50-286

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 1, 1995

Mr. William J. Cahill, Jr.
Chief Nuclear Officer
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. M91732)

Dear Mr. Cahill:

The Commission has issued the enclosed Amendment No. 164 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated March 3, 1995, as supplemented April 12, 1995, and November 20, 1995.

The amendment revises the TSs to extend the calibration frequency for the following:

(1) Containment water level monitor instrumentation (specified in TS Table 4.1-1)

(2) Containment building ambient temperature sensors (specified in TS Table 4.1-1)

(3) Seismic monitoring instrumentation (specified in TS Table 4.10-2)

In addition, the amendment added a new surveillance requirement to TS Table 4.1-1 for testing the core exit thermocouples.

These changes allow operation on a 24-month fuel cycle and follow the guidance provided in Generic letter 91-04, "Changes in Technical Specification surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable.

The request to extend the Auxiliary feedwater (AFW) flow rate instrumentation (specified in TS Table 4.1-1) was withdrawn. Additional information regarding the basis for the withdrawal is provided in the safety evaluation.

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

Sincerely,

Jefferey F. Harold

Jefferey F. Harold, Acting Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 164 to DPR-64
2. Safety Evaluation

cc w/encls: See next page

William J. Cahill, Jr.
Power Authority of the State
of New York

Indian Point Nuclear Generating
Station Unit No. 3

cc:

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DATED: December 1, 1995

AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-64-INDIAN POINT UNIT 3

Docket File
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PDI-1 Reading
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cc: Plant Service list

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-286

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 164
License No. DPR-64

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 3, 1995, as supplemented April 12, 1995, and November 20, 1995, 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-64 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 164, 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LBMarsh

Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 1, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 164

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
Table 4.1-1 (sheet 3 of 6)	Table 4.1-1 (sheet 3 of 6)
Table 4.1-1 (sheet 5 of 6)	Table 4.1-1 (sheet 5 of 6)
4.10-4	4.10-4

TABLE 4.1-1 (Sheet 3 of 6)

Channel Description	Check	Calibrate	Test	Remarks
e. Main Steam Lines Process Radiation Monitors (R-62A, R-62B, R-62C, and R-62D)	D	24M	Q	
f. Gross Failed Fuel Detectors (R-63A and R-63B)	D	24M	Q	
16. Containment Water Level Monitoring System:				
a. Containment Sump	N.A.	24M	N.A.	Narrow Range, Analog
b. Recirculation Sump	N.A.	24M	N.A.	Narrow Range, Analog
c. Containment Building Water Level	N.A.	24M	N.A.	Wide Range
17. Accumulator Level and Pressure	S***	18M	N.A.	
18. Steam Line Pressure	S	24M	Q	
19. Turbine First Stage Pressure	S	24M	Q	
20a. Reactor Trip Relay Logic	N.A.	N.A.	TM	
20b. ESF Actuation Relay Logic	N.A.	N.A.	TM	
21. Turbine Trip Low Auto Stop Oil Pressure	N.A.	24M	N.A.	
22. DELETED	DELETED	DELETED	DELETED	
23. Temperature Sensor in Auxiliary Boiler Feedwater Pump Building	N.A.	N.A.	18M	
24. Temperature Sensors in Primary Auxiliary Building				
a. Piping Penetration Area	N.A.	N.A.	24M	
b. Mini-Containment Area	N.A.	N.A.	24M	
c. Steam Generator Blowdown Heat Exchanger Room	N.A.	N.A.	24M	

TABLE 4.1-1 (Sheet 5 of 6)

Channel Description	Check	Calibrate	Test	Remarks
37. Core Exit Thermocouples	D	N.A.	18M	
38. Overpressure Protection System (OPS)	D	18M	18M	
39. Reactor Trip Breakers	N.A.	N.A.	TM(1)	1) Independent operation of under-voltage and shunt trip attachments
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
40. Reactor Trip Bypass Breakers	N.A.	N.A.	(1)	1) Manual shunt trip prior to each use
			24M(2)	2) Independent operation of under-voltage and shunt trip from Control Room manual push-button
			24M(3)	3) Automatic undervoltage trip
41. Reactor Vessel Level Indication System (RVLIS)	D	18M	N.A.	
42. Ambient Temperature Sensors Within the Containment Building	D	24M	N.A.	
43. River Water Temperature # (installed)	S	18M	N.A.	1) Check against installed instrumentation or another portable device
44. River Water Temperature # (portable)	S (1)	Q (2)	N.A.	2) Calibrate within 30 days prior to use and quarterly thereafter
45. Steam Line Flow	S	24M	Q	Engineered Safety Features circuits only

TABLE 4.10-2

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS			
INSTRUMENTS AND SENSOR LOCATION	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. EL 46' -0" VC Base Mat	M*	24M	SA
b. EL 99' -0" VC Wall	M*	24M	SA
2. Triaxial Peak Accelographs			
a. STM GEN #31	NA	24M	NA
b. RC Pump #31	NA	24M	NA
c. Pressurizer	NA	24M	NA
3. Triaxial Response-Spectrum Recorders			
a. EL 46' -0" VC Base Mat**	M	24M	SA

24M - At least once per 24 months

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* Except seismic trigger
** With reactor control room indications.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 164 TO FACILITY OPERATING LICENSE NO. DPR-64

POWER AUTHORITY OF THE STATE OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

1.0 INTRODUCTION

By letter dated March 3, 1995, as supplemented April 12, 1995, and November 20, 1995, the Power Authority of the State of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 3 (IP3) Technical Specifications (TSs). The April 12 and November 20, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration. The requested changes would revise the TSs to extend the calibration frequency for the following:

(1) Containment water level monitor instrumentation (specified in TS Table 4.1-1)

(2) Containment building ambient temperature sensors (specified in TS Table 4.1-1)

(3) Seismic monitoring instrumentation (specified in TS Table 4.10-2)

In addition, the amendment added a new surveillance requirement to TS Table 4.1-1 for testing the core exit thermocouples.

A fourth revision to the auxiliary feedwater (AFW) flow rate instrumentation (specified in TS Table 4.1-1), extending the calibration frequency, was withdrawn by letter dated November 20, 1995.

These changes would allow operation on a 24-month fuel cycle and follow the guidance provided in Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," as applicable. The licensee commenced operating on a 24-month fuel cycle, instead of the previous 18-month fuel cycle, with fuel cycle 9. Fuel cycle 9 started in August 1992; however, the facility has been shut down since February 1993 for a "Performance Improvement Outage" and recently restarted in June 1995.

2.0 EVALUATION

Improved reactor fuels allow licensees to consider an increase in the duration of the fuel cycle for their facilities. A longer fuel cycle increases the

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time interval between refueling outages and the performance of TS surveillance requirements. GL 91-04 provides guidance to support the development of TS revisions to allow 24-month surveillance intervals and includes requirements to evaluate the effect on safety for an increase in surveillance testing and calibration intervals to accommodate a 24-month fuel cycle.

As justification for extending surveillance intervals, the licensee's evaluations should conclude that the net effect on safety is small, that historical plant maintenance and surveillance data support the proposed extended surveillance interval, and that the assumptions in the plant licensing basis are still bounding with the incorporation of a 24-month surveillance interval. The licensee should also address the issue of instrumentation errors/setpoint methodology assumptions when proposing an extended instrumentation calibration interval. Specifically, the licensee must evaluate the effects of an increased calibration interval on instrument uncertainties, equipment qualification, and vendor maintenance requirements to ensure that an extended surveillance interval does not result in exceeding the assumptions stated in the safety analysis.

GL 91-04 provides for either vendor drift data or plant-specific data to be used in the determination of the instrument drift term for a 24-month (maximum of 30 months) surveillance interval. Vendor information and/or licensee operating experience can provide sufficient data to evaluate long-term instrument performance and provide a basis to support an extended surveillance interval of 24 months.

The NRC staff previously reviewed and approved the licensee's methodology for extending instrument calibration intervals to accommodate a 24-month fuel cycle by letter dated December 12, 1993 (Amendment No. 140 to Facility Operating License No. DPR-64). Amendment No. 140 extended Reactor Protection System (RPS) instrument calibrations intervals. By letter dated January 11, 1994, the NRC staff subsequently issued Amendment No. 142 which approved surveillance interval extensions for the Engineered Safety Features (ESF) systems and by letter dated September 1, 1994, the staff issued Amendment No. 150 which approved surveillance interval extensions for ESF indicating instrumentation. Both Amendment Nos. 142 and 150 referenced the methodology used in Amendment No. 140. As described in Amendment Nos. 140, 142, and 150, to justify an increase in calibration interval for instruments that are used to perform safety functions, the licensee addressed a number of issues in its evaluation of extended calibration intervals. The licensee considered past equipment performance and the effect on system safety functions, the results of drift analysis, the importance of the calibration procedure or refueling test in demonstrating equipment operability, and the results of loop accuracy/setpoint calculations and the effect on safety system setpoints.

In order to evaluate past instrument performance, the licensee reviewed applicable surveillances and recorded the historical as-left and as-found drift information. The past drift data was then compared to theoretical Vendor Drift Allowances (VDA) for the corresponding calibration time intervals in order to confirm that each individual instrument drift value was within

acceptable limits for that calibration interval. If no vendor drift limits were available, the past drift data was compared to the calibration tolerance of the instrument.

To predict the future expected drift over a 30-month interval, a Maximum Expected Drift for 30 months (MED30) was determined from the average and standard deviation of actual field calibration data statistically extrapolated to a 30-month period. The method used for extrapolation is consistent with the techniques used for evaluating past performance and is allowed by the Instrument Society of America (ISA) RP67.04, Part II, Draft 9, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," dated March 1991. The loop accuracy/setpoint calculations were updated to include conservative values for 30-month calibration uncertainties (vendor specified uncertainties or MED30, whichever is larger) by accounting for instrument inaccuracies consistent with industry methods described in ISA RP67.04.

IP3's on-line surveillance program as described in Amendment No. 140 provides assurance that potential instrument failures are detected and that operability of the instrument channel is verified. The licensee has committed to establishing a drift monitoring program that will monitor and assess the effects of increased calibration intervals on instrument drift. This program will verify that actual future drift values are within projected limits and will identify instrumentation whose drift is not within the extrapolated values used in the loop accuracy/setpoint calculations.

The same methodology used for Amendment Nos. 140, 142, and 150 were used as justification for extending the surveillance intervals for the indicating instruments discussed below.

2.1 Calibration of Containment Temperature Channels

The operator utilizes containment temperature detectors to monitor the containment atmosphere temperature. Containment temperature is sensed by five detectors, each located by the inlet of a recirculation fan-cooler unit. The average of the temperature devices is displayed on a single indicator in the Control Room. Indication of containment temperature is not required for accident mitigation, and is used in the Emergency Operating Procedures (EOPs) as an indication of adverse conditions in containment.

The licensee Instrument Drift Analysis (IDA) results confirm that the past instrument drift of each of the temperature indicators were within the VDA. The loop accuracy calculations confirmed that increases in drift associated with the longer operating cycle will have negligible effect on the EOPs and plant shutdown.

Based on the information provided by the licensee, the staff finds acceptable, extending to 24 months, the calibration interval for the Ambient Temperature Sensors Within The Containment Building. In addition, to maintain radiation

dosage to plant personnel "as low as reasonably achievable," calibration of these devices should be performed when the reactor is shutdown.

2.2 Calibration of Containment Water level monitoring System Channels

Continuous indication of containment water level during and after an accident is provided by three level measuring systems: Containment sump (EL. 38'3") level, Recirculation sump (EL. 34'0") Level, and Containment building (EL. 46'0") level. Each level measuring system is comprised of two redundant loops; each loop consists of a sensor and a transmitter located inside the containment building, and a recorder and power supply in the control room.

The residual heat removal (RHR) pumps take suction from the containment sump and discharge through the RHR heat exchangers into the reactor coolant system (RCS) cold legs. Containment sump level is monitored by narrow range level transmitters (LT-1255 and LT-1256) located at the 38'3" elevation in containment. These level components continuously indicate level from 0' to 10' and provide water level inventory information to ensure that sufficient water exists for recirculation.

The recirculation pumps trade suction from the recirculation sump and discharge through the RHR heat exchangers into the RCS cold legs via the accumulator feed lines. The recirculation sump level is continuously monitored by transmitters (LT-1251 and LT-1252) located at the 34'0" elevation of the containment building. These level components check that water inventory for the operation of the emergency core cooling system exists and can be used for identifying leaks within containment.

Wide range containment level transmitters (LT-1253 and LT-1254), located at the 46'0" elevation, ensure that sufficient water exists to satisfy the recirculation and RHR pump Net Positive Suction Head (NPSH) requirements, in addition to monitoring the water level to prevent it from exceeding the design flood level. Containment level monitoring (wide range) is required by Regulatory Guide 1.97 as Type A, Category 1 instrumentation. The L1253 and L1254 indicating loops provide post-accident containment water level indication to the control room to ensure that plant safety functions are being accomplished.

The narrow range Barton transmitters (LT-1251, 1252, 1255, and 1256) were recently replaced (in 1990 and 1992) with more accurate Foxboro electronic transmitters with Barton sealed reference legs. Both wide range Barton transmitters were also replaced in 1990. These modifications improved overall channel accuracy for both the wide and narrow range. Future instrument drift for the narrow range Foxboro transmitters could not be statistically predicted because of lack of data points. Although a credible drift analysis could not be completed for the containment and recirculation sump level instrumentation, the calibration interval for these channels can be safely extended with the longer operating cycle. These instruments are required by Regulatory Guide 1.97 as Type B instrumentation (i.e., backup indication). Containment sump and recirculation sump level are not relied upon in the EOPs for accident

mitigation. Rather, the EOPs rely on the wide range recorders to provide level information. As a result, postulated increases in drift will have no affect on accident mitigation or safe plant shutdown for design basis accidents.

The containment building water level (wide range) instrumentation is designated by RG 1.97 as Type A, Category 1, and are relied on in the EOPs for accident mitigation. Future instrument drift was predicted with a 95% probability at a high confidence level. The MED30 was calculated for the level transmitters at a 95% confidence level.

The loop accuracy calculations for containment building water level (wide range) were revised using the larger uncertainty values. The revised calculations show that postulated increases in drift result in slightly larger loop inaccuracies. These inaccuracies are accommodated with a slight increase in the EOP setpoint for transfer to cold leg recirculation during an adverse containment condition. No changes are required to the other EOP steps which rely on containment water level.

The calibration interval for the containment sump, recirculation sump and containment building level components can be extended to a maximum of 30 months because conservative projections of instrument drift have been applied to the setpoint analyses. The resultant EOP setting change ensures that postulated drift will not affect the operators ability to mitigate the consequences of an accident or evaluate post-accident conditions. Based on the information provided by the licensee, the NRC staff finds acceptable, extending to 24 months, the calibration interval for the containment water level monitoring system channels. In addition, in order to keep the radiation dose to plant personnel "as low as reasonably achievable," calibration of these devices should only be performed when the reactor is shutdown.

2.3 Calibration of Seismic Instrumentation Channels

The TSs require that operability of the seismic instrumentation be demonstrated to ensure that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of plant safety features. Table 4.10-2 of the TSs requires that seismic monitoring instrumentation be calibrated once per 18 months. Below is an evaluation of extending the calibration intervals to accommodate the 24-month cycle for each type of instrument listed in Table 4.10-2.

2.3.1 Triaxial Time-History Accelographs (Table 4.10-2, item 1)

Two Kinemetrics strong motion accelographs are installed in the containment building. One at the 46'0" elevation on the containment base mat and one on the containment structure wall at the 99'0" elevation directly above the lower unit. These units are designed to detect and record earth vibrations in three directions; one channel in the vertical plane, and two channels (transverse and longitudinal) in the horizontal plane. The signals from the system are

recorded on a magnetic tape cassette in the control room. The strong motion accelograph system is comprised of one playback unit, two recorders and two transducers.

Instrument drift was evaluated to determine whether the calibration interval for the Kinemetrics components could be safely extended to a maximum of 30 months (24 months + 25%). Overall, past performance of these components tend to be within the calibration tolerance (CT). In some cases, however, past performance exceeded the CT, but discussions with Kinemetrics (the manufacturer) confirmed that this drift has a negligible effect on overall instrument system performance. During these conversations, the manufacturer stated that the strong motion accelographs should not experience unacceptable drift during an extended calibration interval, provided monthly channel checks and semi-annual functional tests are performed. Monthly channel checks and semi-annual functional tests of these seismic instruments are required by TS Table 4.10-2. The monthly channel check verifies that the "Event Indicator" is reading as required; the channel functional test performs all channel checks, tests frequency and damping of all sensor units, and performs a visual inspection of all accessible units. In summary, on-line testing provides assurance that the instruments are functioning as required and should detect any obvious instrumentation failures.

Based on the information provided by the licensee, the NRC staff finds acceptable, extending to 24 months, the calibration interval for the Triaxial Time-History Accelograph instrument channels.

2.3.2 Triaxial Peak Accelographs (Table 4.10-2, item 2)

One Engdahl peak acceleration recorder is installed on each of the following pieces of equipment: one steam generator, one reactor coolant pump, and the pressurizer. These self-contained, passive devices (they require no external power or control connections) sense and record peak accelerations tri-axially. Permanent records are scribed by a diamond stylus on replaceable metal plates. Acceleration data is obtained from the recorder plate by measuring the displacement of the scratched record from the zero line.

The refueling surveillance for the peak acceleration recorder includes: 1) visually inspecting the recorder for physical damage, contamination and corrosion, 2) performing a damping calibration, and 3) checking the acceleration sensitivity of the three sensors.

A review of past calibration data (1985 to 1992) shows physical damage, contamination, and corrosion have not been a concern and that the damping and acceleration sensitivity for the peak acceleration recorder has always been within acceptable limits. A comparison between "as-found" and "as-left" damping calibration data confirms that instrument drift has been negligible and instrument performance is not time dependent. As a result, extension of the calibration interval to a maximum of 30 months can be accommodated.

The peak acceleration recorder is a completely self-contained device of generally simple design, and therefore very reliable. Failures of these instruments due to the longer calibration interval are not expected.

Based on information the provided by the licensee, the NRC staff finds acceptable, extending to 24 months, the calibration interval for the Triaxial Peak Accelograph instrument channels.

2.3.3 Triaxial Response-Spectrum Recorders (Table 4.10-2, item 3)

Three Engdahl peak shock recorders are installed in a triaxial mount at the 46'0" elevation on the base mat. They are completely self-contained, passive devices covering the range of 2 to 25 Hertz in 1/3 octave increments. Twelve reeds of different lengths and weights, one for each frequency, are fabricated from spring steel. A diamond-tipped stylus is attached to the free end of each reed to inscribe a permanent record of the deflection on one of twelve record plates. A calibration sheet for each recorder lists the resonant frequency and "g"-sensitivity of each reed. These units are always in operation and energize the appropriate alarm lights on the peak shock annunciator in the control room. Following a seismic event, the record plates are removed and the etches measured to determine the force exerted on the containment base mat.

The refueling surveillance test for the three peak shock recorders includes: 1) visually inspecting the annunciator for proper mounting, identification and appearance, 2) verifying that all the lamps on the peak shock annunciator illuminate as required, 3) measuring and recording the frequency, damping, and acceleration sensitivity of each of the twelve reeds in each sensor, 4) measuring and recording "g" levels of switch activation, and 5) making necessary adjustments, if possible, to components which exceed manufacturer's specifications.

A review of past calibration data (1985 through 1992) for the three peak shock recorders (PSRs) shows that the damping and frequency of the PSRs have always been within acceptance criteria. In addition, switch settings for the PSRs have always been within tolerance and no adjustments were required. This evaluation confirms that instrument drift associated with these recording devices has been negligible.

In addition to the refueling interval channel calibration, the TSs also require a monthly channel check and a semi-annual channel functional test. The monthly test checks the function of the red and amber lights on the peak shock annunciator. This test provides an immediate check that the annunciator is functioning correctly and should detect failures of the relays to energize.

Surveillance testing of the seismic instrumentation (all three types) can be safely extended with the longer operating cycle because: 1) a review of past calibration data confirm that instrument drift has been generally negligible, and test results do not appear to be time dependent, and 2) on-line testing provides assurance that the strong motion accelographs and peak shock

recorders are functioning as required. Based on the information provided by the licensee, the NRC staff finds acceptable, extending to 24 months, the calibration interval for the Triaxial Response-Spectrum Recorder instrument channels. In addition, in order to keep the radiation dose to plant personnel "as low as reasonably achievable," calibration of these devices should only be performed when the reactor is shut down.

2.4 Testing of Core Exit Thermocouples

There are 65 incore thermocouples to monitor temperatures above the exit flow end of the fuel assemblies. Of the 65 thermocouples, 10 per train have been assigned as a qualified core exit thermocouple system. The core exit thermocouple system is designed to function under all plant normal, abnormal and accident conditions.

The qualified core exit thermocouple system is used to meet the requirements of RG 1.97 to monitor the reactor core temperature and is used in the EOPs. The TSs do not currently require testing of the core exit thermocouple system, but require the system to be operable. This operability criteria is met by a qualified core exit thermocouple functional test which is divided into four major sections. Operability criteria is met if:

1) The difference between the highest and lowest "as-found" thermocouple readings (in each train) of the RVLIS local display is less than or equal to 10 °F.

2) The difference between the "as-found" thermocouple indications on the RVLIS local display and the corresponding QSPDS reading does not exceed 20 °F.

3) The difference between the "as-found" RTD indications at the thermocouple reference junction box and the RVLIS local display does not exceed 5 °F.

4) The difference between the tabulated "as-found" RTD indications and the manufacturers R vs. T curves does not exceed 5 °F.

Further, the licensee clarified in their October 31, 1995, letter, that the RTDs referred to in the above third and fourth sections of the operability criteria are not related to the reactor coolant RTDs, but provides temperature compensation for the thermocouple reference junction box.

This application proposes to revise item 37 of Table 4.1-1 to require a core exit thermocouple test every 18 months to ensure operability of the system. As a result of the instrument drift evaluation findings, a 24-month testing interval is not being pursued at this time. Based on the information provided by the licensee, the NRC staff finds the 18-month calibration interval for the Core Exit Thermocouples instrument channels instrument acceptable.

3.0 SUMMARY

The licensee has evaluated the proposed changes in accordance with the guidance contained in GL 91-04. Its evaluation has concluded that the net effect on safety is small, that historical plant maintenance and surveillance data support the proposed extended surveillance intervals, and that the assumptions in the plant licensing basis are still bounding with the incorporation of a 24-month surveillance interval. The loop accuracy/setpoint calculations have shown that sufficient margin exists between the analytical limits and the corresponding TS trip setpoints and that the assumptions of the safety analysis are not violated. A drift monitoring program will assess the effect of a longer calibration interval on instrument drift. The licensee will evaluate and monitor any component showing as-found conditions beyond the specified acceptance criterion.

Based on the information provided by the licensee, the NRC staff finds the proposed TS changes to increase the calibration surveillance interval from 18 to 24 months (30 months with grace period) for containment water level monitor instrumentation, containment building ambient temperature sensors, and seismic monitoring instrumentation. The NRC staff also finds acceptable the addition of a new surveillance requirement of 18 months to TS Table 4.1-1 for testing the core exit thermocouples.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 24917). Accordingly, the 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 the amendment.

6.0 CONCLUSION

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

Principal Contributors: J. Harold
N. Conicella

Date: December 1, 1995

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

Sincerely,

Original signed by:

Jefferey F. Harold, Acting Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosures: 1. Amendment No. 164 to DPR-64
2. Safety Evaluation

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12