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INDIANA
MICHIGAN
POWER

June 19, 2001

C0601-03
10 CFR 50.90

Docket Nos.: 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
PARTIAL 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 (I&M) to 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 M. W. Rencheck (I&M) to NRC Document Control Desk, "License Amendment Request, Technical Specification Change for Inoperable A.C. or D.C. Distribution Systems in Modes 5 & 6 and for Containment Penetrations During Refueling," C0501-02, dated May 15, 2001.

3) Letter from M. W. Rencheck (I&M) to NRC Document Control Desk, "Technical Specification Change Request Refueling Operations Decay Time," C0501-03, dated May 17, 2001.

4) 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).

A001
Rec'd 7/9/01
DCD

As described in detail below under "Background," this letter provides a partial response to Nuclear Regulatory Commission (NRC) questions pertaining to a previously submitted amendment request that addressed control room habitability issues at Donald C. Cook Nuclear Plant (CNP) Units 1 and 2. This letter addresses those NRC questions or portions of questions involving the proposed fuel handling accident (FHA) analysis submitted in that amendment request. I&M is addressing the questions involving the proposed FHA analysis separately to facilitate timely NRC review of two other amendment requests that are needed to support the upcoming Unit 2 refueling outage. These two other amendment requests both credit the proposed FHA analysis. Responses to the remainder of the NRC questions will be submitted later, by August 17, 2001, since their scope is much more extensive.

Background

In Reference 1, Indiana Michigan Power Company (I&M), the Licensee for CNP Unit 1 and Unit 2, proposed to amend Facility Operating Licenses DPR-58 and DPR-74 to 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." Reference 1 included proposed control room dose analyses using the AST for various accidents and events. I&M also proposed to amend Appendix A, Technical Specifications (T/S), of the Facility Operating Licenses. Reference 1 also included proposed changes to the T/S for ventilation systems pursuant to NRC Generic Letter 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal," dated June 3, 1999.

In References 2 and 3, I&M submitted proposed changes to T/S requirements for containment penetrations during refueling and for shutdown time prior to fuel movement. These proposed changes were based, in part, on the proposed FHA analysis that was included in Reference 1. NRC approval of the T/S changes proposed in References 2 and 3 was requested by October 1, 2001, and October 15, 2001, respectively, to support an upcoming Unit 2 refueling outage.

In Reference 4, the NRC requested additional information regarding the changes proposed in Reference 1. This letter provides that portion of the information requested in Reference 4 needed to support NRC review of the proposed FHA analysis included in Reference 1. Upon issuance of the amendments requested

by Reference 2 and 3, the inputs, assumptions, and methodologies used in the proposed FHA analysis would become part of the CNP licensing basis, and would be controlled as such in accordance with established regulations and procedures. As stated above, I&M will provide the remainder of the information requested by Reference 4 in a supplemental response by August 17, 2001.

Responses to Questions

Attachment 1 to this letter addresses the specific questions transmitted in Reference 4 by either providing the requested information or committing to provide the information in the supplemental response. Attachment 2 provides an assessment demonstrating that the No Significant Hazards Consideration Evaluation provided in Attachment 4 to Reference 1 remains valid. Attachment 3 provides a listing of commitments made in this letter. Enclosed with this letter is a computer disk containing the meteorological data input to the ARCON 96 computer code.

I&M has determined that the environmental assessment provided in Attachment 5 to Reference 1 is not affected by the information transmitted in this letter.

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/Enclosure

c: J. E. Dyer, w/o enclosure
MDEQ - DW & RPD, w/o attachments/enclosure
NRC Resident Inspector, w/o enclosure
R. Whale, w/o attachments/enclosure

bc: R. J. Grumbir, w/o enclosure
S. B. Haggerty, w/o enclosure
D. W. Jenkins, w/o attachments/enclosure
L. A. Lahti, w/o attachments/enclosure
M. W. Rencheck/S. A. Greenlee, w/o attachments/enclosure
E. M. Ridgell, w/o attachments/enclosure
J. F. Stang, Jr., - NRC Washington, DC

ATTACHMENT 1 TO C0601-03

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

Reference 1) Letter from J. F. Stang, Nuclear Regulatory Commission (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).

2) Letter from R. P. Powers, I&M, to NRC Document Control Desk, "License Amendment Request for Control Room Habitability and Generic Letter 99-02 Requirements," C0600-13, dated June 12, 2000.

This attachment addresses the specific questions transmitted by Reference 1 by providing the portion of the information needed to support NRC review of the proposed fuel handling accident (FHA) analysis included in Reference 2, and committing to provide the requested information for the remaining analyses in a supplemental response. Full or partial responses are provided to NRC Questions 1 through 5, 7 through 9, and 19, as they impact the FHA analysis. The supplemental response will be submitted by August 17, 2001.

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)*

Indiana Michigan Power Company (I&M) Response to Question 1

The only charcoal adsorber credited in the proposed FHA analysis is that in the CREVS. However, the following response also addresses ESFVS and the SPVS charcoal adsorbers, since the same methodology was used to determine face velocities and residence times for all three systems.

I&M has calculated the actual face velocities and residence times for the charcoal adsorbers in the CREVS, ESFVS, and SPVS. The face velocities and residence times resulting from the maximum flow allowed by the T/S were determined to be as follows:

| Parameter | CREVS | ESFVS | SPVS |
|---|---|---|---|
| Face Velocity (ft/min) | 43.7 | 45.5 | 46.8 |
| Residence Time (sec) | 0.229 | 0.220 | 0.214 |

This information supercedes that provided on page 18 of Attachment 1 to Reference 2, which stated that the face velocities for all three systems were less or equal to than 40 ft/min and residence times were greater than or equal to 0.25 seconds. The discrepancy between the information provided above and that provided in Reference 2 has been entered in the CNP Corrective Action Program.

The flow rates assumed in calculating these values were the maximum flow rates as specified in the T/S for the CREVS (T/S 3/4.7.5.1), EFSVS (T/S 3/4.7.6.1), and SPVS (T/S 3/4.9.12). These flow rates were 6,600 cubic feet per minute (cfm), 27,500 cfm, and 33,000 cfm, respectively. The total exposed surface area of the charcoal adsorber media was derived from information provided in vendor documents. The exposed surface areas were determined to be 151.2 $ft^2$, 604.8 $ft^2$, and 705.6 $ft^2$ for the CREVS, ESFVS, and SPVS, respectively. The face velocities tabulated above were determined by dividing the flow rates by the respective areas. The charcoal adsorber bed depth for all three systems was determined to be 2 inches from vendor documents. The residence times tabulated above were calculated by dividing the bed depths by the face velocities. The above described methodologies are consistent with those described in the NRC question.

As shown in the above table, the face velocity for the CREVS charcoal adsorbers is less than the value specified in GL 99-02 (110% of 40 ft/min = 44 ft/min). However, the face velocities of the charcoal adsorbers in the ESFVS and the SPVS exceed the specified value. Therefore, I&M will include revised T/S pages specifying the appropriate face velocity in the T/S Surveillance Requirements for these systems in the supplemental response.

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 Question 2

The following response addresses only the CREVS charcoal adsorber since it is the only charcoal adsorber credited in the proposed FHA analysis. Additionally, the response addresses only single-fan operation of the CREVS, since two-fan operation is not assumed in the proposed FHA analysis. The safety factors for the CREVS, ESFVS, and SPVS charcoal adsorbers with respect to other accidents and events will be addressed in the supplemental response.

The current T/S Surveillance Requirements for laboratory testing of the CREVS charcoal adsorber with radioactive methyl iodide, T/S 4.7.5.1.c.3, T/S 4.7.5.1.d.1, and T/S 4.7.5.1.d.2, require that samples of the charcoal demonstrate an efficiency of greater than or equal to 90%. Currently, the Donald C. Cook Nuclear Plant (CNP) licensing basis does not include an analysis of the control room dose resulting from an FHA. Therefore, a safety factor pertaining to the currently required charcoal adsorber efficiency has not been calculated for the control room dose resulting from a FHA.

The T/S Surveillance Requirements proposed in Reference 2 for laboratory testing of the CREVS charcoal adsorber with radioactive methyl iodide require a penetration of less than or equal to 1.0%. As documented in Table 3 of Attachment 6 to Reference 2, the proposed accident analyses assume an efficiency of 95%, i.e., a penetration of 5%, for single-fan operation. The 5% value assumed in the proposed analyses divided by the 1.0% testing criteria specified in the proposed amendment results in a safety factor equal to 5.

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 Question 3

Questions 3(a), 3(b), and 3(c) concern only two-fan operation of the CREVS. Two-fan operation is not assumed in the proposed FHA analysis. Therefore, the response to Questions 3(a), 3(b), and 3(c) will be provided in the supplemental response. The response to Question 3(d) regarding two-fan operation will also be provided in the supplemental response. The response to Question 3(d) regarding single-fan operation is provided below.

The proposed T/S Surveillance Requirements specify 1% penetration at the normal (single fan) CREVS flowrate. As described in the response to Question 1, the single-fan flowrate has been determined to provide a face velocity that is less than 110% of 40 cfm, which is the criterion specified in GL 99-02. Since an assumption of 5% penetration is used in the proposed FHA analysis, the 1% penetration criterion specified in the proposed Surveillance Requirement bounds the analysis assumption.

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 Question 4

The unfiltered makeup flow for CREVS in the normal mode was assumed to be 1000 cfm in the proposed analysis of accidents other than the FHA. A value of 3000 cfm was assumed in the proposed FHA analysis. The following response provides the bases for the value assumed in the proposed FHA analysis. The basis for the unfiltered makeup assumed for other events in which the CREVS is in the normal mode will be addressed in the supplemental response.

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 3000 cfm assumed in the proposed FHA analysis, i.e., 2040 cfm, provides a margin that can be used to account for unfiltered inleakage.

During tracer gas testing, unfiltered inleakage was determined to be 49 ± 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 pressure boundary where a large differential pressure exists to force outside air into the control room. Tracer gas testing demonstrated that at least 17 cfm of the measured 49 ± 49 cfm 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 no more than 49 + 49 cfm minus 17 cfm, i.e., no more than 79 cfm in the normal mode. This is much less than the 2040 cfm available margin.

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 Question 5

The following response is not limited to the proposed FHA analysis.

The proposed accident analyses use 98 cfm of unfiltered inleakage when the CREVS is in the emergency mode. The 98 cfm value is based on the unfiltered inleakage into the Unit 2 control room measured using the constant injection method of tracer gas testing. The results of this testing show a nominal inleakage of 49 cfm with 95% confidence that the actual inleakage is between 0 and 98 cfm. The 98 cfm value is, therefore, the limiting value for the Unit 2 control room.

The equivalent tracer gas test results for the Unit 1 control room showed a nominal inleakage of 144 cfm with 95% confidence that the actual inleakage is between 120 and 168 cfm. However, the majority of the inleakage that occurred during the test was through a normal intake damper that was leaking excessively. Measurements using tracer gas determined the nominal inleakage through the intake damper to be 107 cfm with 95% confidence that the actual flow rate is between 79 and 135 cfm. Therefore, the leakage through the intake damper led to test results that grossly overestimated the actual unfiltered inleakage that could be expected for the Unit 1 pressure boundary after the damper was repaired.

The proposed accident analyses used a reasonable correction for the gross overestimation of unfiltered inleakage for Unit 1. Assuming the minimum measured flow rate through the failed damper of 79 cfm and the maximum measured unfiltered inleakage for Unit 1 of 168 cfm, then the maximum unfiltered inleakage from sources other than the failed intake damper would be 89 cfm. Subsequent to the tracer gas testing, the single normal intake dampers on both units

were each replaced with two dampers in series. Each of the two series dampers is tested periodically. If measured leakage exceeds 5 cfm through either damper, then corrective measures are initiated. Adding the procedurally-allowed limit of 5 cfm to the conservative estimate of unfiltered inleakage for Unit 1 brings the total to 94 cfm (89 + 5), which is still below the 98 cfm value used in the proposed analysis.

I&M, therefore, considers that using the Unit 2 measured leak rate of 98 cfm at 95% confidence for both units is appropriate. The value has been determined to be limiting for Unit 2 by direct measurement. The preceding evaluation demonstrates that the value is conservative for Unit 1 following damper replacement.

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 Question 6

The proposed FHA analysis is based on the assumption that the control room envelope/pressure boundary is operable. NRC Question 6 involves the criteria for defining an operable control room envelope/pressure boundary. I&M's resolution of NRC Question 6 will not alter any assumptions in the proposed FHA analysis. I&M is re-evaluating the proposed changes to the T/S for the CREVS with respect to the concerns identified in NRC Question 6 and is reviewing the provisions of TSTF-287 for potential incorporation into the proposed T/S. I&M's resolution of this question will be provided in the supplemental response.

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 Question 7

The response to Question 8 describes the significant differences between the assumptions used in the proposed FHA analysis and those identified in RG 1.183. These differences constitute proposed alternatives to the RG 1.183 provisions pertaining to FHA analyses. As stated in the response to Question 8 below, I&M does not intend to revise the proposed FHA analysis at this time, but may do so in the future.

The proposed alternatives to the provisions of RG 1.183 for the other analyzed accidents and events will be provided in the supplemental response.

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 Question 8

I&M has identified three differences between the FHA analysis assumptions contained in the proposed FHA analysis and those identified in Appendix B to RG 1.183. These differences constitute proposed alternatives to the assumptions identified in RG 1.183 for a FHA. The proposed alternatives to the provisions of RG 1.183 for the other analyzed accidents and events will be provided in the supplemental response.

Gap Fractions

The fraction of core activity assumed to exist in the gap between the fuel and the cladding differs between the proposed FHA analysis and RG 1.183. A comparison of the values assumed in the two documents is provided in the table below. As shown in the table, higher gap fractions were assumed in the proposed FHA analysis than are identified in RG 1.183 for all nuclides. Therefore, the dose determined by the proposed FHA analysis is at least 1.5 times higher than would have been calculated using the gap fractions identified in RG 1.183.

| | Proposed FHA Analysis | RG 1.183 | Factor of Conservatism in Proposed Analysis |
|---|---|---|---|
| I-131 | 12% | 8% | 1.5 |
| Kr-85 | 15% | 10% | 1.5 |
| Other Halogens and Noble Gasses | 10% | 5% | 2.0 |

Percent of Organic Iodine

The assumed percent of iodine released from the fuel into the spent fuel pool that is in organic form differs between the proposed FHA analysis and that identified in RG 1.183. A value of 0.25% is assumed in the proposed FHA analysis while a value of 0.15% is identified in RG 1.183. Since organic iodine is not retained in the spent fuel pool, the value assumed in the proposed FHA analysis value is conservative.

Elemental Iodine Decontamination Factor

The elemental iodine pool decontamination factor (DF) of 400 used in the proposed FHA analysis differs from the value of 500 identified in RG 1.183. The more conservative value of 400 was used to compensate for rods with elevated internal pressure. The overall pool DF is determined by the percent of iodine that is in elemental form and the elemental iodine DF. With the values identified in RG 1.183, 99.85% of the iodine in elemental form and an elemental iodine DF of 500, the overall DF is 286. With the proposed FHA analysis assumptions, 99.75% of the iodine in elemental form and an elemental iodine DF of 400, the overall DF is 200. Thus, the value for overall DF used in the proposed FHA analysis is lower than that identified in

RG 1.183. Although the chemical form of the iodine released from the spent fuel pool differs due to the differing elemental release fractions and elemental DFs, there is no impact on the calculated dose since the control room filter efficiencies for elemental and organic iodine are identical and no other filtration is credited.

As described above, the proposed FHA analysis contains gap fraction assumptions that are at least a factor of 1.5 more limiting than those identified in RG 1.183. However, I&M does not intend to revise the analysis. I&M may elect to use this margin in future evaluations. For example, if a re-analysis is required for a change in licensed reactor power, peaking factor, or decay time prior to fuel movement, the assumed gap fractions may be reduced to those identified in RG 1.183, thereby allowing a change in other analysis assumptions without an increase in consequences.

NRC Question 9

*Your analyses incorporated revised atmospheric dispersion ($\chi/Q$) 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 $\chi/Q$ 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 Question 9

Since the ARCON96 computer code was used in the proposed analysis of many of the accidents and events, the following response is not limited to the proposed FHA analysis.

Meteorological data input to ARCON96 was collected by CNP meteorological instrumentation. The equipment used to collect the data is described in Section 2.2.1 of the UFSAR and is governed by T/S 3/4.3.3.4, "Meteorological Instrumentation." The tape contains wind direction and wind speed at 10 and 60 meters from ground level collected during 1996, 1997, and 1998. The vendor, EQE/PLG, used the tape to create input for ARCON96 under a contract with I&M.

The information was transferred to a VAX system using the SAVRES program, and to the EQE/PLG network server using the KERMIT protocol. The data, in MIDAS workspace format, was validated by a meteorologist on the EQE/PLG staff to ensure that the wind speed and direction were within normal operating ranges. Invalid data were not used. The MIDAS workspace file was then converted to the format required for input to ARCON96 using the WK2ARCON program. The contract with EQE/PLG required that all work be performed under and subject to 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Both DG-1081 and RG 1.183 state that ARCON96 is generally acceptable for determining control room $\chi/Q$ values, but state that the applicability of ARCON96 processing options and associated input values should be evaluated on a case by case basis. To facilitate NRC evaluation, the Unit 1 and Unit 2 release mode and configuration information for the release point and receptor are given in the following table. The meteorological sensor configuration input to the ARCON96 code is that the sensors are located at 10 and 60 meters from ground level.

| Release Mode and Configuration Information for Release Point and Receptor | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Source-Receptor Pair Description | Rel. Type * | Rel. Height (m) | Building Area (m2) | Vertical Velocity (m/s) | Stack Flow (m3/s) | Stack Radius (m) | Direction to Source (deg) | Distance to Intake (m) | Intake Height (m) |
| U1 Vent to U1 Intake | 1 | 48.77 | 1124.74 | 0.00 | 65.88 | 0.69 | 79.71 | 45.46 | 15.62 |
| U1 E. PORV to U1 Intake | 3 | 25.73 | 1124.74 | 33.40 | 87.73 | 0.91 | 87.91 | 65.07 | 15.62 |
| U1 W. PORV to U1 Intake | 3 | 25.81 | 0.10 | 33.40 | 87.73 | 0.91 | 59.29 | 28.15 | 15.62 |
| U1 Cont. to U1 Intake | 1 | 45.85 | 1124.74 | 0.00 | 0.00 | 0.00 | 79.71 | 26.86 | 15.62 |
| U1 RWST to U1 Intake | 1 | 12.89 | 0.10 | 0.00 | 0.00 | 0.10 | 52.19 | 69.44 | 15.62 |
| U1 Vent to U2 Intake | 1 | 48.77 | 1161.60 | 0.00 | 65.88 | 0.69 | 55.65 | 64.19 | 14.71 |
| U1 E. PORV to U2 Intake | 3 | 25.73 | 1158.76 | 33.40 | 87.73 | 0.91 | 67.42 | 79.37 | 14.71 |
| U1 W. PORV to U2 Intake | 3 | 25.81 | 1161.60 | 33.40 | 87.73 | 0.91 | 37.39 | 53.46 | 14.71 |
| U1 Cont. to U2 Intake | 1 | 45.85 | 1161.60 | 0.00 | 0.00 | 0.00 | 55.65 | 45.58 | 14.71 |
| U1 RWST to U2 Intake | 1 | 12.89 | 1161.60 | 0.00 | 0.00 | 0.10 | 41.77 | 94.75 | 14.71 |
| U2 Vent to U1 Intake | 1 | 48.77 | 1092.48 | 0.00 | 48.85 | 0.69 | 157.19 | 64.28 | 15.62 |
| U2 E. PORV to U1 Intake | 3 | 25.73 | 1124.74 | 33.40 | 87.73 | 0.91 | 145.14 | 79.14 | 15.62 |
| U2 W. PORV to U1 Intake | 3 | 25.81 | 1092.48 | 33.40 | 87.73 | 0.91 | 175.24 | 53.12 | 15.62 |
| U2 Cont. to U1 Intake | 1 | 45.85 | 1092.48 | 0.00 | 0.00 | 0.00 | 157.19 | 45.68 | 15.62 |
| U2 RWST to U1 Intake | 1 | 12.89 | 1092.48 | 0.00 | 0.00 | 0.10 | 171.03 | 94.86 | 15.62 |
| U2 Vent to U2 Intake | 1 | 48.77 | 1158.76 | 0.00 | 48.85 | 0.69 | 133.20 | 45.52 | 14.71 |
| U2 E. PORV to U2 Intake | 3 | 25.73 | 1158.76 | 33.40 | 87.73 | 0.91 | 124.56 | 64.96 | 14.71 |
| U2 W. PORV to U2 Intake | 3 | 25.81 | 0.10 | 33.40 | 87.73 | 0.91 | 152.97 | 27.89 | 14.71 |
| U2 Cont. to U2 Intake | 1 | 45.85 | 1158.76 | 0.00 | 0.00 | 0.00 | 133.20 | 26.91 | 14.71 |
| U2 RWST to U2 Intake | 1 | 12.89 | 0.10 | 0.00 | 0.00 | 0.10 | 160.64 | 69.54 | 14.71 |

*Release Type 1 is a ground release. Release Type 3 is a stack release.

A disk containing the meteorological data input to ARCON96 in the ARCON96 input data format is enclosed with this letter.

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 Question 10

The proposed FHA analysis does not involve the iodine flashing fraction. I&M's resolution of this question will be provided in the supplemental response.

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 Question 11

This question involves SBLOCA analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 12

This question involves SGTR analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 13

This question involves accidents affecting an operating core and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 14

This question involves SBLOCA analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 15

This question involves SBLOCA analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 16

This question involves iodine partitioning in the steam generators and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 17

This question involves only automatic realignment of the CREVS to the emergency mode. Automatic realignment is not credited in the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 18

This question involves containment spray coverage and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 19

The following response is not limited to the proposed FHA analysis.

I&M is not requesting that the NRC review and approve the changes made to the control room ventilation flow rates between 1982 and 1986. The purpose of Attachment 7 to Reference 2 is to address various historical concerns related to control room habitability for CNP. One of the concerns involved a past design control issue. Specifically, the concern noted that filtered makeup, filtered recirculation and unfiltered inleakage flow rates were changed from those accepted by NRC in a 1982 Safety evaluation report.

As indicated in the "Method for Addressing Expectations/Suggestions/Comments" for Item 6, new values for the flow rates in question were used in the analyses proposed as part of this amendment request. As documented in Attachment 6 to Reference 2, the proposed accident analyses demonstrate that the new flow rates provide dose consequences that are within the limits of 10 CFR 50.67.

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 Question 20

This question involves SBLOCA analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

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 Question 21

This question involves locked rotor event analyses and does not affect the proposed FHA analysis. I&M's resolution of this question will be provided in the supplemental response.

ATTACHMENT 2 TO C0601-03

# COMPARISON TO PREVIOUS NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Indiana Michigan Power Company (I&M) has determined that the responses transmitted by this letter do not affect the validity of the original No Significant Hazards Consideration (NSHC) evaluation which was provided as Attachment 4 to its letter to the Nuclear Regulatory Commission (NRC), C0600-13, dated June 12, 2000. The basis for this determination is provided below for the responses to the NRC questions, which have been grouped as follows: Responses to NRC Question 1, Responses to NRC Questions 2 - 5, 8, and 9, Responses to NRC Questions 7 and 19, Commitments made in the responses to NRC Questions 1 - 3, 6, 8, 10 - 18, 20, and 21.

## Response to NRC Question 1

The response to NRC Question 1 provides values for charcoal adsorber face velocities and residence times, and describes the inputs and methodology used to calculate them. Provided below is a description of how the criteria of 10 CFR 50.92(c) were addressed in the original NSHC evaluation for the changes involving charcoal adsorbers, and the reasons that the original evaluation remains valid.

Probability of a Previously Evaluated Accident

The original NSHC evaluation states:

"The proposed changes for the charcoal testing method affect activities in the laboratory only and have no impact on plant operation. Sampling and testing charcoal will not initiate an accident. The charcoal adsorbers are used to mitigate the consequences of an accident and are not operated until after an accident has occurred. Therefore, the probability of an accident previously evaluated is not affected. The [high-efficiency particulate air] HEPA filter/charcoal adsorber units in the [control room emergency ventilation system] CREVS, [engineered safety features ventilation system] ESFVS, and [storage pool ventilation system] SPVS are designed to mitigate the consequences of an accident. They are not assumed to operate until after an accident has occurred. The adsorber units have no impact on the initiation of any evaluated accidents."

The validity of these conclusions is not affected by the response to NRC Question 1 since the function of the charcoal adsorbers is unchanged. Sampling and testing charcoal do not initiate an accident. The charcoal adsorbers are used only to mitigate the consequences of an accident.

Consequences of a Previously Evaluated Accident

The original NSHC evaluation states:

"Charcoal testing verifies the ability of the charcoal adsorbers to function as assumed following an accident. The new method for testing the CREVS samples provides more accurate and reproducible laboratory results. These results provide assurance that the charcoal adsorbers will meet the assumed radioiodine removal efficiency following an accident. Therefore, the consequences of accidents previously evaluated are not increased. The proposed change to incorporate the new laboratory testing standard for charcoal adsorbers in the ESFVS and SPVS is administrative because the test conditions are consistent with the standard referenced in the [Technical Specifications] T/S."

The validity of this conclusion is not affected by the response to NRC Question 1. The efficiency of the charcoal adsorbers assumed in the proposed accident analyses and verified by surveillance testing is not changed by the response to NRC Question 1. Although the face velocity for the ESFVS and SPVS has been determined to exceed 110% of 40 cubic feet per minute (cfm), I&M has committed to revise the proposed T/S to assure the required efficiency is verified through surveillance testing in accordance with Generic Letter (GL) 99-02.

Possibility of a New or Different Kind of Accident

The original NSHC evaluation states:

"The CREVS is designed to mitigate the consequences of an accident. It is not assumed to operate until after an accident has occurred. The proposed changes to the CREVS requirements do not introduce any new plant equipment or new methods of operating the equipment. No new failure mechanisms are introduced. The proposed change to incorporate the new testing requirements of ASTM D3803-1989 is administrative in nature. It affects activities in the laboratory only and has no impact on plant operation. The change does not affect the method for obtaining the charcoal sample. It does not cause any of the ventilation equipment to be operated in a new or different manner. Therefore, it is concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated."

The validity of this conclusion is not affected by the response to NRC Question 1, since the response does not affect the determination that the proposed change involves no new equipment or operation of existing equipment in a new or different manner.

Margin of Safety

The original NSHC evaluation states:

"The margin of safety associated with the CREVS T/S is to maintain control room dose within the limits of GDC-19. The proposed changes to the CREVS requirements ensure that accident analysis assumptions are preserved so that the dose limit is met. The proposed change to incorporate the testing standards recommended for the charcoal adsorbers in GL 99-02 provides assurance that the charcoal adsorbers will remove radioiodine as assumed in the accident analysis. Additional margin is gained by applying a safety factor to the iodine removal efficiency assumed in the accident analysis. This safety factor applies to CREVS, ESFVS, and SPVS. The T/S have also been revised to reflect the iodine removal efficiency assumed in the accident analysis. The acceptance criterion reflects the analysis assumption and the safety factor. In summary, based upon the above evaluation, I&M has concluded that these changes involve no significant hazards consideration."

The validity of this conclusion is not affected by the response to NRC Question 1. Although the face velocity for the ESFVS and SPVS has been determined to exceed 110% of 40 cfm, I&M has committed to revise the proposed T/S to assure the required efficiency is verified through surveillance testing in accordance with GL 99-02. This will assure that the necessary safety factors are maintained. Therefore, the response to NRC Question 1 provides assurance that the required safety factors will be maintained.

Response to NRC Questions 2 - 5, 8, and 9

As described below, the responses to NRC Questions 2 through 5, 8, and 9 provide information that supports the proposed accident analyses using the Alternate Source Term (AST).

The response to NRC Question 2 describes why there is no current iodine removal safety factor for a fuel handling accident (FHA) analysis, and describes how the proposed FHA analysis provides a safety factor of 5. The response to NRC Question 3 describes how single-fan operation of the CREVS provides 1% charcoal adsorber penetration, which is consistent with the value used in the proposed accident analyses. The response to NRC Question 4 describes the basis for the unfiltered makeup value used in the proposed FHA analyses when the CREVS is not operated in the emergency mode. The response to NRC Question 5 describes the basis for the assumption used in the proposed accident analyses that there is 98 cfm of unfiltered inleakage to the control room when the CREVS is operated in the emergency mode. The response to NRC Question 8 provides a comparison of the assumptions used in the proposed FHA analysis with those identified in Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The response to NRC Question 9 provides a description of quality assurance provisions and inputs to the ARCON96 computer code that was used to calculate atmospheric dispersion values in the proposed accident

analyses, and provides a computer disk containing meteorological data used in the calculations. I&M's use of the ARCON96 code was previously identified in the original amendment request.

Provided below is a description of how the criteria of 10 CFR 50.92(c) were addressed in the original NSHC for the proposed accident analyses using the AST, and the reasons that the original NSHC evaluation remains valid.

Probability of a Previously Evaluated Accident

The original NSHC evaluation states:

"The proposed change to implement the AST involves changes to the methodologies and acceptance criterion associated with the control room dose analysis. The actual sequence and progression of accidents are not changed. However, the regulatory assumptions regarding the analytical treatment of the accidents are affected by the change. The use of an AST alone cannot increase the probability of an accident or the core damage frequency. The proposed change to use the AST does not make any changes to equipment, procedures, or processes that increase the likelihood of an accident. It does not affect any accident initiators or precursors. The methodology is used to determine consequences of an accident and has no impact on their likelihood of occurrence. Therefore, this proposed change does not involve a significant increase in the probability of an accident previously evaluated."

The validity of this conclusion is not affected by the response to NRC Questions 2 through 5, 8, and 9. The responses to NRC Questions 2 through 5, 8, and 9 provide information that supports the proposed accident analyses. These responses do not alter the methodology or acceptance criterion for the proposed analyses.

Consequences of a Previously Evaluated Accident

The original NSHC evaluation states:

"The current acceptance criterion specify the dose to personnel in terms of "rem whole body" or equivalent for the duration of the accident, where the dose derived using the AST is given in rem TEDE, as described in 10 CFR 50.67. TEDE includes internal and external exposure; whole body includes external exposure only. The current acceptance criterion focuses on doses to the thyroid and the whole body. It is based on the assumption that the major contributor to dose will be radioiodine. Although this may be appropriate with the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites", source term implemented by RGs 1.4, it may not be true for a source term based on a more complete understanding of accident sequences and phenomenology. The AST includes a larger number of radionuclides than did the TID-14844 source term as implemented in regulatory guidance. The whole body and thyroid dose criteria considered the noble gases and iodine contributors as the

limiting factors. The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent, so they cannot be compared directly. I&M has reanalyzed the loss of coolant accident (LOCA) and non-LOCA events to determine the limiting condition for control room dose using the AST. The calculated dose for all the analyzed events meets the acceptance criterion for GDC-19 as described in 10 CFR 50.67. Therefore, the consequences are not significantly increased."

The validity of this conclusion is not affected by the response to NRC Questions 2 through 5, 8, and 9. The responses to NRC Questions 2 through 5, 8, and 9 provide information that supports the proposed accident analyses. These responses do not alter the number of nuclides or acceptance criterion for the proposed analyses.

Possibility of a New or Different Kind of Accident

The original NSHC evaluation states:

"The use of an AST alone cannot create the possibility of a new or different kind of accident. The proposed change to use the AST does not make any changes to equipment, procedures, or processes. The AST does not create any new accident initiators or precursors. It is merely a method used to predict radionuclides released following an accident. Therefore, this proposed change does not increase the possibility of a new or different kind of accident than previously evaluated."

The validity of this conclusion is not affected by the response to NRC Questions 2 through 5, 8, and 9. These responses only provide information that supports the proposed accident analyses using the AST. These responses do not make any changes to equipment, procedures, or processes, nor do they create any new accident initiators or precursors.

Margin of Safety

The original NSHC evaluation states:

"The proposed change to implement the AST for the revised analysis incorporates the guidance for application of the AST provided in NUREG-1465 and draft RG-1081. The change involves the use of new terminology for the acceptance criterion expressed as 5 rem TEDE. The term TEDE is defined in 10 CFR 20 as the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). The acceptance criteria of 5 rem TEDE and 5 rem whole body are not equivalent. The NRC has revised the current GDC-19 whole body dose criterion with a criterion in terms of rem TEDE for the duration of the accident in 10 CFR 50.67 for the licensee that seeks to revise its current radiological source term with an AST."

"The NRC recognizes that an analysis using the AST may represent a reduction in the margin of safety for some applications. The margin of safety is typically defined as the difference between the calculated parameters (offsite and control room dose) and the associated regulatory or safety limit. Implementing the AST in accordance with draft RG-1081 and 10 CFR 50.67 revises the acceptance criterion (regulatory limit) contained in GDC-19 to 5 rem TEDE. The calculated control room dose is below the new acceptance criterion. In 10 CFR 50.67, the rule considers the 5 rem whole body, or its equivalent to any part of the body is accounted for in the definition of TEDE and by the 5 rem TEDE annual limit. Therefore, revising the control room dose analysis using the new terminology for the AST does not involve a significant reduction in a margin of safety."

The validity of this conclusion is not affected by the response to NRC Questions 2 through 5, 8, and 9. The responses to NRC Questions 2 through 5, 8, and 9 only provide information that supports the proposed accident analyses. These responses do not alter the parameters calculated by the analyses. Therefore, the margin of safety remains as described above.

Response to NRC Questions 7 and 19

The response to NRC Question 7 describes the process that I&M intends to use for addressing differences between the proposed analyses, which is based on Draft Guide (DG)-1081, and the applicable provision of RG 1.183. The response to NRC Question 19 provides clarification that I&M is not requesting the NRC to review and approve changes made to the control room ventilation flow rates between 1982 and 1986.

The responses to NRC Questions 7 and 19 describe regulatory positions. The responses do not affect any of the proposed accident analyses, nor do they alter any of the proposed changes to the T/S. The responses are consistent with statements in the original NSHC regarding compliance with DG-1081. Therefore, there is no affect on the probability or consequences of a previously evaluated accident, the possibility of a new or different kind of accident, or the margin of safety as evaluated in the original NSHC.

Commitments Made in Responses to NRC Questions 1 - 3, 6, 8, 10 - 18, 20, and 21

In addition to providing the information described above, the responses to NRC Questions 1, 2, 3, and 8 include commitments to submit responses to portions of the respective questions in a subsequent submittal. The responses to NRC Questions 6, 10 through 18, 20, and 21 consist of commitments to submit responses to the entire question in a subsequent letter. These commitments do not, of themselves, affect any of the proposed accident analyses, nor do they alter any of the proposed changes to the T/S. Therefore, the commitments do not affect the probability or consequences of a previously evaluated accident, the possibility of a new or different kind of accident, or the margin of safety as evaluated in the original NSHC. The

subsequent submittal providing these responses will address the affect of the responses on the validity of the original NSHC.

ATTACHMENT 3 TO C0601-03

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 will provide a supplemental response which will include:<br><br>Revised technical specification pages specifying the appropriate face velocity in the Surveillance Requirement for the engineered safety features ventilation system (ESFVS) and the storage pool ventilation system (SPVS) in response to NRC Question 1.<br><br>A response to NRC Question 2 addressing iodine removal safety factors for the control room emergency ventilation system (CREVS), the ESFVS, and the SPVS with respect to accidents and events other than the fuel handling accident (FHA).<br><br>Responses to NRC Questions 3(a), 3(b) and 3(c) in their entirety.<br><br>A response to NRC Question 3(d) pertaining to two-fan operation of the CREVS.<br><br>A response to NRC Question 4 providing the basis for the unfiltered makeup assumed for events, other than the FHA, in which the CREVS is in the normal mode.<br><br>A response to NRC Question 6 in its entirety.<br><br>A response addressing the proposed analyses of accidents and events other than a FHA in response to NRC Question 8.<br><br>Responses to NRC Questions 10 through 18, 20, and 21 in their entirety.<br><br>An evaluation addressing affect of the supplemental responses on the validity of the original No Significant Hazards Consideration Evaluation. | August 17, 2001 |