumes that part of the break flow, known as the flash fraction, will immediately flash to steam and the entrained gases be released to the environment with no holdup in the SGs. The portion of the break flow that does not flash is assumed to mix with the bulk water of the SGs and be released at the steaming rate of the SGs. The iodine release rates are reduced to account for partitioning between the liquid and vapor phases. The licensee did not obtain time-variant values for the flash fraction but, instead, determined a single bounding value. The licensee identified two conditions: pre-trip and post-trip. Since the flash fraction decreases over time, the licensee assumed a flash fraction associated with the RCS (hot leg) and ruptured SG pressure and temperature conditions at the time of the reactor trip. While the flash fraction would be greater before the trip, the associated releases would be via the main condenser which would afford some iodine mitigation. The limiting radiological release conditions would occur just following the reactor trip.

The licensee assumes the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by TSs. The initial noble gas inventory in the RCS is based on fuel damage equivalent to 1.0 percent failed fuel. Two iodine spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and that the RCS iodine inventory is at 60 µCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. In this case, iodine is released from the fuel to the RCS at a rate 500 times the normal iodine appearance rate. This multiplier is more conservative than the 335 times multiplier specified by RG 1.183 and is acceptable. The licensee determined the iodine appearance rates assuming a letdown system flow rate of 120 gpm, radioactive decay, and a primary system leakage of 12 gpm. The staff believes that the licensee's appearance rates are acceptably conservative, in that plant operation at these maximum conditions is typically infrequent and of short duration. The licensee states that at the iodine release rate is such that the iodine inventory of the fuel rod gap will be depleted by six hours. In response to staff questions, the licensee provided additional information on this assumption in their January 15, 2002, letter. The licensee assumes that the spike duration is a function of the accident-induced iodine appearance rate (188.3 Ci/min) and the iodine available for release from the fuel gap of fuel rods with defects (60,588 Ci -- based upon 0.495 percent fuel defects). This evaluation yields a release duration of 5.36 hours. Since the normal appearance rate corresponds to 0.26 percent fuel defects, a fuel gap inventory based upon 0.495 percent fuel defects is conservative. The staff finds the licensee's justification of the 6-hour spike duration acceptable.

Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The estimated control room dose for the postulated SGTR was found to be acceptable.

#### 3.1.2.4 Main Steamline Break (MSLB) Radiological Consequences

This DBA postulates a failure, which can not be isolated, of one of the four main steamlines at a location outside of containment, resulting in the release of steam from the affected steamline. The faulted SG will rapidly depressurize and release its entire liquid inventory and dissolved radioiodines through the faulted steamline to the environment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Since a loss of offsite power is assumed to occur with the reactor trip, the main condenser is not available as a heat sink and the unaffected SGs are used to cool down the plant by dumping steam to the environment. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The radiological consequences of a break outside CNMT will bound those results from a break inside CNMT. Thus, only the break outside CNMT is considered with regard to control room dose.

The licensee assumes that the faulted SG boils dry in two minutes. The radionuclides transferred from the RCS via the primary-to-secondary leakage of 500 gallons per day (gpd) are released to the environment without holdup in the faulted SG over the 30-day duration of the accident. The licensee assumes that primary-to-secondary leakage transfers 940 gpd (total) to the three unaffected SGs used for cooldown. This leakage mixes with the bulk SG water. Transferred noble gases are released without a holdup. Iodine is released to the environment at the steaming rate of the SGs with credit for iodine partitioning. The steam releases continue for 30 days. The licensee's assumptions regarding the RCS inventory and iodine spiking are the same as those discussed above for the SGTR. No fuel damage is projected for the MSLB.

Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The estimated control room dose for the postulated MSLB was found to be acceptable.

#### 3.1.2.5 Loss of Offsite Power (LOOP) Radiological Consequences

In the event of a loss of offsite AC power, emergency diesel generators will start and power vital loads. Major plant loads that would be lost include reactor coolant pumps, main feedwater pumps, main circulating water system, turbine auxiliaries, and the main condenser. A reactor trip will occur. Without circulating water, the main condenser will not be available to receive and condense steam generated during the plant cooldown. As such, this steam is released to the environment by steam safety valves and atmospheric dump valves. This steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage).

The licensee assumes that primary-to-secondary leakage transfers 1.0 gpm (total) to the four SGs. This leakage mixes with the bulk SG water. Transferred noble gases are released without a holdup. Iodine releases are released to the environment at the steaming rate of the SGs with credit for iodine partitioning. The steam releases continue for 30 days. The licensee's assumptions regarding the RCS inventory and iodine spiking are the same as those discussed above for the SGTR. The loss of offsite power event is not expected to result in a safety injection signal. As such, the licensee assumes no isolation of the control room. The analysis for this accident assumes that the control room remains in its normal alignment (1000 cfm unfiltered makeup flow) for the duration of the event.

Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The estimated control room dose for the postulated LOOP was found to be acceptable.

3.1.2.6 Locked Rotor Accident (LRA) Radiological Consequences

For this DBA, a reactor coolant pump rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. The radiological consequences are due to leakage of the contaminated reactor coolant to the SG and from there, the environment. A LOOP is conservatively assumed to occur when the reactor trips, rendering the main condenser unavailable. With the main condenser unavailable, the plant is cooled down by releases of steam to the environment.

The licensee stated that fuel damage is not expected for a locked rotor accident at either CNP Unit. This assumption differed from that documented in the UFSAR, which states that less than 7 percent of the fuel rods at Unit 1 will not meet the departure for nucleate boiling ratio (DNBR) criteria, and that less than 11 percent of the fuel rods at Unit 2 do likewise. The staff requested additional justification for the proposed non-fuel damage assumption. In the August 17, 2001, and June 5, 2002, letters, the licensee provided additional information. The staff traveled to the CNP site and performed an assessment of the licensee's analysis documentation.

D. C. Cook's licensing basis includes analyses for a locked rotor departure from a nucleate boiling (DNB) event to demonstrate that core coolability is maintained consistent with the requirements of 10 CFR Part 50, Appendix A, GDC 27 and 28 and to determine if any fuel rods are in DNB. A fuel rod is considered to be in DNB if the calculated departure from the nucleate boiling ratio (DNBR) for the rod is less than the established DNBR limit. If fuel rods are determined to be in DNB, radiological consequence analyses are performed to demonstrate compliance with 10 CFR Part 100 limits. The percentage of fuel rods in DNB is used in the radiological consequence analyses. The licensee for D. C. Cook has established an acceptance criteria of no rod in DNB, which, if met, would eliminate the need to perform the radiological consequence analyses for the locked rotor event.

The licensee performs DNBR analyses using the NRC-approved Revised Thermal Design Procedure described in WCAP-11397-P-A. The licensee utilizes the LOFTRAN, FACTRAN, and THINC computer codes to perform the locked rotor DNB analyses. The LOFTRAN code is used to calculate the loop and core flow during the transient, the time of reactor trip, the nuclear power transient, and the primary system pressure and temperature transient. The FACTRAN uses parameters for nuclear power and flow from the LOFTRAN code and calculates the heat flux transient. The THINC code uses parameters for heat flux from the FACTRAN code and flow from the LOFTRAN code and calculates the DNBR during the transient. The licensee does not perform cycle specific LOFTRAN and FACTRAN analyses. The licensee uses bounding LOFTRAN and FACTRAN analyses which are confirmed to remain bounding on a cycle specific basis. The THINC analyses are performed each cycle using cycle specific axial power shapes. This approach is implemented by the licensee because it provides a greater margin to DNBR than if bounding THINC analyses are used.

The staff reviewed the licensee's assumptions and inputs for the locked rotor analyses. The licensee's assumptions for initial conditions related to power, RCS flow rate, and average reactor coolant temperature are presented in UFSAR Table 14.1.0.2 for each unit. The staff reviewed these parameters and concluded that, when instrument uncertainty is considered, the parameters bound plant operation allowed by the TSs for each unit. In addition, the staff reviewed the licensee's assumptions related to the plant's pressure response during the transient. The LOFTRAN analyses calculate a pressure increase due to the heatup and consequent expansion of the RCS inventory into the pressurizer. However, the lower initial pressure is maintained in the THINC code calculation of DNBR. Holding the pressure constant at the lower initial value is acceptable for DNBR calculations because it results in lower (more limiting) DNBR values.

The licensee assumed that offsite power remains available for the duration of the locked rotor analyses. This allows for a greater RCS flow rate during the transient due to operation of the reactor coolant pumps not affected by the locked rotor. This also results in a higher (i.e., less limiting) calculated DNBR value. The licensee showed that this assumption is consistent with the original licensing basis of the plant. Therefore, the staff finds this assumption acceptable.

The licensee's analyses show that the minimum calculated DNBR for the thimble and typical cells are greater than the established DNBR limits and therefore no rods are in DNB. The licensee's calculations account for several DNBR margin allocations whose values were determined by plant-specific or generic sensitivity studies. The staff reviewed the results of the licensee's analyses and the margin allocations and determined they are acceptable.

Based on the above, the staff finds the licensee locked rotor DNB analyses acceptable.

Without postulated fuel damage, the radiological consequences of this event would be bounded by the postulated consequences of the LOOP accident. RG 1.183 states that, without fuel damage, the consequences of a LRA would be bounded by those for a MSLB outside CNMT. The licensee showed, by sensitivity analyses, that the consequences from a LOOP would be more limiting than those for a MSLB. The staff finds this position to be an acceptable alternative to the guidance in RG 1.183. In any case, no additional radiological analyses need to be done.

#### 3.1.2.7 Rod Ejection Accident (REA) Radiological Consequences

This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected. This failure breeches the reactor pressure vessel head resulting in a LOCA to the CNMT. A reactor trip will occur. From a radiological analysis standpoint, the REA is similar to an SBLOCA, but with increased fuel releases.

The release to the environment is assumed to occur through two pathways:

- Release of CNMT atmosphere (i.e., design leakage)

- Release of RCS inventory via primary-to-secondary leakage through SGs

The licensee assumed that 15 percent of the fuel rods suffer sufficient damage to result in the release of all of their gap inventory to the RCS or the CNMT. The licensee assumed that 12 percent of the I-131 inventory of the core was in the fuel rod gap, along with 15 percent of the Kr-85, and 10 percent of all other iodines, noble gases, and alkali metals. RG 1.183 provides for 10 percent of the core inventory of noble gases and iodines are in the fuel rod gap. A small fraction of the fuel in the failed rods is assumed to melt because of the accident. The licensee estimated this damage to be limited to 0.375 percent of the core. The licensee assumed that this fuel melting releases 35 percent of the iodine, 25 percent of the alkali metals and 95 percent of noble gases in the fraction of fuel melted to the RCS or the CNMT. These fuel melt release fractions are not consistent with the guidance of RG 1.183, which assumes 100 percent release of noble gases, 50 percent of iodines to the RCS, and 25 percent to the CNMT. (RG 1.183 retained the release fractions of RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," until such time as issues related to fuel damage in reactivity insertion accidents were resolved.) For the CNMT case, the licensee assumption is conservative. For the secondary release case, the assumed 35 percent release fraction is less than the 50 percent fraction provided in RG 1.183. The licensee has committed to correcting this assumption in future re-analyses. The licensee states, and the staff agrees, that the dose contribution from the secondary release is negligible in comparison with that for CNMT leakage. The staff determined there was reasonable assurance that, in the event of a DBA, the radiological consequences would continue to comply with applicable acceptance criteria and that revision of the analysis can be deferred. The licensee conservatively assumed that a pre-incident iodine spike occurred just before the event such that the RCS specific activity has a concentration of 60 µCi/gm dose equivalent I-131.

For the CNMT leakage case, the iodine released is 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee assumes the iodine species in the secondary to be 100 percent elemental. This assumption differs from staff guidance in RG 1.183 that provides 97 percent elemental and 3 percent organic. However, since the licensee does not credit filtration before release and since the efficiency of the control room ventilation system filters is identical for elemental and organic forms, there can be no impact on the postulated doses. This alternative to the staff guidance is acceptable.

The CNMT is projected to leak at its design leakage of 0.25 percent of its contents by weight per day for the first 280 hours and then at 0.125 percent for the remainder of the 30-day accident duration. This assumption differs from staff guidance in RG 1.183 that provides for this reduction at 24 hours. Site-specific analyses at CNP show that the CNMT pressure at 280 hours will be 25 percent of the design leakage. Because of this, the licensee conservatively delays the reduction in leakage until 280 hours. This alternative to the staff guidance is acceptable. This leakage is not collected and enters the environment without processing.

The primary-to-secondary leakage of 1 gpm and the SG steaming continue until the RCS pressure decrease to less than the SG pressures. (This assumption was also made in the analysis for the SGTR accident.) While the staff recognizes that the primary-to-secondary leakage will stop, the SG will continue to steam until a cold shutdown is reached. Since the analysis assumes the holdup of iodine and alkali metals in the SG, a radioactivity release could continue past the time that the primary-to-secondary leakage stops. However, on review, the transfer coefficient from the RCS to the secondary was small in comparison to the transfer coefficient. The staff did a confirmatory calculation with a 30-day steam release and found only a negligible increase in postulated doses.

For the CNMT leakage pathway, the licensee did not take credit for any iodine removal mechanisms. An iodine partition factor of 0.01 was assumed in the SGs. This factor is also conservatively used for the alkali metal inventory in the SGs. Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The estimated control room dose for the postulated REA was found to be acceptable.

3.1.2.8 Gas Decay Tank (GDT) and Volume Control Tank (VCT) Radiological Consequences

The CNP licensing basis includes analyses of the radiological consequences of a rupture of a GDT and a rupture of the VCT. A GDT is used to store processed radiogases removed from the reactor coolant to allow for radioactive decay before the controlled release to the environment. The VCT is a component in the plants' chemical and volume control systems that serves as a surge volume to balance differences in letdown and makeup flow rates while maintaining reactor coolant inventory. Part of the reactor coolant (known as letdown) is removed from the RCS, cooled, filtered, demineralized, and degassed. Purified letdown water is sprayed into the VCT, which is normally less than 25 percent full. Radiogases collect in the top of the VCT.

The GDT case assumes that the entire inventory of gases in the RCS based on continuous operation with 1.0 percent failed fuel is in the GDT and is released over a period of five minutes.

The initial inventory of noble gases in the VCT is based on continuous operation with 1.0 percent failed fuel, without purging of the VCT. The initial inventory of iodines is based on continuous operation with an RCS specific inventory of 60 µCi/gm dose equivalent I-131, with a 90 percent removal by the letdown demineralizers. The initial inventory is assumed to be released over a period of five minutes. After the event starts, letdown flow to the VCT is assumed to continue at the maximum flow rate of 120 gpm for 15 minutes, after which letdown is isolated. Iodines in the letdown flow are reduced by 90 percent by the letdown demineralizers.

The analysis for this accident assumes that the control room remains in its normal alignment (1000 cfm unfiltered makeup flow) for the duration of the event. Details on the assumptions found acceptable to the staff are presented in Table 1 on page 19 of this SE. The resulting control room dose is presented in Table 2. The estimated control room doses for the postulated GDT and VCR ruptures were found to be acceptable.

3.1.3 Atmospheric Dispersion

In performing the radiological consequence analyses supporting the control room habitability amendment submitted on June 12, 2000, the licensee calculated atmospheric dispersion values ($\chi$/Q) using a different methodology than had been used previously in the CNP licensing basis. The licensee did not provide sufficient information for the staff to confirm the new $\chi$/Q values and the staff requested additional information. The licensee responded to this request by providing a table of configuration parameters and floppy disk files of the hourly meteorological observation data for the calendar years 1996-1998. In its review of this information, the staff identified concerns with the quality of the meteorological observation data. The licensee personnel subsequently notified the staff that an error had been uncovered in the processing of the collected meteorological data. This error affected all of the $\chi/Q$ values used in the June 2000, application. The licensee had the affected values re-calculated. Since the licensee had determined that the data for 1998 had been affected by instrumentation problems, data from the years 1995 through 1997 were used in the re-analyses. In lieu of reanalyzing the dose calculations that used these values, the licensee conservatively determined adjustment factors that could be applied to the results of those analyses. Since the LOCA analysis needed to be re-performed, the revised $\chi/Q$ values were used directly in the analysis, and the revised dose results submitted by letter dated June 5, 2002.

In the original submittal, the licensee had classified the power-operated relief valves (PORVs) as elevated stacks based on the energetic nature of the steam releases. Although the staff agrees that the releases will have significant mechanical and buoyant rise, there is insignificant data to support a conclusion that all of the release will be ejected above the aerodynamic flow field around adjacent buildings. In the absence of such data, it must be assumed that some of the release will be entrained in the building wake and diffuse to the ground without the dilution associated with an elevated plume. The staff has taken the position that a stack must be 2.5 times taller than adjacent structures to be considered an elevated release point. The top of CNP steam generator releases are below the height of some adjacent buildings. In lieu of treating these releases as elevated, the licensee has elected to re-calculate the $\chi/Q$ values as ground level releases and apply a deterministic adjustment for plume rise as provided in "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," DG-1115. The vertical velocity of the steam released from the PORVs is approximately 75 miles per hour which exceeds the $95^{th}$-percentile wind speed by about a factor of six. The CNP design-basis analyses assume that the plant remains at hot standby with the PORVs used to remove decay heat as needed for the 30-day duration of the event. The duration of the releases may change as decay heat is removed from the reactor core, but the vertical velocity will remain relatively constant. The licensee states that the dose from a ground level release (NB., unadjusted $\chi/Q$) during a 24-hour cooldown would be bounded by the calculated dose from a 30-day ground level release with a $\chi/Q$ value that has been reduced by the factor of five assumed for plume rise. The staff has determined that the calculated CNP PORV $\chi/Q$s values to be acceptable.

The staff requested the corrected meteorological data be submitted for review and performed statistical checks on the data to confirm adequate quality. The staff also made confirmatory $\chi/Q$ analyses using the ARCON96 code. The $\chi/Q$ values found acceptable by the staff are tabulated in Table 1 on page 19 of this SE.

### 3.1.4 Environment Qualification Impact

The licensee performed an evaluation of the impact of the AST on the integrated radiation doses to equipment subject to environmental qualification criteria. A report on this evaluation was provided as Attachment 8 to the submittal. Paragraphs 1.3.5 and 6 of RG 1.183 address the need to address the impact of the AST implementation on equipment qualification (EQ) doses. The guidance in this section changed between DG-1081 and the final RG 1.183. The current guidance calls for an assessment only if assumptions or inputs of the EQ analyses are affected by the plant modification associated with the AST implementation, if any. The guidance states the licensees need not assess the effect of increased cesium on the sump integrated doses pending the resolution of a generic safety issue (GSI-187). The licensee has proposed no plant modifications that would appear to warrant the analysis. The GSI was closed with the determination that a generic action was not warranted. Therefore, the licensee may continue to use the TID14844 source term in future EQ analyses.

The staff reviewed the sensitivity analysis addressing the impact of increased cesium on the CNMT centerline airborne dose and the dose at the center of a water-filled sump. Based on the information provided, the analysis assumptions and methods were deemed to be acceptable for the stated purpose. The analysis results show that the CNMT atmosphere centerline integrated doses calculated using the TID14844 source term would bound the results obtained using the design basis AST for all periods. The results also show the CNMT sump integrated doses calculated using the TID14844 source term would bound the results obtained using the design-basis AST for periods up to a year.

### 3.1.5 Summary Alternate Source Term 10 CFR 50.67

The staff has reviewed the AST implementation proposed by the licensee for the D.C. Cook Nuclear Plant, Units 1 and 2. The staff also reviewed the proposed changes to the TSs associated with this license amendment request. In doing this review, the staff relied upon information placed on the docket by the licensee, staff experience in doing similar reviews and, where deemed necessary, on staff confirmatory calculations. This licensing action is considered a selective implementation of the AST. While the licensee adopted all characteristics of the AST, their assessment was limited to control room habitability. The licensee did not do offsite dose analyses. With the approval of this amendment, the AST, the TEDE criteria, and the analysis methods, assumptions and inputs used become the licensing basis for the assessment of radiological consequences of design-basis accidents with regard to control room habitability. All future radiological analyses done to show compliance with control room habitability requirements shall use this approved licensing basis. This approval is limited to this specific application. The AST and TEDE criteria may not be extended to other aspects of plant design or operation (e.g., offsite doses) without prior NRC review pursuant to 10 CFR 50.67.

The staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed changes. The staff finds that the licensee used analysis methods and assumptions consistent with the conservative guidance of RG 1.183, with the exceptions discussed and accepted earlier in this SE. The staff compared the control room doses estimated by the licensee to the applicable acceptance criteria and to the results estimated by the staff in its confirmatory calculations. The staff finds, with reasonable assurance, that the licensee's estimates of the TEDE in the control room due to design-basis accidents will comply with the requirements of 10 CFR 50.67 and the guidance of RG 1.183. Therefore, the staff finds the use of the AST is acceptable.

**TABLE 1**

## CONTROL ROOM ANALYSIS ASSUMPTIONS

### Assumptions Common to One or More Analyses

| | | |
|---|---|---|
| Reactor power, MWt | | 3660 |
| RCS mass, gm | | 2.41E8 |
| RCS specific activity, µCi/gm dose equivalent I-131 | | 1.0 |
| RCS to secondary leak rate, gal/min | | 1.0 |
| Dose conversion factors | | FGR11 and FGR12 |
| Iodine spike appearance rate (includes 500 multiplier), Ci/hr | | |
| I-131 | | 11284 |
| I-132 | | 36624 |
| I-133 | | 26284 |
| I-134 | | 17052 |
| I-135 | | 17120 |
| Iodine spike appearance rate based on | | |
| Letdown flow rate, gpm | | 120 |
| RCS leakrate, gpm | | 12 |
| Demineralizer DF | | $\infty$ |
| Iodine spike duration, hrs | | 6 |
| Control room volume, $ft^3$ | | 50616 |
| Normal ventilation makeup flow, cfm | | 1000 |
| Emergency ventilation system | 1 fan | 2 fan |
| Filtered air makeup, cfm | 1000 | 2000 |
| Filtered recirculation, cfm | 4400 | 8800 |
| Unfiltered inleakage, cfm | 98 | 98 |
| Filter efficiency, elemental, percent | 95 | 80 |
| Filter efficiency, organic, percent | 95 | 80 |
| Filter efficiency, particulate, percent | 98 | 98 |
| Control room switchover from normal to emergency after receipt of SI signal, seconds | | 66 |
| Time to secure second pressurization fan, hours | | 2 |
| Control room breathing rate, $m^3$/sec | | 3.47E-4 |
| Control room occupancy factors | | |
| 0-24 hours | | 1.0 |
| 1-4 days | | 0.6 |
| 4-30 days | | 0.4 |

Control room $\chi$/Q, sec/m$^3$

| Period | | Unit Vent | CNMT Surface | S/G |
|---|---|---|---|---|
| 0-2 hrs | 1.77E-3 | 8.99E-4 | 8.00E-4 | |
| 2-8 hrs | 1.24E-3 | 6.29E-4 | 7.00E-4 | |
| 8-24 hrs | | 5.14E-4 | 2.59E-4 | 3.60E-4 |
| 1-4 days | | 3.64E-4 | 1.93E-4 | 2.30E-4 |
| 4-30 days | | 2.77E-4 | 1.57E-4 | 1.77E-4 |

**Assumptions for LOCA Analyses**

Core Inventory — Calculated by RADTRAD

Core release fractions and timing–CNMT atmosphere

| Duration, hrs | 0.0083E+00 | 0.5000E+00 | 0.1300E+01 |
|---|---|---|---|
| Noble Gases: | 1.2500E-04* | 0.5000E-01 | 0.9500E+00 |
| Iodine: | 1.2500E-04 | 0.5000E-01 | 0.3500E+00 |
| Cesium: | 1.2500E-04 | 0.5000E-01 | 0.2500E+00 |
| Tellurium: | 0.0000E+00 | 0.0000E+00 | 0.5000E-01 |
| Strontium: | 0.0000E+00 | 0.0000E+00 | 0.2000E-01 |
| Barium: | 0.0000E+00 | 0.0000E+00 | 0.2000E-01 |
| Ruthenium: | 0.0000E+00 | 0.0000E+00 | 0.2500E-02 |
| Cerium: | 0.0000E+00 | 0.0000E+00 | 0.5000E-03 |
| Lanthanum: | 0.0000E+00 | 0.0000E+00 | 0.2000E-03 |

*Based on 0.25 percent F.F. = 1.0 µCi/gm d.e.I-131 RCS activity, with 100 percent F.F = 5 percent core inventory

Core release fractions and timing–ECCS leakage

| Duration, hrs | 0.0083E+00 | 0.5000E+00 | 0.1300E+01 |
|---|---|---|---|
| Iodine: | 1.2500E-04 | 0.5000E-01 | 0.3500E+00 |
| Cesium: | 1.2500E-04 | 0.5000E-01 | 0.2500E+00 |

| Iodine species fraction | Atmosphere | Sump |
|---|---|---|
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 100 |
| Organic | 0.15 | 0 |

| Containment volume, ft$^3$ | |
|---|---|
| Upper | 900,000 |
| Active lower | 300,000 |
| Dead lower | 62,000 |
| Total | 1,260,000 |

| Containment release, percent/day | |
|---|---|
| 0-280 hours | 0.25 |
| 280-720 hours | 0.125 |

Duration of release, days — 30

| | | |
|---|---|---|
| Containment mixing flow, cfm | | |
| Upper CNMT to lower CNMT (3 minute delay) | | 39,000 |
| Between lower and dead areas | | 500 |
| Containment spray lambda, $hr^{-1}$ | Upper | Lower |
| Elemental, injection phase | 20.0 | 8.9 |
| Elemental, recirculation phase | 7.8 | 3.3 |
| Particulate, injection phase | 3.8 | 1.8 |
| Particulate, recirculation phase | 6.3 | 1.8 |
| Containment spray DF | | |
| Elemental spray cutoff | | 200 |
| Particulate spray reduction | | 50 |
| Containment spray timings | | |
| Injection spray initiation, sec | | 126 |
| Injection spray stops, hrs | | 1.25 |
| Recirculation spray starts, hrs | | 1.33 |
| Particulate spray reduction, hrs | | 2.41 |
| Elemental spray cutoff, hrs | | 2.78 |
| Spray terminated, hrs | | 6.0 |
| Containment purge flow rate 0-10 seconds, cfm | | 16,000 |
| Ice condenser | | |
| elemental iodine removal efficiency, percent | | 30 |
| DF | | 200 |
| DF reached, hrs | | 2 |
| Credit for ice condenser iodine removal ends, hrs | | 1 |
| ECCS leak rate, gpm | | 0.2 |
| Duration of release, days | | 30 |

Containment sump volume, gals

| Period | Volume | Period | Volume |
|---|---|---|---|
| 0-2 hours | 2.1E5 gal | 150-200 hours | 4.61E5 gal |
| 2-8 hours | 3.29E5 gal | 200-250 hours | 4.79E5 gal |
| 8-24 hours | 3.53E5 gal | 250-300 hours | 4.91E5 gal |
| 24-48 hours | 3.77E5 gal | 300-336 hours | 5.03E5 gal |
| 48-100 hours | 4.01E5 gal | > 336 hours | 5.1E5 gal |
| 100-150 hours | 4.31E5 gal | | |

Containment surface $\chi/Q$ used for containment leakage

Plant vent used for purge release, ECCS leakage, and passive failure

**Assumptions for Small-Break LOCA Analyses**

| | |
|---|---|
| Core Inventory | |
| CNMT Atmosphere | Calculated by RADTRAD |
| Secondary Release | Table 14.A.4-4, UFSAR |

Core release fractions and timing–CNMT atmosphere

| Duration, hrs | 0.0083E+00 | 0.5000E+00 |
|---|---|---|
| Noble Gases: | 7.1000E-03* | 0.5000E-01 |
| Iodine: | 7.1000E-03 | 0.5000E-01 |
| Cesium: | 7.1000E-03 | 0.5000E-01 |

*Based on 14.2 percent F.F. = 60.0 µCi/gm d.e.I-131 RCS activity, with 100 percent F.F = 5 percent core inventory

| | | |
|---|---|---|
| Secondary release source term | 5 percent gap activity plus 60 µCi/gm iodine spike | |
| Iodine species fraction | Atmosphere | SG |
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 100 |
| Organic | 0.15 | 0 |
| Containment volume, $ft^3$ | | |
| Upper | 900,000 | |
| Lower | 360,000 | |
| Total | 1,260,000 | |
| Containment release, percent/day | | |
| 0-280 hours | 0.25 | |
| 280-720 hours | 0.125 | |
| Duration of release, days | 30 | |
| Containment mixing flow, cfm | | |
| Upper CNMT to lower CNMT (3 minute delay) | 39,000 | |
| Containment particulate sedimentation lambda, $hr^{-1}$ | 0.1 | |
| Containment sedimentation DF | | |
| Particulate spray reduction | 1000 | |
| Cutoff, hrs | 65 | |
| Steam generator mass (all four), g | 1.65E8 | |
| Steaming rate (total), g/min | 1.75E6 | |
| Release duration, hours | 2 | |
| Steam partition coefficient | 0.01 | |

Containment surface $\chi/Q$ used for containment leakage

SG $\chi/Q$ used for steam generator releases

## Assumptions for Rod Ejection Accident Analyses

| | | |
|---|---|---|
| Core Inventory | | Table 4, Submittal |
| RCS activity | | Table 14.A.4-4, UFSAR normalized to 60 µCi/gm d.e.I-131 |
| Fraction of rods that exceed DNB | | 0.15 |
| Gap fraction, all nuclide groups | | 0.05 |
| Fraction of rods that exceed DNB that experience melt | | 0.00375 |
| Melt isotopic composition | | |
| Noble gases | | 0.95 |
| Iodine | | 0.35 |
| Alkali metals | | 0.25 |
| Secondary release source term | | Fuel release plus 60 µCi/gm iodine spike |
| Iodine species fraction | Atmosphere | SG |
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 100 |
| Organic | 0.15 | 0 |
| Containment release, percent/day | | |
| 0-280 hours | | 0.25 |
| 280-720 hours | | 0.125 |
| Duration of release, days | | 30 |
| Steam generator mass (all four), g | | 2.087E8 |
| Steaming rate (total), g/min | | 1.74E6 |
| Release duration, hours | | 2 |
| Steam partition coefficient | | 0.01 |

Containment surface $\chi$/Q used for containment leakage

SG $\chi$/Q used for steam generator releases

## Assumptions for Small-Break LOCA Analyses

| | |
|---|---|
| Core Inventory | |
| CNMT Atmosphere | Calculated by RADTRAD |
| Secondary Release | Table 14.A.4-4, UFSAR |

Core release fractions and timing–CNMT atmosphere

| Duration, hrs | 0.0083E+00 | 0.5000E+00 |
|---|---|---|
| Noble Gases: | 7.1000E-03* | 0.5000E-01 |
| Iodine: | 7.1000E-03 | 0.5000E-01 |
| Cesium: | 7.1000E-03 | 0.5000E-01 |

*Based on 14.2 percent F.F. = 60.0 µCi/gm d.e.I-131 RCS activity, with 100 percent F.F = 5 percent core inventory

| | | |
|---|---|---|
| Secondary release source term | 5 percent gap activity plus 60 µCi/gm iodine spike | |
| Iodine species fraction | Atmosphere | SG |
| Particulate/aerosol | 95 | 0 |
| Elemental | 4.85 | 100 |
| Organic | 0.15 | 0 |
| Containment volume, $ft^3$ | | |
| Upper | | 900,000 |
| Lower | | 360,000 |
| Total | | 1,260,000 |
| Containment release, percent/day | | |
| 0-280 hours | | 0.25 |
| 280-720 hours | | 0.125 |
| Duration of release, days | | 30 |
| Containment mixing flow, cfm | | |
| Upper CNMT to lower CNMT (3 minute delay) | | 39,000 |
| Containment particulate sedimentation lambda, $hr^{-1}$ | | 0.1 |
| Containment sedimentation DF | | |
| Particulate spray reduction | | 1000 |
| Cutoff, hrs | | 65 |
| Steam generator mass (all four), g | | 1.65E8 |
| Steaming rate (total), g/min | | 1.75E6 |
| Release duration, hours | | 2 |
| Steam partition coefficient | | 0.01 |
| Containment surface $\chi$/Q used for containment leakage | | |
| SG $\chi$/Q used for steam generator releases | | |

### Assumptions for SGTR Analyses

| | |
|---|---|
| RCS activity | |
| Co-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 1.0 µCi/gm d.e.I-131 |
| Pre-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 60.0 µCi/gm d.e.I-131 |
| Primary-to-secondary break flow, lbm | 162,000 |
| Break flow flash fraction | 0.168 |

| | |
|---|---|
| Break flow duration, min | 30 |
| Steam generator mass @, lbm | 99,000 |
| Steam release from ruptured SG (30 minutes), lbm | 73,000 |
| Steam release from unaffected SGs, lbm | |
| 0-2 hours | 565,000 |
| 2-8 hours | 1,505,000 |
| 8-24 hours | 1,482,000 |
| 1-7 days | 9,729,000 |
| 7-30 days | 36,871,000 |
| Pre-event steam generator activity | 0.1 µCi/gm d.e.I-131 |
| Steam partition coefficient | 0.01 |
| Control room HVAC switchover, min | 7 |
| SG χ/Q used for steam generator releases | |

**Assumptions for MSLB Analyses**

| | |
|---|---|
| RCS activity | |
| Co-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 1.0 µCi/gm d.e.I-131 |
| Pre-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 60.0 µCi/gm d.e.I-131 |
| Faulted SG blowdown (100 percent) duration, minutes | 2 |
| Primary-to-secondary leakage | |
| To faulted SG, gpd | 500 |
| To unaffected SGs, gpd | 940 |
| Primary to secondary leakage duration, days | 30 |
| Steam generator mass @, lbm | 99,000 |
| Steam release from faulted SG (2 minutes), lbm | 99,000 |
| Steam release from unaffected SGs, lbm | |
| 0-2 hours | 456,000 |
| 2-8 hours | 1,186,000 |
| 8-24 hours | 1,347,000 |
| 1-7 days | 8,844,000 |
| 7-30 days | 33,519,000 |
| Pre-event steam generator activity | 0.1 µCi/gm d.e.I-131 |
| Steam partition coefficient (unaffected SG) | 0.01 |
| Control room HVAC switchover, min | 5 |
| SG χ/Q used for unaffected SG releases | |
| Plant vent χ/Q used for faulted SG releases | |

## Assumptions for LOOP Analyses

| | |
|---|---|
| RCS activity | |
| Co-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 1.0 µCi/gm d.e.I-131 |
| Pre-incident iodine spike case | Table 14.A.4-4, UFSAR, normalized to 60.0 µCi/gm d.e.I-131 |
| Primary-to-secondary leakage, gpm | 1 |
| Primary to secondary leakage duration, days | 30 |
| Steam generator mass @, lbm | 99,000 |
| Steam release from faulted SG (2 minutes), lbm | 99,000 |
| Steam release from unaffected SGs, lbm | |
| 0-2 hours | 738,000 |
| 2-8 hours | 1,345,000 |
| 8-24 hours | 1,590,000 |
| 1-7 days | 11,030,000 |
| 7-30 days | 36,871,000 |
| Pre-event steam generator activity | 0.1 µCi/gm d.e.I-131 |
| Steam partition coefficient (unaffected SG) | 0.01 |
| Control room HVAC switchover, min | never |
| SG $\chi$/Q used for unaffected SG releases | |

## Assumptions for GDT/VCT Failure Analyses

<u>Gas Decay Tank Failure Assumptions</u>

| | |
|---|---|
| Inventory of GDT | Equal to the activity in the entire RCS at 1.0 percent F.F. |
| Release duration, minutes | 5 |

<u>VCT Failure Assumptions</u>

| | |
|---|---|
| Basis of inventory of VCT, noble gases | Continuous full power operation with 1.0 percent F.F. with no purge |
| Basis of inventory of VCT, iodine | RCS activity at 60 µCi/gm with letdown demin DF of 10 |
| VCT activity release | |
| Noble gases | 100 percent in 5 minutes |
| Iodine | 1 percent in 5 minutes |
| Letdown isolation time, minutes | 15 |
| Letdown flow rate, gpm | 120 |

| | |
|---|---|
| Letdown demin DF for iodine | 10 |
| Unit Vent $\chi$/Q used | |

3.2 TSTF -287 "Ventilation System Envelope Outage Time"

The proposed changes would allow up to 24 hours to restore the Control Room Envelope/Pressure Boundary (CRE/PB) to operable status when two inoperable independent pressurization trains are inoperable due to an inoperable CRE/PB in MODES 1, 2, 3, and 4. In addition, a Limiting Condition for Operation (LCO) Note would be added to allow the CRE/PB to be opened intermittently under administrative control without affecting the Control Emergency Ventilation System (CREVS). The applicable TS Bases will be revised to document the TS changes and to provide supporting information. These proposed changes are based on TSTF change request-287 to the Standard Technical Specifications (STS).

The existing LCO 3.7.5.1 surveillance requirements (SR) 4.7.5.1.e.3 that test the integrity of the CRE/PB require a positive pressure limit to be satisfied with one independent pressurization train. While other SRs in the same specification test the operability and function of the independent pressurization train, currently the intent of the positive pressure test is to ensure that the CRE/PB leak tightness is adequate to meet design assumptions for post-accident operator doses.

Currently, there are no corresponding Actions, or completion times specified in TS LCO 3.7.5.1 should the CRE/PB not be met. Under the existing specifications, LCO 3.0.3 must be entered (for two-train inoperability). Requiring the plant to enter LCO 3.0.3 when the CRE/PB is not intact does not provide time to effect required repairs or corrective maintenance activities.

The proposed change is similar in nature to STS LCOs for secondary containment for boiling-water reactor and shield building for pressurized-water reactor which allows 24 hours to restore secondary containment or shield building envelope to operable status before requiring an orderly shutdown from operating conditions.

The proposed changes to TS 3/4.7.5 are:

1. A note has been added to LCO 3.7.5.1 for the CREVS to allow the CRE/PB to be opened intermittently under administrative control. Corresponding Bases have been added which establish the administrative control that is required to minimize the consequences of the open boundary.

2. A new Action c in Modes 1, 2, 3, and 4 is added to LCO 3.7.5.1 to specify that 24 Hours are allowed to restore an inoperable CRE/PB to operable status. All other conditions have been administratively re-labeled to support this change. Corresponding TS Bases 3/4.7.5 are added to support this change.

3. An existing LCO 3.7.5.1.a is relocated to a new LCO 3.7.5.2, "Control Room Air Conditioning System," An existing "LCO 3.7.5.1.b" is renamed as "LCO 3.7.5.1.a" by renaming "pressurization fan" to "pressurization train." Also, an existing "LCO 3.7.5.1.c" is renamed to be a new "LCO 3.7.5.1.b" by renaming "filter train" to

"filter unit." (These changes are administrative changes and do not result in any change in the actual requirements) and associated TS Bases 3/4.7.5 are revised accordingly.

4. An existing "Action a" to LCO 3.7.5.1 and SR 4.7.5.1.a are relocated to a new LCO 3.7.5.2 and SR 4.7.5.2, respectively for the newly created TS, "Control Room Air Conditioning System," as a subset of TS 3/4.7.5. An existing "Action b" is renamed as "Action a" by renaming "pressurization fan" to "pressurization train." Also, an existing "Action c" is renamed to be a new "Action b" by renaming "filter train" to "filter unit." (These changes are administrative changes and do not result in any change in the actual requirements).

The LCO is modified by a Note allowing the CRE/PB 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 control room area isolation is indicated.

If the CRE/PB is inoperable in MODES 1, 2, 3, and 4, such that the CREVS trains cannot establish or maintain the required pressure, action must be taken to restore CRE/PB to an OPERABLE status within 24 hours.

The proposed changes would allow 24 hours (during Modes 1, 2, 3, and 4 ) to restore the capability to maintain CRE/PB pressure before requiring the unit to perform an orderly shutdown and also allows intermittent opening of the CRE/PB under administrative controls. During the period that the CRE/PB is inoperable, appropriate compensatory measures consistent with the intent of GDC 19 will be utilized to protect the control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity and to ensure physical security. These preplanned measures will be available to address these concerns for intentional and unintentional entry into the condition. For example, when the CRE/PB is opened for other than entry through doors, the proposed bases require that, in addition to other necessary measures, a dedicated individual be stationed in the area keeping continuous contact with the control room to rapidly restore the CRE/PB.

Additionally, the proposed change is considered acceptable because of the low probability of an event (design-basis accident) requiring an intact CRE/PB occurring during the 24-hour action completion time associated with Action "c" as indicated in Item 2 above.

Based on the low probability of an event occurring in this time and the availability of compensatory measures consistent with GDC 19 to minimize the consequences during an event, the proposed change is considered acceptable and is in conformance with TSTF-287.

3.2.1 Summary TSTF 287

These changes are based on TSTF change request 287 to the STS.

The Staff finds these changes acceptable because they conform to TSTF change request 287 to the STS, and there are other acceptable changes that are administrative and do not affect the performance of any equipment.

The Staff concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

## 4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (65 FR 51356). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 6.0 CONCLUSION

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

Principal Contributor: S. LaVie
J. Raval
M. Shuaibi

Date: November 14, 2002