February 23, 1989

Docket No. 50-315

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Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-58 (TAC NO. 69064)

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in partial response to your application dated August 9, 1988, as revised January 10, 1989.

This amendment provides surveillance interval extensions for ice basket weighing and resistance temperature detector calibrations to permit Unit 1 to continue operation until the upcoming Unit 1 Cycle 10/11 refueling outage, which is now scheduled for March 1989. The editorial change to TS Table 4.3-2 proposed by your August 9, 1988 application is being handled separately.

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

original signed by

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 121 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

~~WHodges~~
J Craig
2/10/89

LA/PD31:DRSP	PM/PD31:DRSP	SRXB	(A)D/PD31:DRSP	OGC
RIngram	JStang:cr	SNewberry	TQuay	SH Lewis
2/1/89	2/1/89	2/10/89	2/13/89	2/22/89

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
February 23, 1989

Docket No. 50-315

Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

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Sincerely,

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
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Enclosures:
1. Amendment No.121 to DPR-58
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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Mr. S. Brewer
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Service Corporation
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Columbus, Ohio 43216

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated August 9, 1988, as revised January 10, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore Quay, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 23, 1989

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4 0-3	3/4 0-3
3/4 3-1	3/4 3-1
3/4 3-12	3/4 3-12
3/4 3-15	3/4 3-15
3/4 3-31	3/4 3-31
3/4 3-33	3/4 3-33
3/4 3-54	3/4 3-54
3/4 3-56	3/4 3-56
3/4 6-27	3/4 6-27

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing criteria	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.0.6 By specific reference to this section, those surveillances which must be performed on or before July 31, 1987, and are designated as 18-month surveillances (or required as outage-related surveillances) may be delayed until the end of the Cycle 9-10 refueling outage (currently scheduled to begin during the second quarter of 1987). For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1987 refueling outage.

4.0.7 By specific reference to this specification, those surveillances which must be performed on or before April 1, 1989, may be delayed until the end of the Cycle 10-11 refueling outage (currently scheduled to begin during the latter part of the first quarter of 1989.) For these specific surveillances under this section, the specified time intervals required by Specification 4.0.2 will be determined with the new initiation date established by the surveillance date during the Unit 1 1989 refueling outage.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.+

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1. *

* The provisions of Specification 4.0.6 are applicable.

\+ The provisions of Specification 4.0.7 are applicable.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1,2,3*,4*,5*
2. Power Range, Neutron Flux	S	D(2,8),M(3,8) and Q(6,8)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R (6)	M	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R (6)	M	1, 2
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2 and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)[+,**]	M	1, 2
8. Overpower ΔT	S	R(9)[+,**]	M	1, 2
9. Pressurizer Pressure--Low	S	R[+]	M	1, 2
10. Pressurizer Pressure--High	S	R[+]	M	1, 2
11. Pressurizer Water Level--High	S	R[+]	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

+ The provisions of Specification 4.0.6 are applicable.

** The provisions of Specification 4.0.7 are applicable.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2. +

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the automatic actuation logic test. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation. *, +

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3. *

* The provisions of Specification 4.0.6 are applicable.

\+ The provisions of Specification 4.0.7 are applicable.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R^{+}	M(3)	1, 2, 3
d. Pressurizer Pressure--Low	S	R^{+}	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R^{+}	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	$R^{+,*}$	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R^{+}	M(3)	1, 2, 3

+ The provisions of Specification 4.0.6 are applicable.

* The provisions of Specification 4.0.7 are applicable.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R+	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low Pressure--Low	S	R+, *	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R+	M	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R+	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R+	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	R+	1, 2

+The provisions of Specification 4.0.6 are applicable.

*The provisions of Specification 4.0.7 are applicable.

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The post-accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-11, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.*

* The provisions of Specification 4.0.7 are applicable.

TABLE 4.3-7
POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R*
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R+
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R+
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R*
6. Steam Line Pressure	M	R*
7. Steam Generator Water Level - Narrow Range	M	R*
8. RWST Water Level	M	R
9. Boric Acid Tank Solution Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R+
12. PORV Position Indicator - Limit Switches	M	R*
13. PORV Block Valve Position Indicator - Limit Switches	M	R*
14. Safety Valve Position Indicator - Acoustic Monitor	M	R*
15. Incore Thermocouples (Core Exit Thermocouples)	M	R(1)
16. Reactor Coolant Inventory Tracking System (Reactor Vessel Level Indication)	M(2)	R(3)
17. Containment Sump Level**	M	R
18. Containment Water Level**	M	R

(1) Partial range channel calibration for sensor to be performed below P-12 in MODE 3.
(2) With one train of Reactor Vessel Level Indication inoperable, Subcooling Margin Indication and Core Exit Thermocouples may be used to perform a CHANNEL CHECK to verify the remaining Reactor Vessel Indication train OPERABLE.
(3) Completion of channel calibration for sensors to be performed below P-12 in MODE 3.
* The provisions of Specification 4.0.6 are applicable.
** The requirements for these instruments will become effective after the level transmitters are modified or replaced and become operational. The schedule for modification or replacement of the transmitters is described in the Bases.

+ The provisions of Specification 4.0.7 are applicable.

shall be constituted of one basket each from Radial Rows 1, 2, 4, 6, 8 and 9 (or from the same row of an adjacent bay if a basket from a designated row cannot be obtained for weighing) within each bay. If any basket is found to contain less than 1220 pounds of ice, a representative sample of 20 additional baskets from the same bay shall be weighed. The minimum average weight of ice from the 20 additional baskets and the discrepant basket shall not be less than 1220 pounds/basket at a 95% level of confidence.

The ice condenser shall also be subdivided into 3 groups of baskets, as follows: Group 1 - bays 1 through 8, Group 2 - bays 9 through 16, and Group 3 - bays 17 through 24. The minimum average ice weight of the sample baskets from Radial Rows 1, 2, 4, 6, 8 and 9 in each group shall not be less than 1220 pounds/basket at a 95% level of confidence.

The minimum total ice condenser ice weight at a 95% level of confidence shall be calculated using all ice basket weights determined during this weighing program and shall not be less than 2,371,450 pounds.* *

3. Verifying, by a visual inspection of at least two flow passages per ice condenser bay, that the accumulation of frost or ice on flow passages between ice baskets, past lattice frames, through the intermediate and top deck floor grating, or past the lower inlet plenum support structures and turning vanes is restricted to a nominal thickness of 3/8 inches. If one flow passage per bay is found to have an accumulation of frost or ice greater than this thickness, a representative sample of 20 additional flow passages from the same bay shall be visually inspected. If these additional flow passages are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one restricted flow passage per bay is evidence of abnormal degradation of the ice condenser.*

c. At least once per 40 months by lifting and visually inspecting the accessible portions of at least two ice baskets from each 1/3 of the ice condenser and verifying that the ice baskets are free of detrimental structural wear, cracks, corrosion or other damage. The ice baskets shall be raised at least 12 feet for this inspection.

* The provisions of Specification 4.0.6 are applicable.
** The provisions of Specification 4.0.7 are applicable.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

DOCKET NO. 50-315

1.0 INTRODUCTION

By letter dated August 9, 1988, as revised January 10, 1989, the Indiana Michigan Power Company (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-58 for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed amendment would allow a one-time extension of the surveillance intervals of certain surveillances required to be performed at 9- or 18-month intervals. The extensions involve the following four groups of TS surveillances:

1. Ice basket weighing;
2. Resistance temperature detector (RTD) calibrations;
3. Ice condenser flow passage inspections; and
4. Ice condenser inlet door testing.

The proposed extensions associated with the latter two categories were withdrawn by the licensee via a letter dated January 10, 1989. This was because an unanticipated outage allowed the licensee to perform the necessary surveillances. The outage was not of sufficient duration to permit performance of the surveillances associated with the first two categories.

Accordingly, the specific surveillances considered in this evaluation are as follows:

Technical Specification	Description
1. 4.6.5.1.b.2	Ice basket weighing
2a. Table 4.3-1, Item 7	Overtemperature ΔT channel calibration
2b. Table 4.3-1, Item 8	Overpower ΔT channel calibration
2c. 4.3.2.1.2	Total interlock function test for P-12

2d. Table 4.3-2, Items 1.f and 4.d	Low-low Tavg channel calibrations
2e. Table 4.3-7, Item 2	Reactor coolant outlet temperature - T_{HOT} channel calibration
2f. Table 4.3-7, Item 3	Reactor coolant inlet temperature - T_{COLD} channel calibration
2g. Table 4.3-7, Item 11	Reactor coolant system subcooling margin monitor channel calibration

These extensions are being sought because the length of the current fuel cycle has been extended beyond an 18-month duration due to 1) operation of the unit at 90% power for most of the cycle length to lessen potential stress corrosion cracking of steam generator tubes, and 2) power coast down of the unit to avoid a dual unit refueling outage. In addition, the licensee's August 9, 1988 application proposed an editorial change to Table 4.3-2. This change is being handled separately.

2.0 EVALUATION

Group 1: Ice Basket Weighing

The licensee requests extension of the 9-month ice basket weighing required by TS 4.6.5.1.b.2. The current TS requires these surveillances to be performed by February 26, 1989. The licensee proposes to extend the surveillance interval to allow the surveillance to be performed during the Cycle 10-11 refueling outage which is scheduled to begin in March, 1989. The ice basket weighing can be partially performed at power, with personnel access limited to upper containment. Baskets in Rows 1 and 9, however, often cannot be weighed without first freeing the baskets due to their tendency to become frozen in place. This additional operation requires personnel to enter lower containment which raises ALARA concerns.

The licensee has evaluated past ice basket weights and the effect of sublimation to determine the impact of the proposed extension on the ability of the ice condenser to perform its safety function. TS 4.6.5.1.b.2, which requires weighing of ice baskets at 9-month intervals, also requires a minimum weight of 1220 pounds of ice per basket with a total ice condenser weight of 2,371,450 pounds. The minimum weight of 1220 pounds per basket contains a 10% conservative allowance for ice loss through sublimation with the intent to assure a minimum ice weight of 1098 pounds at the end of the surveillance interval.

The licensee, using data from past surveillances, has performed calculations to estimate the amount of ice that will be present in each basket at the end of the current surveillance interval including an extension period. Specifically, calculations were performed to estimate ice weights on April 1, 1989. The calculations were performed per basket for each ice condenser bay and each row group; this distinction being required by the TSs.

The first set of calculations estimated ice losses using data from the last seven surveillance intervals. The ice loss rate calculations were performed using average expected values and values at the lower 95% confidence level. These ice loss rates, both average rates and rates at the lower 95% confidence level, were then applied to the "as-left" ice weight of the latest surveillance, March 1988.

The results of the licensee's calculations performed at the lower 95% confidence level indicate that all bays except bay 24 are expected to have average basket weights above 1220 pounds. Importantly, bay 24 is expected to have a lower 95% confidence level weight of 1200 pounds, significantly above the 1098 minimum acceptable ice weight. Estimates of basket weights for row groups resulted in the prediction that all row groups except 8-3 and 9-3 are expected to have weights above 1220 pounds. Again, it is important to note that the exceptions are expected to have lower 95% confidence level weights of 1206 pounds and 1166 pounds, also significantly above 1098 pounds.

The NRC staff has considered the arguments provided by the licensee and concurs that the proposed change to TS 4.6.5.1.b.2 to allow a one time extension of the surveillance interval for weighing the ice baskets is warranted and does not present a significant safety impact. The surveillance interval extension proposed by the licensee involves a relatively brief time period of operation at power, and analysis indicates that the ice condenser, over that time period, will contain sufficient ice, adequately distributed, to perform its safety function.

Group 2: RTD Calibrations

The licensee requests extension of surveillance intervals associated with calibration of RTDs. The affected TSs are listed as Items 2.a through 2.g of Section 1 of this Safety Evaluation. The licensee indicates that the current TSs require these surveillances to be performed by March 26, 1989. The licensee proposes to perform the testing during the Cycle 10-11 refueling outage, which is expected to begin the same month. The licensee requests relief for calibration of the RTDs only. The licensee has indicated that the calibration cannot be performed at power since the calibration requires temperature data to be taken with the reactor coolant system at temperatures from the operating range (approximately 568°F) down to approximately 250°F.

The licensee has reviewed past RTD performance and has found the devices to be very stable, with no significant drift. The channels involved with the RTDs undergo channel checks and channel functional tests during power operation, which would be expected to alert plant operators to drift. Since the extension period is brief and since the equipment is subject to periodic surveillance checks, there is not likely to be any impact on safety, and therefore the NRC staff finds the one-time extension request acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

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

Date: February 23, 1989

Principal Contributor: John Stang