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INDIANA MICHIGAN POWER

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C0102-04
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Docket No.: 50-315
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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 SECOND NUCLEAR REGULATORY COMMISSION REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR CONTROL ROOM HABITABILITY

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.

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4) Letter from M. W. Rencheck (I&M) to NRC Document Control Desk, "Final Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," (TAC Nos. MA9394 and MA9395)," C0801-02, dated August 17, 2001.

5) 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 August 16, 2001 (TAC Nos. MA9394 and MA9395).

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 to address control room habitability issues. Reference 2 transmitted an NRC Request for Additional Information (RAI) regarding the proposed amendment. References 3 and 4 transmitted I&M's responses to that RAI. Reference 5 transmitted a second RAI pertaining to the proposed amendment. In phone conferences held August 28, 2001, October 12, 2001, and November 13, 2001, members of the NRC staff identified concerns regarding I&M's responses to the first RAI.

Attachment 1 to this letter provides the information requested by the second RAI (Reference 5). Attachment 2 to this letter addresses the NRC concerns as identified in the above noted phone conferences regarding the responses to the first RAI (Reference 2). As indicated in Attachments 1 and 2, certain additional information will be provided in a subsequent letter. Attachment 3 provides a listing of new commitments made in this letter.

The information provided in this letter consists of supporting information for the amendment request previously submitted by Reference 1. The information provided in this letter does not alter the requested amendment and does not affect the validity of the original evaluation of significant hazards considerations performed in accordance with 10 CFR 50.92 as documented in Attachment 4 to Reference 1. The environmental assessment provided in Attachment 5 to Reference 1 also remains valid.

Should you have any questions, please contact Mr. Gordon P. Arent, Manager of Regulatory Affairs, at (616) 697-5553.

Sincerely,

A. C. Bakken, III
Senior Vice President, Nuclear Operations

/bjb

Attachments/Enclosure (computer disc)

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

AFFIDAVIT

I, A. Christopher Bakken, III, being duly sworn, state that I am Senior Vice President, Nuclear Operations of American Electric Power Service Corporation and 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.

American Electric Power Service Corporation

A. C. Bakken, III
Senior Vice President, Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15th DAY OF January, 2002

Julie E. Newmiller
Notary Public

My Commission Expires 8-22-04

JULIE E. NEWMILLER
Notary Public, Berrien County, MI
My Commission Expires Aug 22, 2004

ATTACHMENT 1 TO C0102-04

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

This attachment provides Indiana Michigan Power Company's (I&M) response to a Nuclear Regulatory Commission (NRC) Request for Additional Information transmitted by a 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 August 16, 2001.

NRC Question 1

The meteorological data set (ARCON96 format) provided by I&M as an attachment to the June 19, 2001, letter appears to contains data which are questionable. For example:

- *For the year 1996, stability class A was reported for 4912 hours out of the available 8760 hours; 4404 hours in 1997; and 4653 hours in 1998. These appear to be unusually large fractions and are inconsistent with historic data reported in Table 2.2-4 of the Updated Final Safety Analysis [Report] (UFSAR).*

- *There are periods in the data set in which the reported stability class did not change for numerous hours; 56 hours in one case. Given that this encompasses two diurnal cycles, the constant stability class suggests a potential instrumentation or data processing problem resulting in invalid data that perhaps should have been flagged as such.*

- *Over 25 percent of the observations of stability class A for the 3 years were reported between the evening hours of 1900 to 0700. This appears to be an untypically large fraction.*

The I&M response to Question 9 indicates that the data were validated by a meteorologist on I&M's contractor's staff to ensure that the wind speed and direction were within normal operating ranges. The response also states that invalid data were not used. The response does not explicitly state that a similar validation was performed on the stability class data. (The staff did determine that a wind rose prepared using the submitted wind speed and wind direction data showed a good correlation to the 1992 data reported in the UFSAR.)

Although the staff recognizes that local temporal meteorological conditions can often result in observations that appear askew, the large quantity of stability class A observations in the D. C.Cook data set raises a question regarding the representativeness of the reported data.

Since stability class A is generally more favorable with regard to dispersion than the other classes, the reported χ/Q values may not be adequately conservative. Please provide a suitable explanation of the conditions identified above. If the conditions described above cannot be reasonably explained, or are deemed to be the result of instrumentation or processing problems, please provide a justification of why these data are appropriate for use in determining short-term dispersion estimates for design-basis calculations.

I&M Response to NRC Question 1

Two errors have been identified in the processing of the data that was transmitted by the 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," C0601-03, dated June 19, 2001.

One error involved a failure to normalize atmospheric temperature data to the differential height assumed in the computer program used to convert meteorological data to the ARCON96 format. The NRC staff was verbally notified of this error on August 1, 2001.

The second error involved the manner in which invalid data was identified. The error resulted in invalid data being used in the calculation of new χ/Q values. The NRC staff was verbally notified of this error on September 4, 2001. As part of data re-validation activities conducted following discovery of the second error, condition reports and calibration results were reviewed to identify potential instrumentation problems. This review resulted in elimination of data from 1998 due an instrument cable problem that may have resulted in an extended period of inaccurate data.

The χ/Q values have been re-calculated using differential temperature data that has been properly normalized and meteorological data with invalid data excluded. Data from 1995 was used with the data from 1996 and 1997 to provide the three-year span required by the ARCON96 program. The data used to recalculate the χ/Q values is provided on the computer disc enclosed with this letter. The recalculated χ/Q values are summarized in Table 1.

Table 1: Recalculated χ/Q Values

Release Location	**0-2 hrs**	**2-8 hrs**	**8-24 hrs**	**1-4 days**	**4-30 days**
Unit Vent	1.77E-03	1.24E-03	5.14E-04	3.64E-04	2.77E-04
PORV	4.00E-03	3.49E-03	1.79E-03	1.15E-03	8.84E-04
Containment Surface	8.99E-04	6.29E-04	2.59E-04	1.93E-04	1.57E-04

For comparison, Table 2 below provides the χ/Q values that were calculated from the data transmitted by I&M's letter dated June 19, 2001.

Table 2: χ/Q Values calculated from Previous Data

Release Location	**0-2 hrs**	**2-8 hrs**	**8-24 hrs**	**1-4 days**	**4-30 days**
Unit Vent	1.74E-03	1.00E-03	4.19E-04	3.09E-04	2.69E-04
PORV	5.01E-04	3.16E-04	1.35E-04	1.06E-04	7.51E-05
Containment surface	8.95E-04	6.91E-04	2.79E-04	2.23E-04	2.63E-04

The recalculated χ/Q values given in Table 1 above were used to establish bounding doses for the events identified in the original amendment request transmitted in a 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 was accomplished as follows:

- The recalculated χ/Q values shown in Table 1 above were divided by the corresponding χ/Q values calculated from the data transmitted by I&M's letter dated June 19, 2001. This results in five χ/Q multiplication factors (one for each time interval) for each pathway. The χ/Q s for the power operated relief valve (PORV) ground releases were reduced by a factor of 5 as described in the response to NRC Question 3 below.

- The previous integrated doses for each event were multiplied by the single largest χ/Q multiplication factor that was applicable to the event, considering both the duration of the event and the release pathways involved.

- In some cases it was also necessary to apply the χ/Q multiplication factor for each release pathway to the dose resulting from that pathway, and/or apply the χ/Q multiplication factor for each time interval to the dose for that time interval, in order to obtain a calculated dose below the acceptance criteria.

Table 3 below provides a summary of the bounding doses that were determined as described above. As indicated in the table, the bounding doses (except those from a large break loss of coolant accident [LOCA]) were determined to be below the 10 CFR 50, Appendix A, General Design Criterion 19 limit of 5 rem total effective dose equivalent. The calculated dose from a LOCA is affected by the resolution of the NRC concern regarding I&M's response to previous Question 10 as discussed in Attachment 2 to this letter. Consistent with that response, I&M will provide the calculated dose from a LOCA in a separate letter no later than July 10, 2002.

Table 3: Recalculated Bounding Doses

Event	Dose (Rem TEDE)
Large Break LOCA	To be submitted later
Small Break LOCA	1.90
Steam Generator Tube Rupture (SGTR) with Accident Initiated Iodine Spike	0.53
SGTR with Pre-Accident Iodine Spike	1.06
Rod Ejection Accident	3.21
Fuel Handling Accident	1.74
Main Steam Line Break (MSLB) with Accident Initiated Iodine Spike	1.32
MSLB with Pre-Accident Iodine Spike	0.32
Gas Decay Tank Rupture	0.11
Volume Control Tank Rubture	0.41
Loss of Offsite Power (LOOP) with Accident Initiated Iodine Spike	4.25
LOOP with Pre-Accident Iodine Spike	1.06

The specific examples identified in NRC Question 1 are addressed as follows:

- The NRC noted that distribution of stability classes was inconsistent with that given in the Updated Final Safety Analysis Report (UFSAR). Table 4 below provides a comparison of the stability class distribution derived from the data in UFSAR Tables 2.2-4 through 2.2-10, the stability class distribution derived from the data provided in I&M's June 19, 2001, letter, and the stability class distribution derived from 1995, 1996, and 1997 data that has been recalculated to address the two errors identified above. The table shows that the distribution based on the corrected stability data is consistent with the distribution based on data from the UFSAR.

Table 4: Comparison of Stability Class Distributions

Class	UFSAR	June 19, 2001 letter	Recalculated Data
A	23%	53%	18%
B	7%	3%	6%
C	9%	3%	7%
D	33%	10%	35%
E	15%	11%	19%
F	6%	6%	7%
G	7%	13%	7%

- The NRC noted that there were periods in which the stability class did not change for numerous hours. Table 5 below identifies all instances in which the stability class remains unchanged for 24 hours or more, based on the corrected data. The table shows that the majority of periods of unchanged stability occur during the colder months (November through February). Since constant stability is more likely to occur during these months, I&M considers that the data does not indicate a potential instrumentation or data processing problem. Additionally, the stability class changes, on average, every three hours.

Table 5: Periods of Constant Stability

Year	Ending Day	Ending Hour	Stability Class	Duration (hrs)
1995	Jan. 23	1000	D	45
1995	Mar. 28	1000	D	26
1995	Oct. 8	0900	D	25
1996	Feb. 7	2300	D	32
1996	May 11	1700	A	32
1996	May 17	1300	A	25
1996	Nov. 11	0700	D	25
1996	Nov. 25	1300	A	36
1997	Jan. 16	1300	D	31
1997	Jan. 28	0900	D	42
1997	Nov. 2	1500	D	38
1997	Nov. 5	0300	D	27
1997	Nov. 30	0900	A	26

- The NRC noted that an untypically large fraction (25%) of the observations of stability class A occurred between 1900 and 0700. The corrected data shows the percent of Category A stability occurring between 1900 and 0700 to be less than 7%. I&M considers this to be typical.

In NRC Question 1, the staff also noted that I&M's letter dated June 19, 2001, did not explicitly state that a validation similar to that performed for wind speed and direction data had been performed on the stability class data. To address this concern, a description of the validation process that was applied to the wind speed data, wind direction data, and the temperature data used to determine stability class is provided below.

The data was collected at CNP in 1995, 1996, and 1997, from instruments on the site meteorological tower. Technical Specification 4.3.3.4 requires that these instruments undergo a channel check daily and calibration at least every 184 days. Each instrument has a specific range of operation that the data collection software checks it against at the time the data is collected. The data was reviewed to determine if there were any periods of bad data. This was done by

using software programs that plot, average, and summarize each data point. The first program that was run plots each parameter on an axis. All of the wind speeds were plotted on one axis, the directions on another, temperatures on another and the differential temperature was plotted separately. In this way, like parameters were easily compared so that irregularities could be identified. Additionally, a consulting meteorologist reviewed the data.

A program that determines the number of hours of each wind speed and direction group versus stability class was run for each year (1995, 1996, and 1997) and for the composite of the three years. The data from the individual years was then compared with each other and with the three-year composite to ensure that the data points were consistent. Finally, the electronic data that had been processed as described above was compared to the hard copy record to ensure that the process had not introduced errors.

In the manner described above, the data used in running the ARCON96 model was determined to be consistent in wind speed, direction, and stability classification.

NRC Question 2

The staff is unable to duplicate the χ/Q values for the containment release pathway using the data in the table provided in the June 19, 2001, letter. Please identify whether I&M used the diffuse area source mode for determining the χ/Q value. If yes, please identify the initial values for sigma-Y and sigma-Z input to ARCON96 and explain the basis of these values. The staff has previously approved diffuse area source χ/Q values that were calculated using initial sigma-Y and sigma-Z values based on the building width divided by 6 and the building height divided by 6, respectively. If I&M used a different approach to establish these initial values, please provide a justification for the approach used.

I&M Response to NRC Question 2

I&M used the diffuse area source mode for determining the χ/Q value for the containment release pathway. The initial values for sigma-Y and sigma-Z that were input to ARCON96 were 6.2 and 7.64, respectively. These values are based on containment diameter divided by 6 and the containment effective height divided by 6, respectively. The containment effective height is the height of a cylindrical building with the same radius and projected area.

NRC Question 3

The pressure-operated relief valve (PORV) release paths were treated as stack releases. The staff's regulatory guidance in Regulatory Guide 1.145 describes a stack release as a release point of which the physical height is 2½ times the height of adjacent solid structures or higher. At an elevation of about 25 meters, the PORVs do not appear to meet the definition of a stack

release. While the staff may consider plume rise in cases where the vertical velocity of the plume can be shown to exceed the wind speed by a factor of 5, the staff does not generally allow plume rise to be used in demonstrating that the 2½-times-height criterion is met. Please provide a justification for treating these release points as stack releases rather than ground level releases.

I&M Response to NRC Question 3

The χ/Q values for the PORV releases have been recalculated as a ground release with a reduction by a factor of five to account for the plume rise resulting from the vertical velocity. The vertical velocity causes the release to rise above ground level, escape building wake effects, and disperse before returning to ground level. The vertical velocity of the PORV release, approximately 75 miles per hour, exceeds the 95th percentile wind speed, 11.8 miles per hour, by more than a factor of five, and the release point is uncapped and vertically oriented. This reduction is consistent with draft guidance published by the NRC in Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Habitability Assessment at Nuclear Power Plants," dated December 2001.

NRC Question 4

I&M tabulated the stack flow and vertical velocity for some of the release points. Please explain the basis for these values and, in particular, why these values are considered to be bounding for the entire duration of the release or each of the meteorological time intervals. Include in your explanation the potential impact of single failures, loss of offsite power, actions directed by emergency operating procedures, and changes in plant parameters such as steam pressure and temperature, as appropriate.

I&M Response to NRC Question 4

The table provided in I&M's letter dated June 19, 2001, contains values for stack flow for releases from the PORVs and unit vents and values for vertical velocity for releases from the PORVs. As described below, these values are based on a conservative treatment of the condition of the plant and equipment during the event.

The PORV stack flow and vertical velocity values are those that would occur with the reactor coolant system (RCS) at its no–load temperature of 547 degrees Farenheit. This stack flow and vertical velocity are conservatively assumed to remain constant for the 30-day duration of the event. The events were modeled assuming that the plant remains at hot standby with the PORVs removing decay heat. Conditions in the steam generator upstream of the PORV may change slightly, and the number of open PORVs may change as decay heat decreases. However, when the PORV is open, the effluent velocity is expected to be at or near the value used in the analyses. The worst case 30-day dose that would result from these assumed conditions is that

associated with a LOOP. The 30-day dose from a LOOP would be 4.25 Rem TEDE. Only 5% of this dose would result from releases during the first 24 hours of the accident. Therefore, the dose from a ground release during a reasonably justifiable 24-hour cool down would be bounded by the calculated dose from a 30-day ground release with the χ/Q for the 30-day release reduced by a factor of five as described in the response to NRC Question 3.

The stack flow of the unit vent release is based on a single train of ventilation operating for the duration of the event. The value for the Unit 1 vent stack flow is higher because it includes the spent fuel pool ventilation system exhaust. There is no vertical velocity assumed for the unit vents because a discharge hood directs all flow horizontally. The unit vent flow is consistent with a loss of offsite power with the loss of one emergency diesel generator.

Additionally, a review of χ/Q data shows that the largest unit vent χ/Q value for each time period does not necessarily correspond to the largest flow rate, indicating that the unit vent flow rate is not a dominating factor in determining the χ/Q value.

ATTACHMENT 2 TO C0102-04

RESPONSES TO NUCLEAR REGULATORY COMMISSION (NRC) CONCERNS IDENTIFIED IN PHONE CONFERENCES

The following provides Indiana Michigan Power Company's (I&M) responses to NRC concerns regarding I&M's August 17, 2001, response to the first NRC Request for Additional Information (RAI), dated March 29, 2001. These concerns were identified in telephone discussions conducted August 28, 2001, October 12, 2001, and November 13, 2001, between members of the NRC staff and I&M personnel.

NRC Concern Regarding I&M Response to Question 8

The NRC identified concerns with I&M's disposition of the differences between the assumptions used in the proposed accident analyses and those identified in RG 1.183. The NRC considers that approval of alternatives to the assumptions stated in RG 1.183 must be based on a technical justification as to why the value in the RG assumption does not apply to a facility. The NRC considers that the differences between the assumptions used in the proposed accident analyses and those identified in RG 1.183 may be addressed by a commitment to use the assumptions in RG 1.183 if the analyses are re-performed in the future.

I&M Response to NRC Concern

As described in the June 19 and August 17, 2001, responses to Question 8 in the NRC's March 29, 2001, RAI, I&M considers that the assumptions used in its analyses provide results that are consistent with or conservative with respect to results that would be obtained by using the assumptions specified in RG 1.183, or are appropriate based on Donald C. Cook Nuclear Plant (CNP) design and licensing basis. Therefore, except as noted in the response to the NRC concern regarding I&M's response to Question 10, I&M does not intend to revise the previously submitted analyses to support the amendment requested in its June 12, 2000, letter. However, if any of these analyses are re-performed in the future, the analysis will be revised to use assumptions that are at least as conservative as those specified in RG 1.183 with the following exceptions:

- I&M does not intend to re-perform the small break loss of coolant accident (LOCA) control room dose analysis in the future. Analysis of a small break LOCA is not required by RG 1.183 and I&M's previous analyses demonstrated that the control room dose from a small break LOCA is bounded by that from a large break LOCA.

- For CNP, the control room dose from a locked rotor event is bounded by the dose from a loss-of-load event, which is bounded by the dose analysis for a loss-of-offsite-power event. This differs from the assumption in R.G. 1.183 that a locked rotor event would be bounded

by the dose analysis for a steam line break event. The reason that the RG 1.183 assumption does not apply to CNP is provided in the letter from M. W. Rencheck, (I&M) to NRC Document Control Desk, "Final Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request for Control Room Habitability," C0801-02, dated August 17, 2001.

NRC Concern Regarding I&M Response to Question 10

I&M's response provided justification for use of an iodine airborne fraction of 10^{-4} in the proposed large break LOCA analysis to determine the iodine released from ECCS leakage, rather than the 10^{-1} value given in RG 1.183. I&M's justification was based on laboratory experiments described in the original CNP FSAR and a theoretical study of the same vintage. The NRC considers that justification for use of an iodine airborne fraction that differs so greatly from that given from the value given in RG 1.183 requires a numeric argument linked to the actual conditions in the plant.

I&M Response to NRC Concern

I&M has determined that additional analyses are needed to resolve the NRC concern. Accordingly, I&M will provide its response to this concern in a separate letter no later than July 10, 2002.

NRC Concern Regarding I&M Response to Question 13

I&M's response to this question provided justification for the assumption of 6 hours for the termination of an accident initiated iodine spike, rather than the 8 hours stated in RG 1.183. The NRC requested that I&M provide additional information on the percentage of failed fuel and iodine appearance rate assumed in its analyses.

I&M Response to NRC Concern

For a given core iodine inventory and gap fraction, the duration of the accident-initiated iodine spike is determined by the accident-initiated iodine appearance rate and the percent of failed fuel. Values for these parameters were derived as follows.

Accident-Initiated Iodine Appearance Rate

The accident-initiated iodine appearance rate assumed in the analyses is 500 times the normal appearance rate that would be expected for operation with maximum iodine removal. The normal appearance rate that would be expected for operation with maximum iodine removal is equal to the total reactor coolant system (RCS) activity at the Technical Specification (TS) limit

of 1 micro-curie (μCi)/gram (g) of Dose Equivalent (DE) I-131, times the maximum removal rate. The maximum removal rate accounts for purification at the maximum letdown flow rate, perfect filtration, and a maximum RCS leak rate. The resulting normal appearance rate with maximum iodine removal is 0.3765 curie (Ci)/minute. The accident-initiated appearance rate used in determining the duration of the spike is 500 times 0.3765 Ci/minute, or 188.3 Ci/minute.

Percent of Failed Fuel

As described above, determination of the accident-initiated appearance rate was based on the TS limit on RCS activity of 1 μCi/g of DE I-131. Using the International Commission on Radiological Protection (ICRP) publication 30 dose conversion factors (DCFs) for thyroid doses (which are more limiting than DCFs for committed effective dose equivalent inhalation), an RCS activity of 3.92 μCi/g DE I-131 corresponds to 1% failed fuel. Therefore, the TS limit on RCS activity corresponds to about 0.26% failed fuel.

However, a more conservative value of 0.495% failed fuel was actually used in determining the duration of the accident initiated iodine spike. This value was derived by dividing the normal appearance rate determined above by a more conservative appearance rate. The more conservative appearance rate is that which would result from normal letdown flow rates and decontamination factors, and no RCS leakage, applied to the RCS I-131 inventory corresponding to 1% fuel defects. The resulting more conservative appearance rate is 0.76 Ci/minute. Dividing the normal appearance rate determined above, 0.3765 Ci/minute (based on the RCS activity limit in TS), by the more conservative appearance rate, 0.76 Ci/minute (based on 1% failed fuel), results in a value for percent of failed fuel of 0.495%.

In summary, the normal appearance rate used in the analysis, 0.3765 Ci/minute, corresponds to 0.26% of the fuel rods leaking. However, the iodine inventory used to determine the duration of the spike corresponds to 0.495% of the fuel rods leaking.

Spike Duration

The amount of iodine available for release was determined by multiplying the iodine inventory for a 12% gap fraction (1.224×10^7 Ci) by the percent of failed fuel (0.495%). The spike duration was determined by dividing the amount of iodine available for release (60,588 Ci) by the assumed appearance rate (188.3 Ci/minute). This resulted in a calculated duration of 5.36 hours, and a value of 6 hours was conservatively assumed in the dose analyses.

NRC Concern Regarding I&M Response to Question 21

In order for the staff to continue its review of the locked rotor event described in your application, the staff will need the following additional information: (1) a description of and

justification for the initial assumptions used in the new analysis, (2) a comparison of the differences in assumption between the previous and new analyses, (3) the sequence of events for the new analysis, and (4) the results of the analysis including plots of important parameters to show plant response and minimum DNBR.

I&M Response to NRC Concern

This concern was identified by the members of the NRC staff on November 13, 2001, later than the other concerns addressed above. I&M will provide its response to this concern in a separate letter no later than July 10, 2002.

ATTACHMENT 3 TO C0102-04

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 the calculated dose from a large break loss of coolant accident (LOCA) in a separate letter.	July 10, 2002
I&M will provide its response to the NRC concern regarding the fraction of iodine released from emergency core cooling system leakage assumed in the large break LOCA analysis in a separate letter.	July 10, 2002
If any of the analyses described in its June 12, 2000, amendment request are re-performed in the future, the analysis will be revised to reflect the assumptions specified in RG 1.183, except that I&M does not intend to re-perform the small break LOCA analysis, and the dose from a locked rotor event is bounded by the dose from a loss-of-load event, which is bounded by the dose analysis for a loss-of-offsite-power event.	Consistent with reanalysis date
I&M will provide its response to the NRC concern regarding the locked rotor event in a separate letter.	July 10, 2002