Indiana Michigan
Power Company
Cook Nuclear Plant
500 Circle Drive
Buchanan, MI 49107
616-465-5901

INDIANA
MICHIGAN
POWER

August 17, 2001

C0801-02
10 CFR 50.90

Docket No.: 50-315
50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Units 1 and 2
FINAL RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING LICENSE AMENDMENT REQUEST FOR CONTROL ROOM
HABITABILITY
(TAC NOS. MA9394 AND MA9395)

References:
1) Letter from R. P. Powers Indiana Michigan Power Company (I&M) to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for Control Room Habitability and Generic Letter 99-02 Requirements," C0600-13, dated June 12, 2000.

2) Letter from J. F. Stang (NRC), to R. P. Powers (I&M) "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information, License Amendment Request for Control Room Habitability," dated March 29, 2001 (TAC Nos. MA9394 and MA9395).

3) Letter from M. W. Rencheck (I&M) to NRC Document Control Desk, "Partial Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," (TAC Nos. MA9394 and MA9395)," C0601-03, dated June 19, 2001.

A001

In Reference 1, I&M proposed to amend the Facility Operating Licenses, DPR-58 and DPR-74, for Donald C. Cook Nuclear Plant, Unit 1 and Unit 2. The proposed amendment would allow use of the methodology and alternative source term described 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." I&M also proposed changes to Appendix A, Technical Specifications (TS) of the Facility Operating Licenses. The proposed TS changes affected ventilation system requirements contained in Section 3.7 of the TS, and included implementation of actions specified by NRC Generic Letter 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal," dated June 3, 1999.

In Reference 2, the NRC requested additional information regarding the amendment proposed in Reference 1. In Reference 3, I&M provided that portion of the information requested in Reference 2 needed to support NRC review of the proposed fuel handling accident analysis included in Reference 1. In Reference 3, I&M also committed to provide the remainder of the information requested by Reference 2 in a supplemental response.

Attachment 1 to this letter provides the remainder of the information requested by Reference 2. The responses to two of the questions contained in Reference 2 involve revision of certain previously proposed TS changes. Attachments 2A and 2B provide the affected Unit 1 and Unit 2 TS pages marked to show the previously proposed changes as revised by this letter. Attachments 3A and 3B provide clean copies of the affected Unit 1 and Unit 2 TS pages with all changes incorporated. Other pages provided in Reference 1 are unaffected. Attachment 4 provides a listing of new commitments made in this letter.

As described in Attachment 1 to this letter, the No Significant Hazards Consideration Evaluation provided as Attachment 4 to Reference 1 remains valid. The environmental assessment provided in Attachment 5 to Reference 1 also remains valid.

Should you have any questions, please contact Mr. Ronald W. Gaston, Manager of Regulatory Affairs, at (616) 697-5020.

Sincerely,

M. W. Rencheck
Vice President, Nuclear Engineering

/dmb

Attachments

c: J. E. Dyer
MDEQ - DW & RPD
NRC Resident Inspector
R. Whale

AFFIRMATION

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

Indiana Michigan Power Company

M. W. Rencheck
Vice President Nuclear Engineering

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 17 DAY OF August, 2001

Margaret Snegal
Notary Public

My Commission Expires 11-23-2005

ATTACHMENT 1 TO C0801-02

FINAL RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUEST FOR ADDITIONAL INFORMATION
REGARDING PROPOSED CONTROL ROOM HABITABILITY LICENSE AMENDMENT

References: 1) Letter from R. P. Powers Indiana Michigan Power Company (I&M) to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for Control Room Habitability and Generic Letter 99-02 Requirements," C0600-13, dated June 12, 2000.

2) Letter from J. F. Stang, (NRC), to R. P. Powers (I&M) "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Additional Information, License Amendment Request for Control Room Habitability," dated March 29, 2001 (TAC Nos. MA9394 and MA9395).

3) Letter from M. W. Rencheck (I&M) to NRC Document Control Desk, "Partial Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," (TAC Nos. MA9394 and MA9395)," C0601-03, dated June 19, 2001.

In Reference 1, I&M proposed to amend the Facility Operating Licenses, DPR-58 and DPR-74, for Donald C. Cook Nuclear Plant (CNP), Unit 1 and Unit 2. The proposed amendment would allow use of the methodology and alternative source term (AST) described 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." I&M also proposed changes to Appendix A, Technical Specifications (TS) of the Facility Operating Licenses. The proposed TS changes affected ventilation system requirements contained in Section 3.7 of the TS, and included implementation of actions specified by NRC Generic Letter (GL) 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal," dated June 3, 1999.

In Reference 2, the NRC requested additional information regarding the amendment proposed in Reference 1. In Reference 3, I&M provided that portion of the information requested in Reference 2 needed to support NRC review of the proposed fuel handling accident analysis included in Reference 1. In Reference 3, I&M also committed to provide the remainder of the information requested by Reference 2 in a supplemental response.

This attachment provides the remainder of the information requested by Reference 2. The responses to NRC Questions 2 through 5, and 7 through 21 provide only technical information supporting the previously submitted amendment request and, therefore, do not affect the original evaluation of significant hazards considerations performed in accordance with 10 CFR 50.92 as documented in Attachment 4 of Reference 1. The responses to NRC Questions 1 and 6 involve revision of certain parts of the previously submitted amendment request. As described in detail in the responses to NRC Questions 1 and 6 below, these responses also do not affect the original evaluation of significant hazards considerations.

NRC Question 1

Requested Action 2 of generic letter (GL) 99-02 states, "If the system has a face velocity greater than 110 percent of 0.203 m/s [40 ft/min], then the revised technical specification (TS) should specify the face velocity."

Please refer to or provide docketed information which indicates the actual system face velocity and/or the actual residence time for the control room emergency ventilation system (CREVS), engineered safety features ventilation system (ESFVS), and storage pool ventilation system (SPVS) and describes how it is calculated for these systems.

The actual system face velocities can be calculated by dividing the maximum accident condition system flow rates specified in the TS (nominal + typically 10 percent upper value) by the total exposed surface area of the charcoal filter media. (The guidance on calculation of the residence times in American Society of Mechanical Engineers (ASME) AG-1-1997, Division II, Sections FD and FE, Articles I-1000, or in American National Standards Institute (ANSI) N510-1975 can be used to calculate the actual system face velocities). It should be noted that the face velocity should be consistent with the bed depth and residence time. (Bed Depth = Face Velocity x Residence Time)

I&M Response to NRC Question 1

As documented in Reference 3, I&M has determined that the face velocity for the CREVS charcoal adsorbers at the maximum accident condition system flow rates specified in the TS is 43.7 ft/min, which is less that the value specified in GL 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal," (110% of 40 ft/min = 44 ft/min). However, the face velocities for the charcoal adsorbers in the ESFVS and the SPVS were determined to be 45.5 ft/min and 46.8 ft/min respectively, which exceed the value specified in GL 99-02. The statement in Attachment 1 of Reference 1 that the face velocities for the three systems are less than 40 ft/min were based on erroneous information in the original Question and Answer portion of the Final Safety Analysis Report. Therefore, the previously proposed changes to the Unit 1 and 2 TS Surveillance Requirements for the ESFVS and the SPVS, TS 4.7.6.1 and TS 4.9.12, have been revised to include the appropriate face velocity for the system, 45.5 ft/min or 46.8 ft/min, in the charcoal adsorber sample test conditions.

The TS pages affected by the revision, marked to show all currently proposed changes to the pages, are provided in Attachments 2A and 2B for Unit 1 and 2, respectively. Attachments 3A and 3B show these TS pages with all currently proposed changes to the pages incorporated.

The revision provides assurance that the conditions under which the charcoal adsorber samples are tested are consistent with the potential face velocities for the ESFVS and the SPVS, and is in accordance with the guidance provided in GL 99-02. The revised surveillance requirement is more restrictive than the previously proposed change, since it requires that the face velocity used in the surveillance test be higher than that required by the referenced testing standard, ASTM D3803-1989, "Standard Test Method for Nuclear Grade Activated Charcoal." The revision of the previously proposed change does not involve new plant equipment or operation of existing plant equipment in a new or different manner. The revision of the previously proposed change does not affect or create new accident initiators or precursors, and does not affect safety margins. Therefore, the No Significant Hazards Consideration Evaluation provided as Attachment 4 to Reference 1 remains valid.

NRC Question 2

In order for the staff to verify that a safety factor as low as two is used, the staff needs to know the charcoal adsorber removal efficiencies which are credited in the current and proposed radiological accident analyses for organic iodide.

I&M Response to NRC Question 2

The current charcoal adsorber efficiencies and those proposed in Reference 1 are provided below for the CREVS, ESFVS and SPVS, along with the associated safety factors.

CREVS

The current TS Surveillance Requirements for laboratory testing of the CREVS charcoal adsorber with radioactive methyl (organic) iodide, TS 4.7.5.1.c.3, TS 4.7.5.1.d.1, and TS 4.7.5.1.d.2, require that charcoal samples demonstrate an efficiency of greater than or equal to 90% i.e., 10% penetration. The only control room dose analysis in the current CNP licensing basis is an analysis of the dose resulting from a large break loss of coolant accident (LOCA). As documented in Updated Final Safety Analysis Report (UFSAR) Unit 1 Table 14.3.5-9 and Unit 2 Section 14.3.5.3.7, the CREVS charcoal adsorber efficiency assumed in this analysis is 95%, i.e., 5% penetration. Therefore, the current TS do not provide a safety factor with respect to the CREVS charcoal adsorber efficiency assumed in the current analysis. I&M has implemented procedural controls requiring that laboratory testing of the CREVS charcoal adsorber with radioactive organic iodide demonstrates a penetration of less than or equal to 1.0%. The 5% penetration value assumed in the current analyses divided by the 1.0% testing criteria specified in the administrative controls results in a safety factor equal to 5.

As described in the response to NRC Question 2 in Reference 3, the 5% penetration value assumed in the proposed analyses divided by the 1.0% penetration testing criteria specified in the proposed amendment results in a safety factor equal to 5 for single fan operation of the CREVS.

The effect of two-fan operation on the CREVS was not evaluated in the current accident analysis. The charcoal adsorber removal efficiency assumed in the proposed accident analyses when two CREVS fans are operating was determined by an I&M calculation. This calculation is described in the response to NRC Question 3(a) below. The calculation demonstrates that, for a flow rate of 12,000 cfm, the charcoal adsorber penetration would be 10%. An efficiency of 80%, i.e., a penetration of 20% was assumed in the proposed accident analyses. The 20% value assumed in the proposed analysis divided by the 10% calculated penetration results in a safety factor equal to 2.

ESFVS

The current TS Surveillance Requirements for laboratory testing of the ESFVS charcoal adsorber with radioactive organic iodide, TS 4.7.6.1.b.4, TS 4.7.6.1.c.1, and TS 4.7.6.1.c.2, require that samples of the charcoal demonstrate an efficiency of greater than or equal to 90%, i.e., 10% penetration. The only control room dose analysis in the current CNP licensing basis is an analysis of the dose resulting from a large break LOCA. As documented in UFSAR Unit 1 Table 14.3.5-9 and Unit 2 Section 14.3.5.3.7, no credit for the ESFVS charcoal adsorbers was assumed in this analysis. Therefore, there is no current safety factor applicable to the ESFVS charcoal adsorber efficiency for control room doses. Operation of the ESFVS charcoal adsorbers was not credited in any of the current off-site dose analyses.

The proposed TS Surveillance Requirements for laboratory testing of EFSVS charcoal adsorber samples with radioactive organic iodide require a penetration of less than or equal to 5.0%. In the manner described below, the proposed large break LOCA analysis indirectly credits charcoal adsorber removal efficiency greater than or equal to 90% i.e., 10% penetration. The 10% value assumed in the proposed analyses divided by the 5.0% testing criteria specified in the proposed amendment results in a safety factor equal to 2.

The proposed large break LOCA analysis credits the ESFVS charcoal adsorbers as follows. Some potential leak locations in the emergency core cooling systems (ECCS) are subject to filtration by the ESFVS while others are not. A total effective unfiltered leak rate of 0.2 gpm was assumed in the proposed accident analysis. This assumption is maintained by the methodology and limits imposed by CNP procedure for determining ECCS leakage. This procedure is used to measure ECCS leakage and categorize the leakage as filtered or unfiltered by the ESFVS based on its source. If the leakage is categorized as filtered, then the leak rate value is reduced by a factor of 10 to account for removal of 90% of the iodine by filtration. If the leakage is categorized as unfiltered, then the leak rate value is not reduced. The following table provides three examples of filtered and unfiltered leak rate combinations all resulting in a total effective unfiltered leak rate of 0.2 gpm. In this manner, the total effective unfiltered leak rate is maintained below 0.2 gpm even though the actual total leak rate may exceed this value.

Examples of Summing Filtered and Unfiltered Sources of ECCS Leakage		
Filtered Leak Rate (gpm)	Unfiltered Leak Rate (gpm)	Total Effective Unfiltered Leak Rate (gpm)
0	0.2	0.2
1.0	0.1	0.2
2.0	0.0	0.2

SPVS

The current TS Surveillance Requirements for laboratory testing of the SPVS charcoal adsorber with radioactive organic iodide, TS 4.9.12.b.4, TS 4.9.12.c.1, and TS 4.9.12.c.2, require that samples of the charcoal demonstrate an efficiency of greater than or equal to 90%, i.e., a penetration of 10%. The SPVS charcoal adsorbers were not credited in any control room dose analysis in the current CNP licensing basis. Therefore, there is no current safety factor applicable to the SPVS charcoal adsorber efficiency for control room doses.

As documented in UFSAR Unit 1 Section 14.2.1.4 and Unit 2 Section 14.2.1, the current licensing basis analysis of the off-site dose resulting from a fuel handling accident in the auxiliary building credits a SPVS charcoal adsorber decontamination factor of 10, i.e., a penetration of 10%. Therefore, the current TS do not provide a safety factor with respect to the SPVS charcoal adsorber efficiency assumed in the current off-site dose analysis. I&M has implemented procedural controls requiring that laboratory testing of the SPVS charcoal adsorber with radioactive organic iodide demonstrates a penetration of less than or equal to 5.0%. The 10% penetration value assumed in the current analyses divided by the 5.0% testing criteria specified in the administrative controls results in a safety factor equal to 2.

The proposed TS Surveillance Requirements for laboratory testing of SPVS charcoal adsorber samples with radioactive organic iodide require a penetration of less than or equal to 5.0%. The proposed accident analyses take no credit for the SPVS charcoal adsorber. Therefore, there is no safety factor applicable to the proposed SPVS charcoal adsorber efficiency for control room doses.

NRC Question 3

On page 19 of Attachment 1 to Letter C0600-13, it is stated that in case of CREVS the recent accident analyses assume 95 percent iodine removal efficiency for single-fan operation under normal system flow rate and 80 percent removal efficiency for two-fan operation at an increased face velocity during the first two hours of the accident. It is also stated that "...The 80 percent efficiency calculation includes a safety factor of two. To ensure the accident analysis assumptions remain valid for both single and two-fan operation, the surveillance requirement is revised to demonstrate a penetration of less than or equal to 1 percent when tested at normal system flow rate."

(a) *Clarify how at 80 percent filter efficiency the safety factor of two is calculated.*

(b) *For two-fan operation, what is actual increased maximum face velocity across the charcoal bed.*

(c) *Explain how 80 percent filter efficiency at increased face velocity compares with 95 percent filter efficiency at normal system flow rate.*

(d) *Demonstrate how the 1 percent penetration at normal system flow rate as the surveillance requirement bound both single and two-fan operation cases.*

I&M Response to NRC Question 3(a)

The CREVS charcoal adsorber removal efficiency used in the proposed analyses when two fans are operating was determined by an I&M calculation. A charcoal adsorber efficiency of 99%, i.e., a penetration of 1%, was assumed in the calculation for the nominal single fan flow rate, 6000 cfm. This is consistent with the proposed TS Surveillance Requirement for laboratory testing of charcoal with radioactive organic iodide and the existing TS Surveillance Requirement for in-place testing of charcoal adsorbers with halogenated hydrocarbon refrigerant.

The penetration resulting from the higher flowrate through the filter when two fans are running was determined using the method described in Section 9.2 of ASTM D3803-89 for substandard bed depth, by treating the doubling of the face velocity as halving of the bed depth. The conservatism of this approach is demonstrated by a comparison with measured data. Measurements of penetration by NUCON International, Inc. varying both face velocity and the bed depth are summarized in the table below. The first three columns provide the measured bed depth, face velocity and penetration, respectively. The fourth column provides the adjusted penetration using the ASTM method for substandard bed depth. The table shows that calculating the penetration for a 2 inch bed depth at a face velocity of 80 ft/min using the method described in Section 9.2 of ASTM D 3803-89 for substandard bed depth results in a more conservative value than indicated by the NUCON measured data.

Measured Bed Depth (in)	Measured Face Velocity (ft/min)	Measured Penetration (%)	Calculated Penetration (%)
2	80	3.3	5.00
2	70	1.8	3.26
2	60	0.9	1.84
2	50	0.5	0.83
2	40	0.25	0.25

As discussed below in the response to NRC Question 3(b), the actual airflow for two-fan operation is much less than twice the air flow for single fan operation. However, a flow rate of 12,000 cfm was conservatively used in the calculation. Using the method described above, the penetration for 12,000 cfm was calculated to be 10%. The charcoal adsorber efficiency used in the proposed analyses for two-fan operation was 80%, i.e., a penetration of 20%. This provides a safety factor of 2.

I&M Response to NRC Question 3(b)

The actual increase in face velocity across the charcoal bed with two fans operating is much less than that resulting from a 12,000 cfm flow rate. I&M performed a calculation to determine the flow rate with both fans operating, using system resistance curves and fan performance curves. Results of the calculation show that the nominal flow rate through the filter with two fans operating is 7500 cfm. As documented in Reference 3, I&M has determined that the maximum single fan flow rate of 6600 cfm would result in a face velocity of 43.7 ft/min. The face velocity resulting from the actual flow rate of 7500 cfm is therefore 43.7 x 7500/6600 = 49.7 ft/min.

I&M Response to NRC Question 3(c)

The charcoal adsorber efficiency for normal (single) fan operation is higher than the efficiency for two fan operation and therefore provides more protection to personnel in the control room. The proposed analyses account for the reduced protection offered with two fans operating by assuming the second fan is turned off immediately, at 30 minutes into the event, or at 2 hours into the event, whichever is most limiting.

The different efficiencies for single fan operation and for two-fan operation are accounted for in the proposed analyses. An efficiency of 95%, i.e., 5% penetration, was assumed for single fan operation. This provides a safety factor of 5 based on the 1% penetration criterion specified in the proposed Surveillance Requirements. As described in the response to NRC Question 3(a), an efficiency of 80% was assumed in the proposed analysis for two fan operation, also based on the 1% penetration criterion specified in the proposed Surveillance Requirements. The response to NRC Question 3(a) also describes how the 80% value provides a safety factor of 2 for two fan operation.

As also described in the response to NRC Question 3(a), the two fan safety factor of 2 is based on assuming that the two fan flow rate is 12,000 cfm which is twice the single fan flow rate of 6000 cfm. This assumption is conservative in that, as described in the response to NRC Question 3(b), the actual two-fan flow rate has been calculated to be 7500 cfm. A two fan flow rate of 7500 results in a safety factor greater than 5.

I&M therefore considers that both the 95% and the 80% values assumed for single fan and two fan charcoal adsorbers efficiencies in the proposed analyses provide safety factors that meet or exceed the value of 2 specified in GL 99-02.

I&M Response to NRC Question 3(d)

The response to NRC Question 3(d) in Reference 3 provided a description of how the 1% penetration at normal system flow rate specified by the proposed Surveillance Requirement bounds the cases that assume single fan operation.

The above response to NRC Question 3(a) includes a description of how the 1% penetration at normal system flow rate specified by the proposed Surveillance Requirement bounds the cases that assume two-fan operation.

NRC Question 4

For accidents where the CREVS is not operated in the emergency mode, provide the bases for the assumption of only 1000 cfm of unfiltered makeup since there is no indication that other sources of unfiltered inleakage are considered.

I&M Response to NRC Question 4

The response to NRC Question 4 in Reference 3 provided a description of how the unfiltered makeup value that was assumed in the proposed fuel handling accident (FHA) analysis precluded the need to consider other sources of unfiltered inleakage.

The other proposed analyses in which it is assumed that the CREVS is operated in normal mode are the analyses of a loss of offsite power (LOOP), a gas decay tank (GDT) rupture, and a volume control tank (VCT) rupture. In these proposed analyses, it was assumed that the normal ventilation system remains in operation with a maximum makeup rate of 1000 cfm. This makeup is unfiltered, since the HEPA and charcoal adsorbers are in the flow path only in the emergency ventilation mode.

No other sources of unfiltered inleakage were identified and, therefore, none were specifically considered in these analyses. This position is supported by tracer gas testing results. During tracer gas testing, unfiltered inleakage was determined to be 49 $\pm$ 49 cfm when measured with the CREVS in the emergency mode. In the emergency mode, the normal intake dampers are closed, creating a significant differential pressure across these dampers. The closed intake dampers are the only location in the control room envelope/pressure boundary where a large differential pressure exists to force outside air into the control room. Tracer gas testing demonstrated that a portion of the measured inleakage occurred at the normal intake damper.

Since these dampers are open during normal operation, unfiltered inleakage into the pressure boundary is expected to be minimal.

Notwithstanding the lack of identified sources of unfiltered inleakage, the analysis does include margin that could accommodate up to 40 cfm of unfiltered inleakage. Specifically, periodic surveillance testing verifies that the unfiltered makeup flow for CREVS in the normal mode is 740 to 960 cfm. The difference between the maximum allowed surveillance test value of 960 cfm and the 1000 cfm assumed in the proposed FHA analysis, i.e., 40 cfm, provides a margin that can be used to account for potential unfiltered inleakage.

Finally, in the unexpected event that unfiltered inleakage plus normal inflow exceeds 1000 cfm, the potential impact on the proposed analyses of the GDT and VCT rupture events and the LOOP event is minimal, as described below.

As indicated in the proposed analyses, the GDT and VCT rupture events would result in a brief (5 to 15 minute) radiological release. Unfiltered inleakage would therefore increase the initial radionuclide concentration in the control room. However, once the release ended, the unfiltered inleakage would aid in purging the radionuclides from the control room. With CREVS in the normal mode, unfiltered inleakage or makeup would be the only mechanism available for reducing the radionuclide concentration in the control room. Since the 5 to 15 minute release period for these events is short compared to the 30 day duration of the event, the cleanup effect of unfiltered inleakage and makeup would tend to offset the higher initial radionuclide concentration. As a result, the overall impact of changes in unfiltered inleakage or makeup on the dose consequences is minimal.

The release period for the LOOP event is long, 30 days, compared to the ventilation turnover period of the control room, approximately 90 minutes, based on a control room volume of 89,890 cubic feet and a makeup rate of 1000 cfm. Therefore, the radionuclide concentration in the control room would reach equilibrium with the outside atmosphere. Consequently, a change in the amount of unfiltered makeup or inleakage would have little impact on the dose consequences.

Therefore, I&M considers that assuming only 1000 cfm of unfiltered makeup for accidents where the CREVS is not operated in the emergency mode is reasonable and there is no need to consider unfiltered inleakage.

NRC Question 5

For accidents where the CREVS is in the emergency lineup, your submittal assumes 98 cfm of unfiltered inleakage. Please clarify why the 98 cfm of unfiltered inleakage for Unit 2 is limiting following the damper repair in Unit 1. It is not clear how the 98 scfm due to damper repair in Unit 1 was obtained.

I&M Response to NRC Question 5

The response to NRC Question 5, provided in Reference 3, described the basis for assuming an unfiltered inleakage of 98 cfm. As stated in Reference 3, that response applies to all the proposed analyses that assume operation of the CREVS in the emergency mode.

NRC Question 6

On page B3/4 7-4a of your submittal, operability is defined by maintaining a positive pressure of greater than or equal to 1/16 inch water gauge relative to the outside atmosphere. However, industry test results have determined that pressurization (at any level i.e. 1/16, 1/8, etc.) does not demonstrate control room envelope/pressure boundary operability.

a) *Provide the justification for your proposed TS changes defining control room envelope/pressure boundary operability based on 1/16 inch water gauge pressure relative to the outside atmosphere.*

b) *The requested 24-hour allowed outage time (AOT) is tied to the definition of control room envelope/pressure boundary operability. In order for the Nuclear Regulatory Commission (NRC) staff to find the request for a 24-hour AOT acceptable, the request must be in accordance with the Technical Specification Task Force-287 (TSTF-287), which has been generically approved by the staff. Note, TSTF-287 does not include a definition of control room boundary integrity.*

I&M Response to NRC Question 6(a)

Page B3/4 7-4a is a Unit 2 TS Bases page. The changes proposed in the original amendment request submitted by Reference 1 included addition of a statement to this page that the control room envelope/pressure boundary could be considered operable if it could be maintained at a positive pressure of greater than or equal to 1/16 inch water gage relative to the outside atmosphere. An identical statement was included in the proposed changes to Unit 1 TS Bases Page B3/4 7-5.

As described in the response to NRC Question 6(b) below, the proposed change to the Unit 1 and Unit 2 TS has been revised to eliminate the statement indicating that the control room envelope/pressure boundary can be considered operable if it can be maintained at a positive pressure of greater than or equal to 1/16 inch water gage relative to the outside atmosphere.

I&M Response to NRC Question 6(b)

The proposed change to the Unit 1 and Unit 2 TS has been revised to be in accordance with TSTF-287. The TS pages affected by the revision, marked to show all currently proposed changes to the pages, are provided in Attachments 2A and 2B for Unit 1 and 2, respectively. Attachments 3A and 3B show these TS pages with all currently proposed changes to the pages incorporated. The revised pages in Attachments 2A, 2B, 3A, and 3B include editorial changes that do not affect requirements or intent, and changes to the Bases that have been made under the provisions of 10 CFR 50.59.

The revision affects the previously proposed change to the TS 3/4.7.5.1 Limiting Conditions for Operations (LCO) and Actions, and the Bases for TS 3/4.7.5 as follows:

TS 3/4.7.5.1 LCOs

The proposed change to TS 3/4.7.5.1 has been revised to eliminate a previously proposed new LCO, 3.7.5.1.c. This LCO required that the control room envelope/pressure boundary be operable. This LCO is not needed since, consistent with TSTF-287, the requirement in LCO 3.7.5.1.a that two independent CREVS pressurization trains be operable cannot be met if the control room envelope/pressure boundary is inoperable. This part of the revision to the previously proposed change is administrative since the control room envelope/pressure boundary must still be operable to satisfy the LCO.

The proposed change to TS 3/4.7.5.1 has also been revised to include a note stating that the control room envelope/pressure boundary may be opened intermittently under administrative control. In the previously proposed change, this allowance had been provided in the Bases for this specification. Providing this allowance via a note in the LCO is consistent with the structure specified by TSTF-287. This part of the revision is administrative since the allowance for the control room envelope/pressure boundary to be opened intermittently previously existed in the proposed change to the Bases.

TS 3/4.7.5.1 Actions

The previously proposed new Action "c" TS 3/4.7.5.1 has been revised such that the Action addresses inoperability of the pressurization trains due to inoperability of the control room envelope/pressure boundary, rather than addressing inoperability of the control room envelope/pressure boundary directly. This is consistent with TSTF-287 and the above noted revision of the proposed change to LCO 3.7.5.1.c. The requirement in the previously proposed Action "c" that the unit be in cold shutdown within 30 hours has not been revised, even though it is more restrictive than the 36 hour period specified in TSTF-287. The 30-hour period is consistent with the existing Action requirements for an inoperable pressurization fan or an inoperable filter. This part of the revision is administrative since the revised Action "c" requires the same measures as the previously proposed Action "c" if the control room envelope/pressure boundary is inoperable.

TS 3/4.7.5 Bases

The proposed change to the Bases for TS 3/4.7.5 has been revised to eliminate the statement that the control room envelope/pressure boundary can be considered operable if it can be maintained at a positive pressure of greater or equal to 1/16 inch water gage relative to the outside atmosphere. This revision reflects the potential for the CREVS to be rendered inoperable by conditions other than the inability to meet the 1/16 inch criterion, such as an opening in the control room envelope/pressure boundary without the specified administrative controls. I&M will continue to consider the CREVS inoperable if the 1/16 inch water gage pressure requirement is not met, since TS Surveillance Requirement 4.7.5.1.e.3 will continue to require periodic verification that the CREVS maintains this pressure. This part of the revision makes the proposed change more restrictive.

The proposed change to the Bases for TS 3/4.7.5 has also been revised by relocating the allowance to have the control room envelope/pressure boundary open intermittently under administrative controls to a note in the LCO as described above. This part of the revision of the previously proposed change is administrative.

The proposed change to the Bases for TS 3/4.7.5 has also been revised to make the description of the Actions for an inoperable control room envelope/pressure boundary consistent with that specified in TSTF-287. The revised description contains a discussion of preplanned compensatory measures that was not included in the previously proposed change. This description is consistent with that provided in TSTF-287 with one exception. TSTF-287 identifies several potential hazards, including toxic chemicals. The proposed change to the CNP Bases does not include toxic chemicals in the examples because, as documented in Items 5 and 8 of Attachment 7 to Reference 1, an evaluation has demonstrated that there is no need for toxic gas protection for personnel in the control room because of the low probability of a toxic gas related event. In accordance with TSTF-287, I&M commits to have written procedures available describing the compensatory measures to be taken in the event that the control room envelope/pressure boundary is inoperable in Modes 1, 2, 3, and 4. This part of the revision renders the proposed change more restrictive.

As noted above, the various parts of the revision to the previously proposed change are either administrative or render the specification more restrictive. The revision does not involve new plant equipment or operation of existing plant equipment in a new or different manner. The revision does not affect or create new accident initiators or precursors, and does not affect safety margins. Therefore, the No Significant Hazards Consideration Evaluation provided, as Attachment 4 to Reference 1 remains valid.

NRC Question 7

In numerous locations, your submittal references NUREG-1465 and Draft Guide-1081 as basis for your submittal. Please provide a commitment to the applicable provisions of Regulatory Guide (RG) 1.183, in lieu of the NUREG-1465 and DG-1081 referenced in your submittal, identifying proposed alternatives, if any, for staff consideration.

(The staff used some information from NUREG-1465 as part of the basis for the development of the regulatory guidance in DG-1081 and the final RG 1.183. However, the staff has not endorsed NUREG-1465 for use by currently licensed power reactors since NUREG-1465 is not specifically applicable to currently licensed power reactors, especially those with fuel burnups in excess of 40 GWD/MTU. It is the staff's intent that the guidance of RG 1.183 be used by licensees in preparing their initial application under 10 CFR 50.67 and that guidance, less any approved alternatives, would become the facility's alternate source term (AST) design-basis.)

I&M Response to NRC Question 7

The response to NRC Question 8 below describes the significant differences between the assumptions used in the proposed analysis provided by Reference 1 and those identified in RG 1.183. These differences constitute proposed alternatives to the RG 1.183 provisions. I&M commits to the applicable provisions of RG 1.183, dated July 2000, except for the proposed alternatives identified in the response to NRC Question 8.

NRC Question 8

DG-1081 was published for public comment in December 1999, and the final guide RG-1.183 was issued in July 2000. Your submittal was dated June 2000. In addressing the public comments and preparing the final guide, several analysis assumptions in DG-1081 were revised. As such, some assumptions identified in your submittal differ from those deemed acceptable in RG 1.183. For many of these differences, the staff believes that your submitted analyses could be shown to be bounding using the outdated assumption, and as such, it may be possible to incorporate the updated assumption in your design-basis without resubmitting the analysis. Please compare your analysis assumptions against those provided in RG 1.183 and indicate your intent to either update the assumption or retain the assumption as a proposed alternative to RG 1.183. Provide a justification for each such proposed alternative.

I&M Response to NRC Question 8

The differences between the assumptions used in the proposed accident analyses provided by Reference 1 and those identified in RG 1.183 are provided below for each analyzed accident except a FHA. The differences in the FHA analysis assumptions were provided in Reference 3. The differences identified below and those identified in Reference 3 constitute proposed alternatives to the assumptions identified in RG 1.183. I&M does not intend to update the proposed analyses in support of the requested control room habitability license amendment. As indicated below, I&M may eliminate some of the differences between the assumptions used in the proposed accident analyses and those identified in RG 1.183 if the associated analysis is re-performed in the future.

Large Break LOCA Analysis

The proposed analysis included the assumption of a gross failure of a passive component in the ECCS recirculation line at 24 hours into the accident, resulting in 50 gpm release over a 0.5 hour period. The guidance in RG 1.183 does not include such an assumption. Therefore, the assumption in the proposed analysis is more conservative than is specified by RG 1.183. If this analysis is re-performed in the future, I&M may eliminate this assumption consistent with RG 1.183.

The proposed analysis used an iodine airborne coefficient of 10^{-4} for the ECCS recirculation leakage. This value is significantly lower than the value of 0.1 stated in RG 1.183 as a default value. Therefore, the assumption in the proposed analysis is less conservative than that specified by RG 1.183. This difference is addressed in the response to NRC Question 10.

In the proposed analysis, all the iodine released from postulated leaks in ECCS recirculation piping was assumed to be in the elemental form. RG 1.183 indicates that the iodine should be assumed to be 97% elemental and 3% organic. This difference did not affect the results of the proposed analysis since the assumed charcoal adsorber efficiencies for elemental and organic iodine forms were identical and there was no other iodine removal mechanism credited that differentiates between the two iodine forms. Therefore, the assumption in the proposed analysis is neither more or less conservative than is specified by RG 1.183. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

In the proposed analysis, it was assumed that the containment leaked at the design basis rate for the first 280 hours of the accident, followed by a 50% reduction in the assumed leak. The RG 1.183 guidance indicates that the assumed leak rate can be reduced by 50% at 24 hours. The assumption of a higher containment leak rate for a longer period results in higher calculated doses. Therefore, the assumption in the proposed analysis is more conservative than is specified by RG 1.183.

In the proposed analysis, it was assumed that, of the 5% iodine activity from the fuel that is in the gap, 3% was released from 30 seconds to 90 seconds and the remaining 2% was released over the next 28.5 minutes. This differs from the assumption provided in RG 1.183 that the activity would be released from the core in a linear fashion over the duration of the release phase, or as an alternative, released instantaneously at the start of the particular release phase. The basis for the assumption in the proposed analysis is provided in the response to NRC Question 14. If this analysis is re-performed in the future, I&M may use one of the assumptions provided in RG 1.183.

Small Break LOCA Analysis with No Containment Spray

A small break LOCA without containment spray actuation is not specifically addressed in RG 1.183. The proposed analysis was included for the reasons described in Reference 1. In the proposed analysis, the calculation of activity releases due to containment leakage is consistent with the guidance of RG 1.183 for large break LOCAs with the following differences.

In the proposed analysis, the activity released from the core was assumed to be limited to that within the fuel gap. No "in-vessel" release (release from damaged fuel pellets) was assumed. This is discussed further in the response to NRC Question 20.

In the analysis of iodine releases through the secondary coolant pathway, the iodine was modeled as being entirely in the elemental form. The guidance of RG 1.183, Appendices E, F, and G indicates that the iodine should be considered to be 97% elemental and 3% organic. The assumed iodine form has no impact on the results of the proposed analysis since the charcoal adsorber efficiencies for elemental and organic iodine forms are the same and there were no other iodine removal mechanism modeled in the proposed analysis that would differentiate between the two iodine forms. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

In the proposed analysis, it was assumed that the containment leaked at the design basis rate for the first 280 hours of the accident, followed by a 50% reduction in the assumed leak. The RG 1.183 guidance indicates that the assumed leak rate can be reduced by 50% at 24 hours. The assumption of a higher containment leak rate for a longer period results in higher calculated doses.

Since the RG does not address a small break LOCA, it is not appropriate to designate these differences as more conservative or less conservative.

Main Steam Line Break Analysis

In the proposed analysis, it was assumed that the duration of the accident initiated iodine spike was limited to 6 hours. The guidance in RG 1.183 indicates an accident initiated spike duration of 8 hours should be assumed. The basis for the assumption in the proposed analysis is given in the response to NRC Question 13.

In the proposed analysis, a gap fraction of 12% for I-131 was assumed in determining the duration of the accident-initiated iodine spike. This is greater than the gap fraction value of 8%

given in Table 3 of RG 1.183. The basis for the assumption in the proposed analysis is given in the response to NRC Question 13.

Steam Generator Tube Rupture Analysis

In the proposed analysis, an accident-initiated iodine spike of 500 times the identified equilibrium iodine release rate was assumed. The guidance in RG 1.183 indicates that the spike should be assumed to be 335 times the identified equilibrium iodine release rate. Since use of the higher factor increases the calculated coolant activity levels more rapidly, the assumption in the proposed analysis is more conservative than is specified by RG 1.183. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

In the proposed analysis, it was assumed that the duration of the accident initiated iodine spike was limited to 6 hours. The guidance in RG 1.183 indicates an accident initiated spike duration of 8 hours should be assumed. The basis for the assumption in the proposed analysis is given in the response to NRC Question 13.

In the proposed analysis, a gap fraction of 12% for I-131 was assumed in determining the duration of the accident-initiated iodine spike. This is greater than the gap fraction value of 8% given in Table 3 of RG 1.183. The basis for the assumption in the proposed analysis is given in the response to NRC Question 13.

Locked Rotor Analysis

The radiological consequences of this event were not analyzed since I&M has determined that no fuel rods would exceed the departure from nucleate boiling (DNB) limit for the current Unit 1 and Unit 2 operating cycles. Additionally I&M has committed to conduct cycle specific analyses to demonstrate that a locked rotor event would not result in control room doses that exceed the limits of 10 CFR 50.67. Appendix G to RG 1.183 states that, if no fuel damage is postulated, a radiological analysis is not required. Consistent with this guidance, I&M did not perform a radiological analysis of a locked rotor event.

The proposed analysis determined that, for CNP, the control room dose resulting from a locked rotor event would be bounded by the dose resulting from a loss of load event, which would be bounded by the dose resulting from a LOOP event. This differs from the RG 1.183 position that the dose from a locked rotor event would be bounded by dose from a steam line break event. The basis for this difference is provided below.

The steam line break analysis is typically chosen as the bounding analysis of accidents that would result in a radiological release from the secondary system because it credits no partitioning of iodine in the faulted steam generator, resulting in the largest release. However, the loss of load/LOOP analysis is more limiting for CNP because no credit is taken for CREVS actuation in the analysis. The CREVS actuation that is credited in the steam line break analysis would reduce the dose consequences in the control room.

The key parameters for the steam line break and loss of load/LOOP analyses are compared in the table below. The table also includes the assumptions that would be reasonable for the locked rotor event.

Parameter	Steam Line Break	Loss of Load/LOOP	Locked Rotor
Safety Injection (SI) Signal	Yes	No	No
CREVS Actuation	5 minutes	No	No
Steam Generator (SG) iodine partition factor	0.01 for intact SGs 1.0 for ruptured SG	0.01	0.01
Radiological release point	SG PORVs, for intact SGs Unit vent for ruptured SG	SG PORVs	SG PORVs
Control room dose consequences, Pre-accident iodine spike Control Room (rem TEDE)	0.11	0.4	Bounded by loss of load/LOOP
Control room dose consequences, Accident-initiated iodine spike Control Room (rem TEDE)	0.4	2.0	Bounded by loss of load/LOOP

The table shows that the locked rotor event is more like the loss of load/LOOP event than the steam line break and that the loss of load/LOOP event would result in the more limiting secondary side release for control room dose consequences.

Rod Ejection Analysis

The proposed analysis accounted for the release of alkali metals from the fuel while RG 1.183, Appendix H, Item 1 indicates that the only nuclide groups that need be considered are iodines and noble gases. However, in RG 1.183, Appendix H, Item 4, the iodine form is identified as being predominantly cesium iodide so it was assumed that the intent of RG 1.183 is to include the alkali metals. Since the intent of RG 1.183 was met, the assumption in the proposed analysis is neither more or less conservative than is specified by the RG.

Some of the fission product gap fractions used in the proposed analysis differ from those identified in RG 1.183 as shown in the table below.

	Proposed Analysis	RG 1.183	Factor of Conservatism in Proposed Analysis
I-131	12%	10%	1.2
Kr-85	15%	10%	1.5

As shown in the table, the gap fractions assumed in the proposed analysis for I-131 and Kr-85 are more conservative than those identified in RG 1.183. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

The proposed analysis assumed that 35% of the iodine activity from melted fuel would be available for release from the containment. This is more conservative than the RG 1.183 assumption that 25% of the iodine from the melted fuel would be available for release from the containment. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

In the proposed analysis, it was assumed that 35% of the iodine activity from melted fuel would be released to the reactor coolant, becoming available for leakage to the secondary side. This is less conservative than the RG 1.183 assumption that 50% of the iodine from the melted fuel would be released to the primary coolant. However, the contribution to the control room dose from iodine in the ECCS leakage pathway has been estimated to be less than 0.15%. The effect of the less conservative assumption in the proposed analysis is therefore negligible. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

In the proposed analysis, it was assumed that the containment leaked at the design basis rate for the first 280 hours of the accident, followed by a 50% reduction in the assumed leak. The RG 1.183 guidance indicates that the assumed leak rate can be reduced by 50% at 24 hours. The assumption of a higher containment leak rate for a longer period results in higher calculated doses.

In the proposed analysis, an average core activity in the damaged fuel was assumed. A core radial peaking factor was not applied to determine the radionuclide inventory of the damaged rods as stated in Section 3.1 of RG 1.183. The assumption in the proposed analysis was not as conservative as that in the RG in that use of the peaking factor would increase the calculated dose. However, use of the radial peaking factor would increase the calculated dose by no more than a factor of 1.65, since this is a bounding radial peaking factor for CNP. The proposed analysis includes an assumption that 15% of the fuel rods are damaged which is an increase of

50% over the value of 10% used in the current licensing basis analysis as identified in Unit 1 Section 14.2.6.19 and Unit 2 Section 14.2.6.3 of the UFSAR. This increase in the assumed level of fuel damage was made solely to increase the conservatism of the analysis. This increased the calculated dose by a factor of 1.50, offsetting all but 15% of the non-conservatism that resulted from omitting the 1.65 radial peaking factor.

As discussed in the above, the impact of not incorporating the 1.65 radial peaking factor in the rod ejection analysis was largely offset by the arbitrary increase in the assumed level of fuel damage. Further adjustments to the analysis to incorporate1) reduced gap fraction for I-131 from 12% to 10%, 2) reduced fuel melt release fraction to the containment for iodines from 35% to 25%, and 3) increasing the fuel melt release fraction to the primary coolant for iodines would result in an overall reduction in doses such that the doses would be approximately equal to those currently reported in the proposed analysis. If this analysis is re-performed in the future, I&M may use the assumption provided in RG 1.183.

NRC Question 9

Your analyses incorporated revised atmospheric dispersion (X/Q) values calculated using the ARCON96 computer code. The staff considers this to be a change in analysis methodology requiring staff approval. Please provide sufficient information for the staff to evaluate the acceptability of your X/Q values. The information should include:

a. *Confirmation that the meteorological data input to ARCON96 was collected by the site's meteorological instrumentation as described in the updated final safety analysis report (UFSAR) or T/S and subject to 10 CFR Part 50, Appendix B quality assurance requirements.*

b. *Unit 1 and Unit 2 release point and receptor configuration information (e.g., height, velocity, distances, direction, etc.), release mode (e.g., ground, elevated, surface), and meteorological sensor configuration, as input to ARCON96.*

c. *A floppy disk containing the meteorological data input to ARCON96, in the ARCON96 input data format.*

I&M Response to NRC Question 9

The requested information was provided by Reference 3. An error was subsequently identified in the processing of the stability data provided by Reference 3. The NRC staff was notified of the error on August 1, 2001. I&M will address the consequences of the error in a separate submittal by September 14, 2001.

NRC Question 10

Your analyses incorporated an iodine flashing fraction of 10^{-4} for emergency core cooling system (ECCS) leakage, contrary to the default 10^{-1} assumption provided in RG 1.183. On Pages 5 and 6 of Attachment 1 to your submittal, you attempted to justify these assumptions on an experiment reported in your existing UFSAR, and on theoretical iodine partitioning of 10^{-8}. The staff does not believe that the provided justification supports the use of 10^{-4} for the ECCS flash fraction. Based on the description of the experiment, the staff questions whether the experimental drying to evaporation can appropriately model leakage that could be sprayed from the leakage paths, or as droplets fall through air and impinge on nearby surfaces. The staff also questions how well Eggleton's mathematical treatment of steady state vapor partial pressures between the gas and liquid phases can adequately model the more dynamic situation associated with leakage from pressurized systems as is the case here. Your submittal quoted partitioning of 10^{-8} which appears to be at odds with the abstract for Eggleton work which reports partitioning values ranging from 0.012 at high iodine concentrations and low pH to less than 0.0001 at high pH and low iodine concentrations. Please provide additional justification, including consideration of sump pH and area ventilation rates and iodine entrainment in evaporated vapor, in support of your assumption.

I&M Response to NRC Question 10

The iodine flashing fraction or airborne fraction is the fraction of iodine contained in liquid water that becomes airborne. An airborne fraction of 10^{-4} was used in the proposed large break LOCA analysis to determine the iodine released from ECCS leakage at temperatures below 212°F. The value of 10^{-4} used in the proposed analysis is smaller than the 0.10 value given in RG 1.183, Appendix A, Paragraph 5.5.

I&M considers that the guidance in RG 1.183 allows use of alternative airborne fractions since Paragraph 5.5 of Appendix A to RG 1.183 states that smaller airborne fractions can be justified based on the actual sump pH history and area ventilation rates. The value of 10^{-4} used in the proposed analysis is the same value used in the current licensing basis large break LOCA analysis as documented in Unit 1 Section 14.3.5.19 and Unit 2 Section 14.3.5.7 of the UFSAR. Use of the 10^{-4} value was also documented in the original FSAR which was used by the NRC for initial issuance of the operating license. I&M considers that use of the 10^{-4} value remains justified based on the discussions provided below.

As described in Unit 1 Section 14.3.5.19 of the UFSAR, the airborne fraction of 10^{-4} was experimentally derived. The experiment was performed using several conservative conditions to maximize the airborne fraction. In the experiment, a sample of boric acid and sodium hydroxide solution was "spiked" with molecular iodine. Use of molecular iodine was conservative as compared to particulate iodine because particulates are less volatile. The sample was evaporated to dryness in moving hot air. Use of moving hot air was conservative since stagnant air and partial evaporation would reduce the measured volatility. Use of moving hot air promotes removal of iodine as it is released from the leakage. Continuous removal of the released iodine keeps the airborne concentration low thereby maximizing the driving force for release of additional iodine. Additionally, hot air flowing over a sample solution will remove airborne iodine more effectively than general building ventilation.

The experiment was conducted with the sample at a temperature of 200°F. This is conservative since, as stated in Section 6.1 of the UFSAR, the containment sump is expected to be no more than 190°F during the recirculation phase of a LOCA. As noted above, the experiment involved evaporating the sample solution to dryness. This addresses the concern regarding modeling of sprayed leakage impinging on nearby surfaces. A pool of solution has a smaller surface area than a sprayed solution or droplets of solution impinging on solid surfaces. The pool of solution therefore has a lower rate of evaporation and rate of iodine release. However, with either a spray or a pool, the release will end when all of the liquid evaporates. By evaporating the solution to dryness, the experiment bounded both the spray and pool forms of the solution even though it did not physically model the spray form. Additionally, no credit was taken in the proposed analysis for un-evaporated pools of water or collection of water by floor drains.

The NRC question also indicated concern that Eggleton's mathematical treatment of steady state vapor pressures did not model dynamic effects. However, the airborne fraction used in the proposed analysis was based on the conservative experimental results as described above, rather than the Eggleton study. The relevance of the Eggleton study is discussed in subsequent paragraphs.

Iodine Entrainment in Evaporated Vapor

The preceding discussions apply to volatile iodine. It is possible for particulate iodine to be entrained in the evaporated water vapor. The energy of a spray or droplet impinging on a surface may enhance entrainment mechanisms. The experiment did not duplicate this mechanism because it used a pool of solution as the iodine source. Therefore, the proposed analysis may not bound iodine entrainment. However, if iodine were entrained in the auxiliary building, the concentration would be reduced by natural deposition in the building and ventilation ductwork prior to release. Highly efficient (99%) HEPA filters in the ESFVS would also remove the majority of the particulates prior to release.

Eggleton Study

The Eggleton study, "A Theoretical Examination of Iodine-water Partition Coefficients," Atomic Energy Research Establishment, Harwell (England), AERE-R 4887, 1967, was based on equilibrium conditions between a static liquid and a static gas. As described below, that study showed that a partition fraction of 10^{-8} could theoretically be expected. The study, therefore, confirms that the airborne fractions measured in the experiment are theoretically possible. I&M recognizes that the study does not address dynamic effects, such as ventilation and mechanical entrainment of iodine into the air. Therefore, the airborne fraction of 10^{-4} used in the proposed analysis was larger than those predicted by the Eggleton study to account for dynamic effects.

The Eggleton study determined equilibrium iodine partition coefficients, (i.e., g-iodine per liter water / g-iodine per liter gas) as a function of temperature, iodine concentration in the water, pH value, and whether an iodate reaction can be credited. The effect of these variables and the values that apply to CNP are described below

Temperature – More iodine would be released to the atmosphere at lower solution temperatures. The Eggleton report provided two graphs showing the partition coefficient based on the variables involved. One graph presented the partition coefficient for a solution temperature of 25°C and another graph presented the partition coefficient for a solution temperature of 100°C. The graphs are reproduced below. The post-LOCA containment sump water temperature is expected to be between 139°F and 190°F. This corresponds to 59°C to 88°C. Therefore, use of the graph for a 25°C solution is conservative.

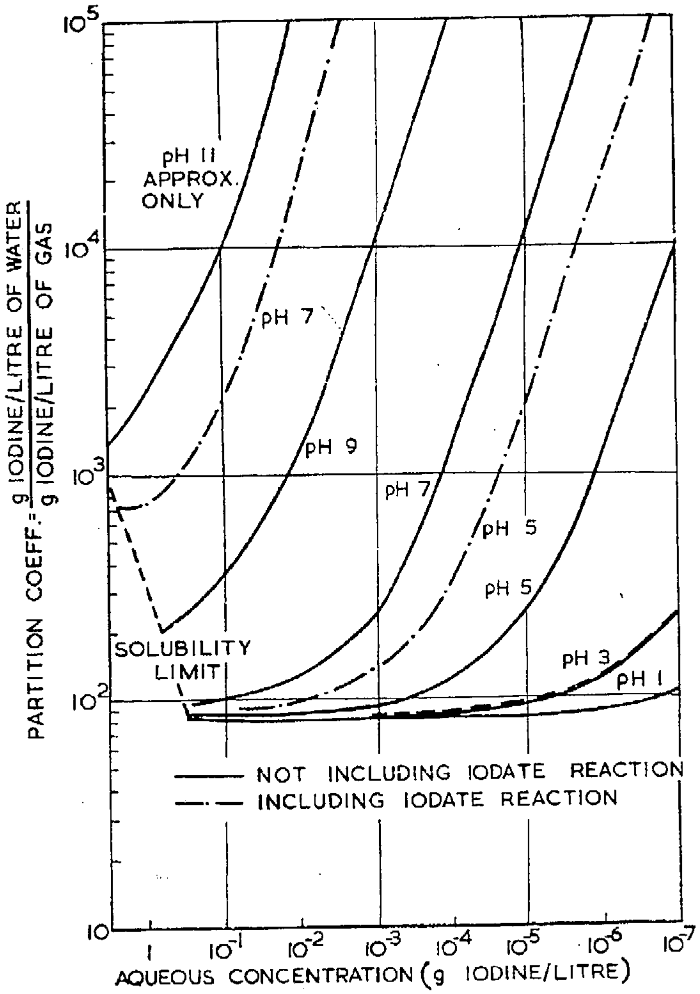


FIG.1. PARTITION OF IODINE BETWEEN GAS PHASE AND WATER AT 25°C

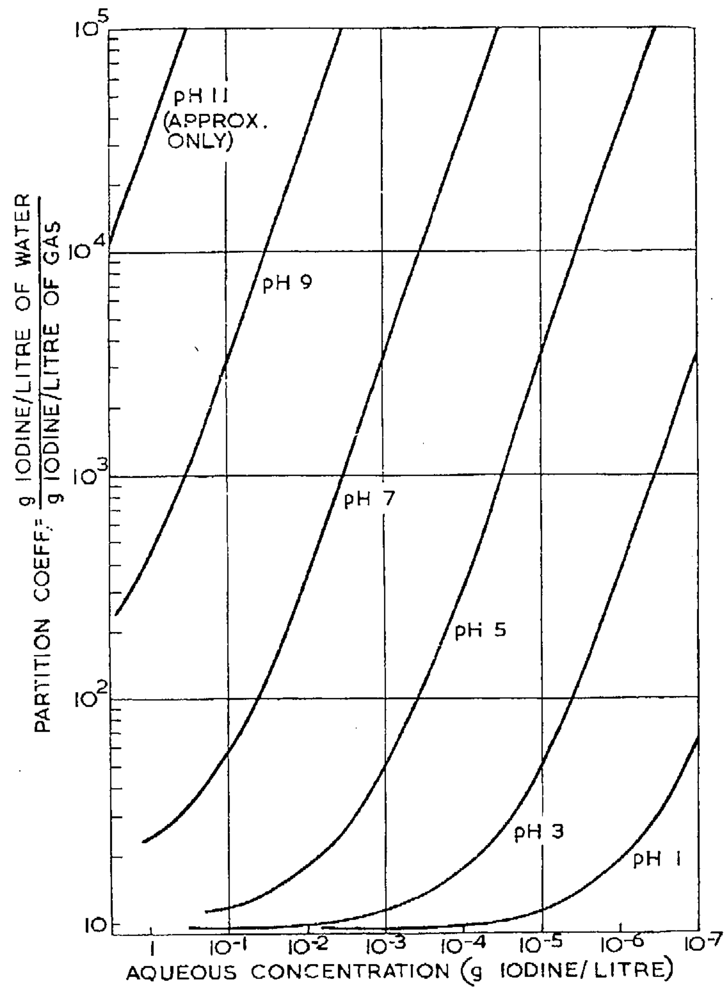


FIG.2. PARTITION OF IODINE BETWEEN GAS PHASE AND WATER AT 100°C

Iodine Concentration - More iodine would be released to the atmosphere at higher sump iodine concentrations. Using the total core iodine activities and the decay constants from Tables 4 and 6 of Attachment 6 to Reference 1, the total core inventory would be approximately 1080 grams of iodine. As indicated in Table 2 of RG 1.183, an early in-vessel release would result in 35% of the iodine being released from the core. Assuming all of this 35% was in the sump, the sump would contain approximately 380 grams of iodine. As indicated by item L25 of the design information transmittal (DIT), DIT-B-00069-06, that was included with Attachment 6 to Reference 1, the minimum sump inventory at switchover to the sump recirculation phase would be 216,000 gallons, or conservatively, approximately 817,000 liters. Therefore the maximum sump iodine concentration would be is approximately 0.0005 grams-iodine/liter. As indicated by item L24 of DIT-B-00069-06, the minimum sump inventory after all ice is melted would be 514,642 gallons. Consequently, the long term steady state iodine concentration would be less than half the 0.0005 grams-iodine/liter value. Therefore, use of a sump iodine concentration of 0.0005 grams-iodine/liter ($5x10^{-4}$ grams-iodine/liter) with the Eggleton graph is conservative.

pH Value- More iodine would be released to the atmosphere at lower pH values. I&M has determined that the minimum sump pH at the time of switchover to sump recirculation would be 7.6, and the minimum pH would increase at the end of ice melt would be 8.1. Therefore, use of a pH value of 7 with the Eggleton graph is conservative.

Iodate Reaction - At elevated temperatures, iodine would react with aqueous alkaline solutions to produce iodate (IO^{-3}). Since iodate is less volatile than elemental iodine, iodate production would reduce the release of iodine. As noted above, the post-LOCA containment sump water temperature is expected to be 59°C to 88°C. At 60°C and a pH value of 7, the iodate reaction approaches completion at one hour per Figure 18 of NUREG/CR-2900, "Predicted Rates of Formation of Iodine Hydrolysis Species at pH Levels, Concentration, and Temperatures Anticipated in light water reactor (LWR) Accidents," J. T. Bell, et al, October, 1982. Considering the duration of the radiological release (30 days), a one hour iodate reaction time is relatively short. Therefore, it is appropriate to credit the iodate reaction in using the Eggleton graph.

Conclusion – Using the Eggleton graph for 25°C, an iodine concentration of $5x10^{-4}$ grams-iodine/liter, a pH value of 7, and crediting the iodate reaction, the partition coefficient would be higher than the upper range of the chart, which is 10^5. Therefore, a steady state partition fraction (which is the inverse of the partition coefficient) as small as 10^{-8} is reasonable.

Based on the theoretical results of the Eggleton study of static conditions and the experimental results described above to account for dynamic effects, I&M considers that continued use of the current licensing basis airborne fraction of 10^{-4} is justified.

NRC Question 11

Your analyses addresses a small break loss-of-coolant accident (LOCA) event in which containment sprays do not start or are terminated early. Page 11 of 30 of DIT-B-00069-06 contains a note that states:

Per DG-1081 Appendix A, gap fractions from Table 3 can be used for small-break loss-of-coolant accident (SBLOCA) if no fuel melt is projected.

While this provision may have been present in a pre-decisional version of the draft guide, this provision was not included in the draft guide published for public comment in December 1999, nor in the final regulatory guide published in July 2000. While the staff agrees with the conclusion that the fuel damage could be less than that assumed for a large-break LOCA, the staff expects the licensee to provide a technical justification for the amount of fuel damage being assumed. Please provide an acceptable basis for this conclusion. See § 3.6 of RG 1.183.

I&M Response to NRC Question 11

This question is closely related to NRC Question 20. Therefore, a single response for both questions has been provided under NRC Question 20 below.

NRC Question 12

On Page 7 of Attachment 1, you note your conclusion that the assumption of a constant break flow for 30 minutes is more limiting than using the actual operator response times. Although this assumption may be valid with regard to mass of reactor coolant system (RCS) transferred to the secondary, what is the sensitivity of other analysis parameters to delays in operator actions, such as break flow flashing fraction, steam release from the affected steam generator, and tube uncovery? The staff is concerned that these other parameters, and the time-dependent buildup of RCS activity due to iodine spiking, could negate the apparent conservatism in the RCS mass transferred. Please confirm your conclusion relative to the postulated dose to the control room operators. Please explain how your amendment request dated October 24, 2000, on steam generator tube rupture (SGTR) analysis methodology affects this control room amendment request.

I&M Response to NRC Question 12

As documented in Unit 1 Section 14.2.4.3 and Unit 2 Section 14.2.4.3 of the UFSAR, the current licensing basis methodology for determining the radiological consequences of an SGTR is based on the assumption of a constant break flow lasting 30 minutes. The amendment request dated October 24, 2000, requested approval of a methodology using operator actions for SGTR recovery based on plant-specific simulator studies, and using the break flow model and associated LOFTRR2 computer code described in Westinghouse WCAP-10698-P-A for determining SGTR break flow. The methodology proposed in the October 24, 2000, amendment request more accurately predicts the response to a SGTR.

The mass releases calculated by the current licensing basis methodology and the methodology proposed in the October 24, 2000, amendment request are tabulated below. Only the mass releases up until break flow termination are tabulated, since these are the critical values for the dose analyses. The mass releases calculated by the current licensing basis methodology bound both units. The mass releases calculated for Unit 2 by the methodology proposed in the October 24, 2000, amendment request are higher than those calculated for Unit 1, so the Unit 2 values will be used for comparison with those calculated by the current licensing basis methodology.

SGTR MASS RELEASE COMPARISON

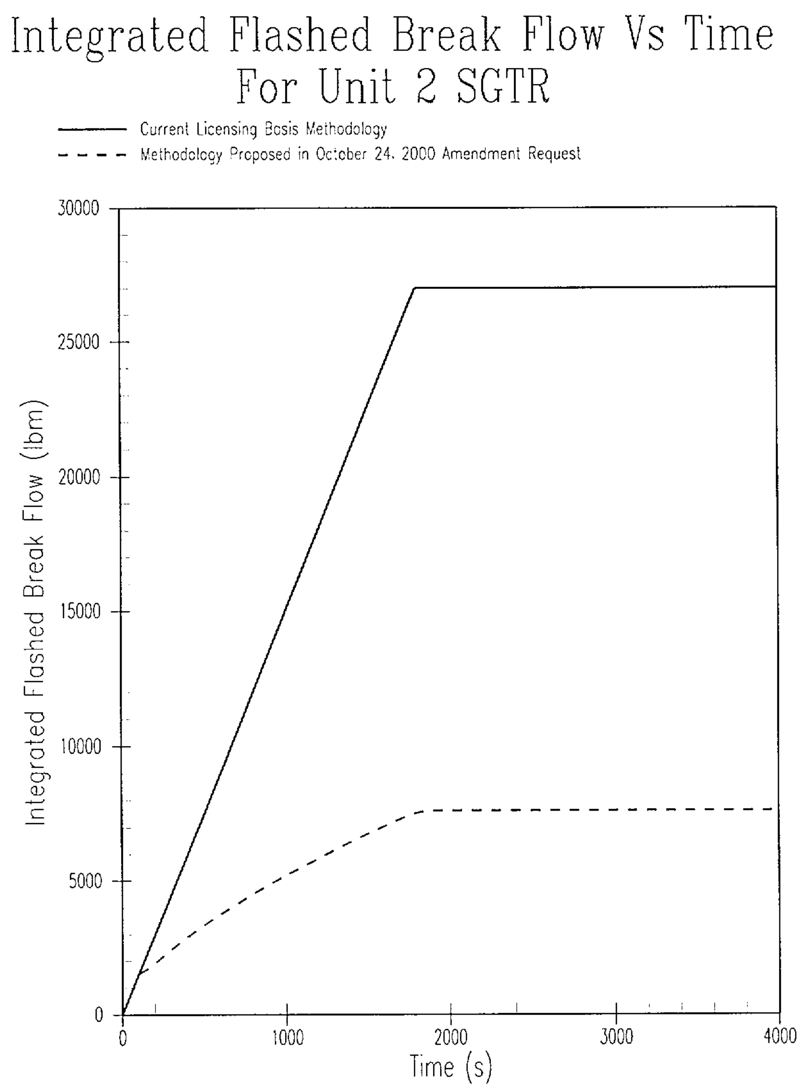
Source of Mass Release	Current Licensing Basis Methodology	Methodology Proposed in Oct. 24, 2000 Am. Request - Unit 1	Methodology Proposed in Oct. 24, 2000 Am. Request – Unit 2
Integrated Flashed Break Flow (lbm)	27000	6920	7617
Ruptured SG Release (lbm)	73000	36970	48500
Integrated Break Flow (lbm)	162000	184500	186100
Intact SG Release (lbm)	313889	298200	341600

The differences in the mass release values determined by the two methodologies are discussed below with respect to their effect on the calculated doses resulting from iodine and noble gasses.

<u>Dose Resulting from Iodine</u>

Integrated Flashed Break Flow

In the above table, the Integrated Flashed Break Flow is the portion of the integrated break flow that flashes to steam. The values given in the above table are graphically illustrated in the following figure.



The proposed SGTR dose analysis models flashed break flow as a direct release of iodine from the RCS to the atmosphere with no partitioning in the secondary side. Therefore, the release of flashed break flow would be the largest contributor to the iodine doses. As indicated in the above table, the mass released by flashed break flow as calculated by the current licensing basis methodology is more than 3.5 times that calculated by the methodology proposed in the October 24, 2000, amendment request. Consequently, the current licensing basis methodology provides the more limiting dose results.

Ruptured SG Release

In the above table, the Ruptured SG Release is the total mass released from the ruptured steam generator to the atmosphere. The higher (~50%) ruptured SG release calculated by the current licensing basis methodology is more limiting than that calculated by the methodology proposed in the October 24, 2000, amendment request.

Integrated Break Flow and Intact SG Release

In the above table, the Integrated Break Flow is the total mass of primary coolant released to the ruptured steam generator. The Intact SG Release is the total mass released from the intact steam generators to the atmosphere. The values for integrated break flow and intact SG release determined by the methodology proposed in the October 24, 2000, amendment request are higher by approximately 15% and 9% than the values determined by the current licensing basis methodology. However, these are more than offset by the higher values for integrated flashed break flow and ruptured steam generator steam releases calculated by the current licensing basis methodology.

Iodine Spiking

The NRC question included a concern regarding the effect on dose of a time-dependent buildup of RCS activity due to iodine spiking and the possibility that the buildup negates conservatism in the RCS mass transferred. Iodine spiking adds iodine to the RCS at a constant rate over the duration of the spike. For the accident-initiated iodine spike, the evolution rate is 500 times the normal rate and the duration is 6 hours. Therefore, the buildup of RCS activity would begin immediately and increase to a theoretical maximum where the appearance rate would equal the release rate. It is conservative to assume that the increase in RCS activity would be linear.

A key difference between the two methodologies is the duration of the break flow, 30 minutes for the current licensing basis methodology and almost 60 minutes for the methodology proposed in the October 24, 2000, amendment request. The concern appears to be that, if the RCS activity buildup is significant, then the amount of iodine released may be larger for the longer break flow duration even though the total RCS mass transferred is smaller. In other words, a smaller break flow at a higher iodine concentration may result in a larger iodine release.

However, the buildup of iodine activity in the RCS does not negate the conservatism in the value for RCS mass transferred. Most of the iodine would be released due to flashing of break flow and no credit is taken for partitioning in the steam generator. As the figure above shows, flashing of break flow would terminate at about 30 minutes for both cases. After flashing is terminated, activity contained in the break flow would be mixed with the secondary coolant, and subject to partitioning prior to release to the atmosphere. A partitioning factor of 0.01 was used in determining the release. RCS activity would have to increase by a factor on the order of 100 for these releases to be significant. Using the conservative assumption that RCS activity increases linearly, the RCS activity would no more than double over the second 30 minutes of the event. Doubling the RCS activity is a small factor compared to the effect of the iodine partitioning coefficient. Thus, there is no concern that the prolonged break flow would be more limiting for the accident-initiated iodine spike due to the buildup of activity in the RCS.

Dose Resulting from Noble Gasses

The noble gas doses would not be dependent on the integrated flashed break flow, the ruptured steam generator release, or the intact steam generator release. Thus, a higher integrated break flow would increase the total noble gas dose proportionally. From the integrated break flow data in the above table, the noble gas doses that would result from the methodology proposed in the October 24, 2000, amendment request would be 15% higher than those calculated by the current licensing basis methodology. However, this is more than compensated for by the conservatism in the iodine release values due to the higher integrated flashed break flow calculated by the current licensing basis methodology. The TEDE dose is the sum of the dose from noble gases released and iodines released (which depends on the initial activity and spike assumptions). The proposed SGTR dose analysis addresses two iodine spiking cases. Less than 8% of the TEDE doses determined by the proposed analysis was from noble gas releases. Since the noble gases are a minor contributor to the total TEDE dose, a 15% increase in noble gas results would be bounded.

Based on the above, the proposed SGTR dose analysis would remain bounding for either the current licensing basis methodology assumption of a constant break flow lasting 30 minutes or the methodology described in I&M's amendment request dated October 24, 2000. Therefore, the amendment request dated October 24, 2000 does not affect the proposed SGTR dose analysis and the current analysis methodology will continue to be the licensing basis methodology used for determining doses from a SGTR.

Tube Uncovery

The NRC question included a concern regarding steam generator tube uncovery. The issue of tube bundle uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible for the limiting SGTR transient. The WOG program concluded that the steam generator tube uncovery issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor Systems Branch, to Lawrence A. Walsh, Chairman of the Westinghouse Owners Group. The letter states "... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue resolved." Consistent with this position, the proposed SGTR dose analysis did not model tube uncovery.

NRC Question 13

Contrary to the guidance of RG 1.183, in some of your analyses you have assumed an iodine spike duration of 6 hours based on the depletion of the 12 percent iodine gap inventory. The iodine spiking phenomenon is generally understood to be the result of RCS liquid flushing out suspended iodine salts from the fuel rod via pin hole leakage. The transfer of iodine from the pellet to the plenum region is dependent, in part, on partial pressures of iodine in the gap and the pellet. In light of these considerations, please explain why basing your assumption on the gap inventory alone is appropriate.

I&M Response to NRC Question 13

While the iodine spiking phenomenon is understood to be the result of RCS liquid flushing the iodine salts that are in the fuel rod gap, there would be no continued diffusion of iodine from the fuel pellets into the gap region after reactor trip. Once reactor trip occurs there is such a large drop in fuel temperature that there is no significant driving force to transport fission products out of the fuel pellets.

The determination of 6 hours for the termination of the iodine spike was based on depletion of the iodine inventory in the fuel rod gap assuming that 12% of the iodine was in the gap. From Table 3 of RG 1.183, the assumed gap fraction should be 8%. The use of an 8% gap fraction would reduce the duration of the iodine spike to about 4 hours. The proposed analyses for a SGTR, a steam line break, and a LOOP conservatively assumed that the spike would last for 6 hours. The proposed analyses also conservatively assumed that all activity in the gap would be released to the RCS. This is conservative because the pinhole defect would restrict the release and a large portion of the gap activity would remain within the fuel rods.

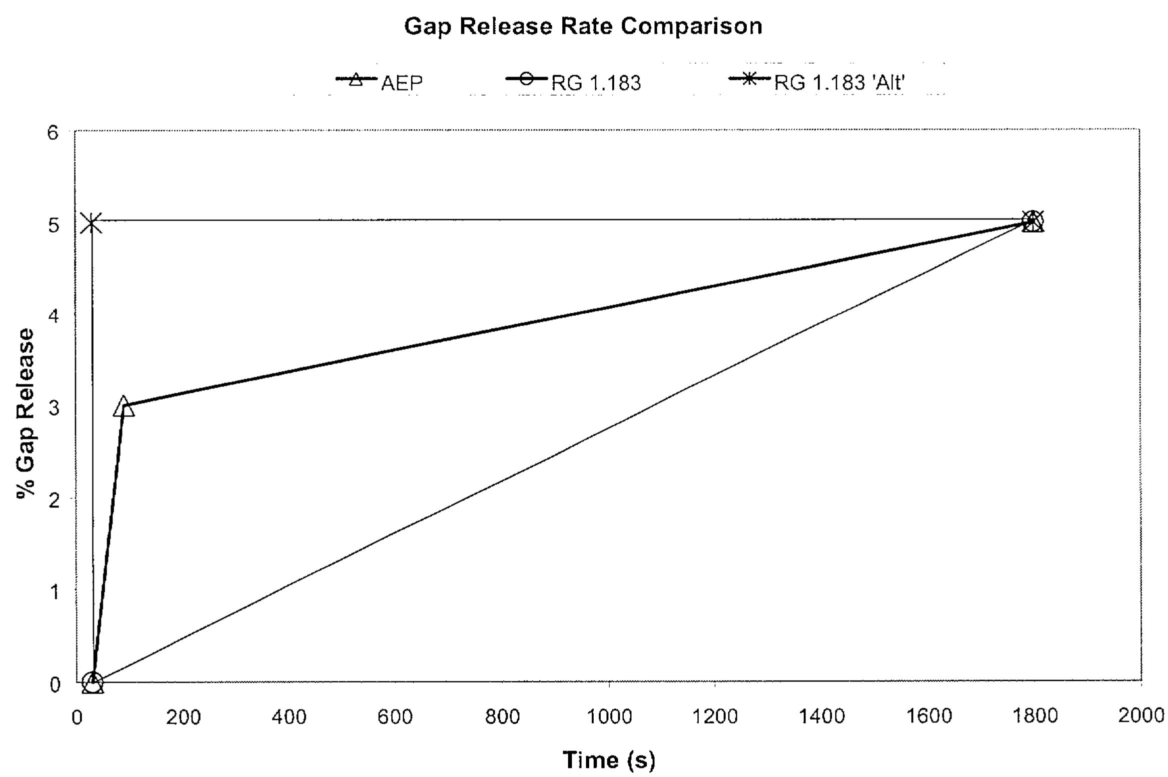
NRC Question 14

§3.1.1 of Attachment 6, identifies the assumption that 3 percent of the gap activity is released from 30 seconds to 90 seconds and the remaining 2 percent of the gap is released over the next 28.5 minutes. RG 1.183 (and DG-1081) provided that the activity would be released from the core in a linear fashion over the duration of the release phase, or as an alternative, released instantaneously at the start of the particular release phase. Please provide a justification for this proposed alternative from RG 1.183.

I&M Response to NRC Question 14

Section 3.0 of Attachment 6 to Reference 1 provides the proposed analysis of a small break LOCA. The gap release timing assumption used in the proposed analysis and the assumption options identified in RG 1.183 are graphed in the figure below. The assumption used in the proposed analysis, an essentially instantaneous release of 3% of the core activity combined with a subsequent linear release of 2% of the core activity, is enveloped by the two simpler alternatives identified in RG 1.183. Therefore, the assumption option chosen had no significant impact on the radiological consequences determined for the event.

Additionally, if it is assumed that either the instantaneous release model or the linear release model would provide more conservative results, the combined release model approach used in the proposed analysis ensured that at least a portion of the gap release was modeled using the more conservative modeling. If the two release models defined in RG 1.183 are considered as equivalent, a combination of the two models would also be equivalent, provided that the model includes release of the defined gap inventory to the containment in the half hour period specified.



NRC Question 15

§3.1.4 of Attachment 6, identifies that the sedimentation removal coefficient is conservatively assumed to be only 0.1 hr^{-1} and that sedimentation does not continue beyond a decontamination factor (DF) of 1000. Please justify the conservatism of these two assumptions against the DFs presented in Table 20 of NUREG/CR-6189, "A simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," and the effective decontamination coefficients presented in Table 24 of the same document.

I&M Response to NRC Question 15

Section 3.0 of Attachment 6 to Reference 1 provides the proposed analysis of a small break LOCA. As indicated in Section 3.1.4 of Attachment 6, the selection of 0.1 hr^{-1} for the sedimentation removal coefficient was based the Containment Systems Experiments. These were described in Industry Degraded Core Rulemaking (IDCOR) Program Technical Report 11.3, "Fission Product Transport in Degraded Core Accidents," Atomic Industrial Forum, December 1983. I&Ms basis for using the IDCOR report, rather than NUREG/CR-6189, is described below.

Use of the NUREG/CR-6189 model for pressurized water reactors (PWR) is inappropriate for CNP since the NUREG specifically models a large dry containment. The much smaller ice condenser containment at CNP would increase the concentration of particulates in containment, resulting in increased removal by sedimentation. Figures 18 and 19 on Pages 98 and 100 of NUREG/CR-6189 show the volume of existing PWR and boiling water reactor (BWR) containments plotted against the nominal thermal power of the reactor. The CNP Units 1 and 2 licensed thermal power is 3250 and 3411 MWt and, as indicated by item L5 of DIT-B-00069-06, the containment volume is approximately $1.26x10^6$ ft^3 or approximately $0.36x10^5$ m^3. As indicated by these figures, the CNP containment is much smaller than PWRs of comparable power and at the small end of the range of BWR containment volumes. Therefore, I&M considers that sedimentation rates for CNP would be higher than those reported for the average PWR.

Table 20 on Page 159 of NUREG/CR-6189 presents reasonable lower bound (10^{th} percentile) decontamination factors for the first 33.3 hours of the accident. I&M does not consider use of these values to be appropriate in evaluating doses over a 30 day period. I&M considers the assumption of a decontamination factor (DF) limit of 1000 for aerosols to be conservative for a 30 day period, since there is no inherent obstacle to complete removal of aerosols within this period. Additionally, use of a DF of 1000 is recognized by the NRC staff. Page 5 of the Safety Evaluation Report (SER) dated July 27, 2000, documenting NRC approval of Amendment 211 to the Indian Point Unit 2 facility operating license states that, "It [a limiting DF of 1000] is an arbitrary value because there is no actual limit for removal of particles by sedimentation. The only reason for having this limit is to establish a cut off point beyond which the rates of removal become so low that they would not have any practical significance."

Although elemental iodine has a removal limit based on the partitioning between the liquid and the gas, aerosols have no such constraint. NUREG 0800, the Standard Review Plan (SRP) Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," states that there is no need to limit the DF associated with removal of aerosols by sprays. Although containment spray is not being credited in this proposed analysis, the same concept would apply to sedimentation removal. Also, the empirical evidence from the testing documented in the IDCOR report shows a continuing reduction in airborne material even at very low air particulate concentrations. Additionally, on Page 200 of NUREG/CR-6189, a DF of 1000 and greater is implicitly supported by the statement that, in some cases, decontamination factors in excess of 10^6 were calculated but that values larger than that could not be justified. Although the statements were made in reference to boiling water reactors, the small volume of ice condenser containments renders the statements applicable to CNP.

Table 24 on Page 164 of NUREG/CR-6189 documents lower bound (10th percentile) decontamination coefficients for PWRs experiencing a severe accident core melt sequence based on NUREG-1465. I&M does not consider these values to be appropriate for application to CNP This set of coefficients reflects a core melt scenario that adds high levels of particulates, heat, and steam into the containment atmosphere that are not applicable to the small break LOCA. To remove any undue influence of core melt thermal hydraulics, the most representative time interval would be between 22.2 and 33.3 hours. In accordance with Table 24, this would yield a removal coefficient of approximately 0.03 hr^{-1}. Additionally, the selection of the 10th percentile coefficients is excessively conservative. According to Page 151 of NUREG/CR-6189, the use of mean values is considered appropriate for conservative analyses. From Table 25 on Page 165 of NUREG/CR-6189, the mean value for the removal coefficient is 0.074 hr^{-1}. This is well below the removal coefficient of approximately 0.35 hr^{-1} indicated by the IDCOR report and somewhat below the 0.1 hr^{-1} used in the proposed small break LOCA analysis. This brings the value to the same order of magnitude as the value used in the proposed analysis.

Finally, the NRC staff has previously accepted the use of a sedimentation removal coefficient of 0.1 hr^{-1} for a gap release in small-break LOCA thermal hydraulic conditions on the Indian Point 2 project. Page 6 of the SER for Indian Point Unit 2 Amendment 211 documents that the NRC staff found the sedimentation removal coefficient of 0.1 hr^{-1} to be reasonable.

NRC Question 16

For the analyses that have credited iodine partitioning in the steam generators, was the impact of steam generator tube uncovery during the transient considered? Was this considered in determining the flash fraction? If not, why not?

I&M Response to NRC Question 16

As described in the response to NRC Question 12, the NRC has accepted the WOG determination that tube uncovery has no significant impact on activity releases.

NRC Question 17

The 3rd and 4th paragraphs on page 27 of Attachment 6, appear to be addressing the same plant response but with different nomenclature. As we understand the system operation, the control room ventilation systems re-align on a safety injection signal, not a containment isolation signal as implied in the 3rd paragraph. Please confirm that the control room re-alignment occurs on an safety injection (SI) signal (e.g., low pressurizer pressure, low steamline pressure, high containment pressure, etc.).

I&M Response to NRC Question 17

Page 27 of Attachment 6 of Reference 1 provides a description of the proposed steam line break analysis. The CREVS would re-align on a safety injection signal within the first minute of a steam line break. The assumption that the CREVS realignment would be delayed for 5 minutes is conservative. The CREVS does not re-align on a containment isolation signal.

NRC Question 18

Items L43 and L44 in DIT-B-00069-06 identifies spray coverage for the three regions in the containment. This parameter was not addressed in the Attachment 6 discussion and was not tabulated in Table 11 of Attachment 6. Please describe how the spray coverage was incorporated into the analysis.

I&M Response to NRC Question 18

The proposed large break LOCA analysis credits removal of elemental iodine and particulates from the containment atmosphere by the containment spray system (operating in both injection and recirculation phases) and by the residual heat removal (RHR) spray in the upper containment region. DIT-B-00069-06 provided design input for the proposed analyses. The data provided in the DIT includes spray coverage, i.e., the percent of the containment region that is sprayed. The spray coverage data are not used in the analysis. The following paragraphs explain why not using the coverage data is appropriate for calculating the spray removal coefficients and how the coefficients were applied to the containment upper, lower, and annular regions.

Spray Removal Coefficients

Separate spray removal coefficients were calculated for the upper and active (non-dead ended) lower containment regions and the RHR spray using data provided in the DIT. The calculation of these spray removal coefficients was based on the assumption that the entire region would be sprayed. This is consistent with the modeling of the containment sprays in the proposed analysis. In addition, this provides conservative removal coefficients, since the equation used in the calculations includes the value for sprayed volume in the denominator. Therefore, using a value for the total volume in the equation results in a lower removal coefficient and higher calculated doses. Other conservative assumptions in the spray removal coefficient calculations include use of a larger value for volume than that considered in the proposed analysis, ignoring higher spray flow rates in the west portion of the upper containment and east portion of the lower containment, and rounding down the value for spray fall height.

Containment Upper And Lower Regions

The data in the DIT indicates that a significant fraction of the upper and lower containments would be unsprayed. The following provides justification for the assumption in the proposed analyses that the upper containment and active lower containment are fully sprayed.

The unsprayed portions of the upper and lower containment regions would not be physically separated from the sprayed portions. The action of the sprays would cause significant entrainment of air and create a high level of turbulence in the compartment air space such that there would be a rapid air exchange between the sprayed and unsprayed portions. The entire region (upper or lower containment) may therefore be considered as a completely sprayed volume. This position is supported by RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Dated June 1, 1984, which states that "Good mixing of the containment activity between the sprayed and unsprayed regions is ensured by natural convection currents and ESF fans."

Containment Annular Region

The data provided by the DIT indicates that most of the annular region would be unsprayed. In the proposed analyses, no credit was taken for the spray in the annular region due to the low spray fall height. The value for region volume was maximized and the value for airflow in and out of the annulus was selected to result in the greatest dose consequences. This conservatively accounts for unsprayed compartments.

Conclusion

I&M considers that the proposed LOCA analysis is justified in not modeling the unsprayed portions of the upper and active lower containment regions as separate volumes. The overall conservative modeling of the different containment regions and calculation of the spray removal coefficients in these regions, together with the expected mixing of the containment air support this conclusion. This conclusion is independent of the exact percentages identified as unsprayed in the DIT.

NRC Question 19

The staff has reviewed the information in Attachment 7 to your submittal. Item 6 on page 3 of this attachment addressed an issue related to design controls on changes made in the control room flow rates between 1982 and 1986, and whether or not the consequences of these changes were adequately evaluated. While your current re-analyses using the AST demonstrate compliance with GDC-19 (as revised in late 1999) this conclusion may not be applicable to the issue cited in 1986 since the source term and acceptance criterion were different. The staff expects to approve the current amendment request without accepting this item. Please indicate if you are requesting the NRC review and approval of the changes made to the control room flow rates between 1982 and 1986.

I&M Response to NRC Question 19

The response to NRC Question 19 was provided in Reference 3

NRC Question 20

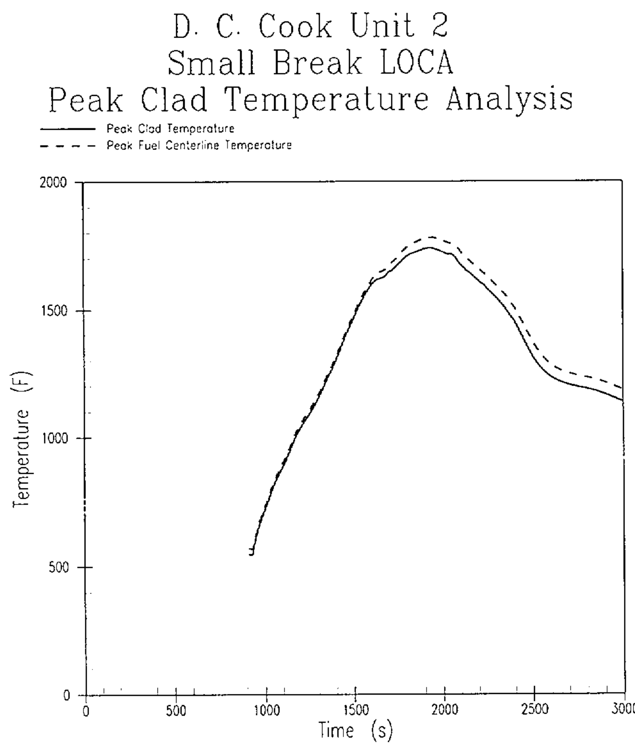
Please provide a description of the SBLOCA T/H analysis that was performed for determining the source term. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.

I&M Response to NRC Question 20

In a small break LOCA, a reactor trip signal would be generated and the control rods inserted terminating power generation. With the rods completely inserted, the potential for fuel failure or melting would come from a loss of decay heat removal, rather than an increase in power as would occur in a rod ejection event. The proposed SBLOCA dose analysis assumes cladding failure in 100% of the fuel rods, thereby releasing all of the gap activity. This is a conservative assumption that was made independent of any thermal-hydraulic (T/H) analyses.

However, the thermal hydraulic analyses conducted in accordance with 10 CFR 50.46 were used to justify the assumption that there would be no fuel pellet failure. As documented in Unit 1 and Unit 2 Sections 14.3.2 of the UFSAR, analyses conducted in accordance with 10 CFR 50.46 demonstrate that the peak cladding temperature (PCT) remains below the acceptance limit of

2200°F specified in the regulation. Since control rod insertion would terminate core power generation, the fuel pellet temperature would not be significantly higher than the cladding temperature. This is illustrated by the figure below.



This figure shows the base transient calculation of PCT for CNP Unit 2, together with the calculated peak fuel centerline temperature. Since the fuel pellet temperature would not be significantly higher than the cladding temperature, the cladding temperature would be maintained below 2200°F, and the fuel would not exceed its melting temperature of 4700°F

For CNP, like other ice condenser plants, it is likely that the containment sprays would actuate even for the small break LOCA due to the low actuation setpoint. However, containment spray was not credited in the small break LOCA analysis. If spray was credited, the small break LOCA dose consequences would be bounded by those calculated for the large break LOCA, since spray is credited in that analysis. However, if a significant amount of energy is not released to containment, as is the case in a the small break LOCA without significant fuel damage, there is a potential that the containment pressure would become sub-atmospheric if the sprays are not stopped. Therefore, the small break LOCA radiological consequences analysis was performed assuming no spray removal as a conservative assumption, although it is expected that spray flow would be initiated and continue until it is stopped by the operators based on the containment pressure response. If fuel melting were to occur, it is expected that the additional energy in containment would increase the pressure and result in extended spray operation.

NRC Question 21

The current licensing bases for D. C. Cook Units 1 and 2, use departure from nucleate boiling ratio (DNBR) as the criterion for determining the degree of fuel damage resulting from a locked rotor event. The licensee has not submitted either a request to modify its licensing basis or sufficient justification to demonstrate that the use of the 2700°F criterion is appropriate. We note that the staff has not accepted the use of the 2700°F criterion at other plants and further that the staff continues to believe that the DNBR criterion is the appropriate criterion for determining the amount of fuel failure. If you choose to use a criterion other than DNBR, please provide the technical justification for that criterion. Also, the description provided for the locked rotor event indicates that no pins exceed the DBNR limit. However, the description of the analysis does not include sufficient information for the staff to conduct its review. Therefore, please provide a description of the analysis for the locked rotor event. Please include a summary of and justification for the initial assumptions used, the sequence of events, the criteria used for determining fuel pin failures and/or fuel melting, the technical basis supporting the decision criteria, and the results of the analysis from the standpoint of justifying the analysis as limiting with respect to source term.

I&M Response to NRC Question 21

I&M agrees that the current licensing bases for CNP Units 1 and 2 uses DNBR rather than a PCT limit of 2700°F as the criterion for determining the degree of fuel damage resulting from a locked rotor event. I&M is not proposing to change this licensing basis.

Accordingly, the criterion used in the proposed locked rotor analysis to determine the amount of fuel failure is that rods with calculated DNBRs below the limit would fail, releasing gap activity. The analysis of the locked rotor event, including the assumptions utilized in the analysis and the results of the analysis, is discussed in Unit 1 Section 14.1.6.4 and Unit 2 Section 14.1.6.2 of the UFSAR. As described in the UFSAR, the analysis uses the LOFTRAN code to calculate the core coolant flow transient and the FACTRAN code to calculate the fuel heat flux transient. The coolant flow and heat flux values are then used to calculate the DNBR using the THINC code. The UFSAR currently states that 7% of the rods experience DNB for Unit 1 and 11% experience DNB for Unit 2. In support of the proposed locked rotor analysis, the DNB calculations using the THINC code were updated. No changes were made to the LOFTRAN flow calculations and FACTRAN heat flux calculations. The updated DNB calculations performed for the current fuel cycles, Unit 1 Cycle 17 and Unit 2 Cycle 12, show that no rods would have DNBRs below the limit.

The reanalysis for CNP Unit 1 Cycle 17 showing no rods would experience DNB used the Cycle 17 limiting normal operation power shape rather than the bounding design axial power shape which had been previously used. In addition, 0.5% unused DNB margin was allocated.

The previous analysis of the CNP Unit 2 Locked Rotor event for rods in DNB was done at a core power of 3588 MWt. However, since the plant is licensed to a power of 3411 MWt, then there was 5% power margin available. The applicable parameters (pressure, temperature, flow rate and heat flux) were reanalyzed for Cycle 12 using 3% of this power margin to show that no rods would experience DNB.

Appendix G of RG 1.183 states: "If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment." Therefore, no analyses of the locked rotor radiological consequences were performed. I&M committed in Reference 1 and in its letter to the NRC, C1100-010, dated November 7, 2000, to conduct cycle-specific reviews of Unit 1 and Unit 2 locked rotor events to demonstrate that the event would not result in control room doses that exceed the 5 rem TEDE limits of 10 CFR 50.67. I&M intends to fulfill this commitment by demonstrating that no rods would experience DNB, unless an alternative criterion is approved by the NRC or is authorized under the provisions of 10 CFR 50.59.

ATTACHMENT 2A TO C0801-02

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 1

3/4 7-19
3/4 7-24
3/4 9-14
3/4 9-15

B 3/4 7-5
B 3/4 7-5a

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

~~3/4.7.5~~CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

~~a. Two independent heating and cooling systems,~~

~~b.~~a. Two independent pressurization ~~fans~~trains, and

~~c.~~b. One charcoal adsorber ~~and~~ /HEPA filter ~~train~~ unit,

------NOTE------
The control room envelope/pressure boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

~~a. With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b.~~a. With one pressurization ~~fan~~train inoperable, restore the inoperable train~~fan~~ to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~c.~~b. With the filter ~~train~~unit inoperable, restore the filter ~~train~~unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two CREVS pressurization trains inoperable due to an inoperable control room envelope/pressure boundary, restore the control room envelope/pressure boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operating of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI N510 1980~~ ASTM D3803-1989 ~~(ASTM D 3803 1979~~, 30°C, 95% R.H.~~, and ≥ 45.5 fpm face velocity)~~. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510 1980 (ASTM D 3803 1979~~, 30°C, 95% R.H.~~, and ≥ 45.5 fpm face velocity)~~; or

2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples shows a penetration of less than or equal to 5%~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance ASTM D3803-1989~~with ANSI N510 1980 (ASTM D 3803 1979~~, 30°C, 95% R.H.~~, and ≥ 45.5 fpm face velocity)~~ and the samples are prepared by either:

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity~~)~~; or

SURVEILLANCE REQUIREMENTS (Continued)

2.. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5%~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity~~)~~ and the samples are prepared by either:

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

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

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~or equal to 6~~4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS operation is not credited during the rupture of a waste gas tank or toxic gas release. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE~~or less whole body, or its equivalent~~. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room. The Limiting Condition for Operation is modified by a Note allowing the control room envelope/pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors to the control room, 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 envelope/pressure boundary isolation is indicated.

If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room envelope/pressure boundary within 24 hours. During the period that the control room envelope/pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a design basis accident occurring during this time period, and the use of compensatory measures. The 24 hour completion time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room envelope/pressure boundary.

The Unit 1 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 2 is normally only available when the Unit 2 ESF actuation system is active in modes 1 through 4 in Unit 2.

The Limiting Condition for Operation requires two independent control room heating cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is $\leq$ 65°F.

The control room air conditioning~~ventilation~~ system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature $\leq$ 102°F during accident conditions with the control room isolated. At control room temperatures of $\leq$ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that adequate cooling is provided for ECCS equipment and that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the ESF ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse affect on any safety-related system.

ATTACHMENT 2B TO C0801-02

TECHNICAL SPECIFICATIONS PAGES
MARKED TO SHOW PROPOSED CHANGES

REVISED PAGES
UNIT 2

3/4 7-14
3/4 7-18
3/4 9-13
3/4 9-14

B 3/4 7-4a
B 3/4 7-4b

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

~~3/4.7.5~~ CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

~~a. Two independent heating and cooling systems,~~

~~b.~~a. Two independent pressurization trains~~fans~~, and

~~c.~~b. One charcoal adsorber ~~and~~/HEPA filter ~~train~~unit,

--------------------------------NOTE--------------------------------
The control room envelope/pressure boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, ~~and~~ 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

~~a. With one heating and cooling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b.~~a. With one pressurization ~~fan~~train inoperable, restore the inoperable ~~fan~~train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~c.~~b. With the filter ~~train~~unit inoperable, restore the filter ~~train~~unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two CREVS pressurization trains inoperable due to an inoperable control room envelope/pressure boundary, restore the control room envelope/pressure boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity.~~)~~ The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity~~)~~; or

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5%~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity~~)~~ and the samples are prepared by either:

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity~~)~~. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5%~~demonstrates a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95%,R.H., and ≥ 46.8 fpm face velocity.~~)~~

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples show a penetration of less than or equal to 5%~~demonstrate a removal efficiency of greater than or equal to 90%~~ for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989~~ANSI N510-1980 (ASTM D 3803-1979~~, 30°C, 95% R.H.~~, and ≥ 46.8 fpm face velocity)~~ and the samples are prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~or equal to 6~~4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

   2. Deleted.

   3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

   4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room ~~EMERGENCY~~emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE ~~5 rem or less whole body, or its equivalent~~. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room. The Limiting Condition for operation is modified by a Note allowing the control room envelope/pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors to the control room, 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 envelope/pressure boundary isolation is indicated.

If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room envelope/pressure boundary within 24 hours. During the period that the control room envelope/pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a design basis accident occurring during this time period, and the use of compensatory measures. The 24 hour completion time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room envelope/pressure boundary.

The Unit 2 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 1 is normally only available when the Unit 1 ESF actuation system is active in modes 1 through 4 in Unit 1.

The Limiting Condition for Operation requires two independent control room heating and cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is ≤ 65°F.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS (Continued)

The control room ~~ventilation~~air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature ≤ 102°F during accident conditions with the control room isolated. At control room temperatures of ≤ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

ATTACHMENT 3A TO C0801-02

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES
UNIT 1

3/4 7-19
3/4 7-24
3/4 9-14
3/4 9-15

B 3/4 7-5
B 3/4 7-5a

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

a. Two independent pressurization trains, and

b. One charcoal adsorber/HEPA filter unit,

---NOTE---
The control room envelope/pressure boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

a. With one pressurization train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the filter unit inoperable, restore the filter unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two CREVS pressurization trains inoperable due to an inoperable control room envelope/pressure boundary, restore the control room envelope/pressure boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operating of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity; or

   2. Verifying within 31 days after removal that laboratory analyses of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity and the samples are prepared by either:

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

   1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity; or

SURVEILLANCE REQUIREMENTS (Continued)

2.. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H. , and ≥ 46.8 fpm face velocity and the samples are prepared by either:

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

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

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

2. Deleted.

3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room emergency ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS operation is not credited during the rupture of a waste gas tank or toxic gas release. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room. The Limiting Condition for Operation is modified by a Note allowing the control room envelope/pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors to the control room, 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 envelope/pressure boundary isolation is indicated.

If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room envelope/pressure boundary within 24 hours. During the period that the control room envelope/pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a design basis accident occurring during this time period, and the use of compensatory measures. The 24 hour completion time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room envelope/pressure boundary.

The Unit 1 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 2 is normally only available when the Unit 2 ESF actuation system is active in modes 1 through 4 in Unit 2.

The Limiting Condition for Operation requires two independent control room heating cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is ≤ 65°F.

The control room air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature ≤ 102°F during accident conditions with the control room isolated. At control room temperatures of ≤ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

3/4.7.6 ESF VENTILATION SYSTEM

The OPERABILITY of the ESF ventilation system ensures that adequate cooling is provided for ECCS equipment and that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations were assumed in the accident analyses.

The 1980 version of ANSI N510 is used as a testing guide. This standard, however, is intended to be rigorously applied only to systems which, unlike the ESF ventilation system, are designed to ANSI N509 standards. For the specific case of the air-aerosol mixing uniformity test required by ANSI N510 as a prerequisite to in-place leak testing of charcoal and HEPA filters, the air-aerosol uniform mixing test acceptance criteria were not rigorously met. For this reason, a statistical correction factor will be applied to applicable surveillance test results where required.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3/4.7.8 HYDRAULIC SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse affect on any safety-related system.

ATTACHMENT 3B TO C0801-02

PROPOSED TECHNICAL SPECIFICATIONS PAGES

REVISED PAGES
UNIT 2

3/4 7-14
3/4 7-18
3/4 9-13
3/4 9-14

B 3/4 7-4a
B 3/4 7-4b

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5.1 The control room emergency ventilation system (CREVS) shall be OPERABLE with:

a. Two independent pressurization trains, and

b. One charcoal adsorber/HEPA filter unit,

---NOTE---
The control room envelope/pressure boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, and during the movement of irradiated fuel assemblies.

ACTION:

MODES 1, 2, 3, and 4:

a. With one pressurization train inoperable, restore the inoperable train to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the filter unit inoperable, restore the filter unit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two CREVS pressurization trains inoperable due to an inoperable control room envelope/pressure boundary, restore the control room envelope/pressure boundary to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

During the movement of irradiated fuel assemblies:

d. With one pressurization train inoperable, restore the inoperable pressurization train to OPERABLE status within 7 days, or initiate and maintain operation of the remaining OPERABLE train in the pressurization/cleanup alignment.

e. With any of the following (1) both pressurization trains inoperable; (2) the filter unit inoperable; or (3) the control room envelope/pressure boundary inoperable, immediately suspend all operations involving the movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 25,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 25,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity; or

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples shows a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 45.5 fpm face velocity and the samples are prepared by either:

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-1980 while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

4. Verifying within 31 days after removal that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity. The carbon samples not obtained from test canisters shall be prepared by either:

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

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

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

5. Verifying a system flow rate of 30,000 cfm plus or minus 10% during system operation when tested in accordance with ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by either:

1. Verifying within 31 days after removal that a laboratory analysis of a carbon sample obtained from a test canister shows a penetration of less than or equal to 5% for radioactive methyl iodide when the sample is tested in accordance with ASTM D3803-1989, 30°C, 95%,R.H., and ≥ 46.8 fpm face velocity.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that laboratory analysis of at least two carbon samples show a penetration of less than or equal to 5% for radioactive methyl iodide when the samples are tested in accordance with ASTM D3803-1989, 30°C, 95% R.H., and ≥ 46.8 fpm face velocity and the samples are prepared by either:

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

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

   Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also verifying that the charcoal adsorbers remove greater than or equal to 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1980 while operating the ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4 inches Water Gauge while operating the exhaust ventilation system at a flow rate of 30,000 cfm plus or minus 10%.

   2. Deleted.

   3. Verifying that on a high-radiation signal, the system automatically directs its exhaust flow through the charcoal adsorber banks and automatically shuts down the storage pool ventilation system supply fans.

   4. Verifying that the exhaust ventilation system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS

The OPERABILITY of the control room EMERGENCY ventilation system (CREVS) ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. In MODES 1-4, the CREVS provides radiological protection to allow operators to take the actions necessary to mitigate the consequences of a design basis accident. The CREVS is also required to be OPERABLE for operations involving the movement of irradiated fuel assemblies to provide protection from a fuel handling accident. The CREVS has two pressurization trains with each pressurization train consisting of a pressurization fan, normal intake air damper, and emergency intake air damper available to align and maintain flow to the control room. The charcoal adsorber/HEPA filter unit consists of the prefilter, charcoal adsorbers, HEPA filter, and filter housing. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to less than or equal to 5 rem Total Effective Dose Equivalent, TEDE. This limitation is consistent with the requirements of General Design Criteria (GDC) 19 of Appendix "A", 10 CFR 50.

The control room envelope/pressure boundary consists of the control room, the control room HVAC equipment room, and the plant process computer room. The Limiting Condition for operation is modified by a Note allowing the control room envelope/pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors to the control room, 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 envelope/pressure boundary isolation is indicated.

If the control room envelope/pressure boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room envelope/pressure boundary within 24 hours. During the period that the control room envelope/pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour completion time is reasonable based on the low probability of a design basis accident occurring during this time period, and the use of compensatory measures. The 24 hour completion time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room envelope/pressure boundary.

The Unit 2 control room emergency ventilation system aligns and operates automatically on a Safety Injection (SI) Signal from either Unit 1 or Unit 2. Both pressurization fans start on the SI signal. Procedures direct realignment of the CREVS to single fan operation within two hours after receiving the SI signal. The automatic start from Unit 1 is normally only available when the Unit 1 ESF actuation system is active in modes 1 through 4 in Unit 1.

The Limiting Condition for Operation requires two independent control room heating and cooling systems. Each cooling system requires a functional air handling unit and associated cooling water supply. Cooling water is provided from a chilled water unit. At the design maximum essential service water (ESW) supply temperature of 86°F, a chilled water unit will maintain the control room temperature below 95°F. Cooling water may also be supplied directly by ESW when ESW supply temperature is ≤ 65°F.

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION AND CONTROL ROOM AIR CONDITIONING SYSTEMS (Continued)

The control room air conditioning system (CRACS) normally maintains the control room at temperatures at which control room equipment is qualified for the life of the plant. Continued operation at the Technical Specification limit is permitted since the portion of time the temperature is likely to be elevated is small in comparison to the qualified life of the equipment at the limit.

Each control room cooling system can maintain control room temperature ≤ 102°F during accident conditions with the control room isolated. At control room temperatures of ≤ 102°F, vital control room equipment remains within its manufacturer's recommended operating temperature range.

ATTACHMENT 4 TO C0801-02

COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
I&M commits to have written procedures available describing compensatory measures to be taken in the event that the control room envelope/pressure boundary is inoperable in Modes 1, 2, 3, and 4.	Upon implementation of the proposed amendment following approval by the NRC
I&M commits to the applicable provisions of RG 1.183, dated July 2000, except for the proposed alternatives identified in the response to NRC Question 8.	Upon implementation of the proposed amendment following approval by the NRC
I&M will address the consequences of the error in processing the stability data provided by Reference 3.	September 14, 2001