Docket File

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

January 15, 1981

Docket No. 50-286

RECEIVED FEB 03 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. George T. Berry, President
and Chief Operating Officer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 34 to Facility Operating License No. DPR-64 for the Indian Point Nuclear Generating Unit No. 3. This amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated September 29, 1980, November 7, 1980, and two applications dated October 31, 1980.

The amendment revises the Technical Specifications in several areas to make them more consistent with the Standard Technical Specifications, revises the Technical Specifications to assure at least 23 feet of water over the top of the reactor pressure vessel flange during movement of fuel assemblies, revises the Technical Specification to require that two valves in the miniflow line for the Residual Heat Removal Pumps be kept open, and revises the license condition dealing with steam generator inspection.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 34 to DPR-64
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
See next page

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

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. 34
License No. DPR-64

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

   A. The applications for amendment by Power Authority of the State of New York (the licensee) dated September 29, 1980 and November 7, 1980, and two applications dated October 31, 1980, comply 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 applications, 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.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, by changing paragraphs 2.C(2) and 2.J to read as follows:

   2.C(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 34 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   2.J The plant shall be brought to the cold shutdown condition within twelve equivalent months of operation from achieving criticality after the Cycle 3 mid-cycle outage, but in any event, no later than February 1, 1982. For the purpose of this requirement, equivalent operation is defined as operation with reactor coolant temperature greater than 350°F. An inspection of all four steam generators shall be performed and Nuclear Regulatory Commission approval shall be obtained before bringing the reactor critical following this inspection.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 15, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 34

FACILITY OPERATING LICENSE NO. DPR-64

DOCKET NO. 50-286

Revise Appendix A as follows:

Remove Pages	Insert Pages
1-4	1-4
1-5	1-5
3.1-12	3.1-12
3.1-13	3.1-13
3.1-21	3.1-21
3.1-25	3.1-25
3.3-2	3.3-2
3.3-3	3.3-3
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-10	3.3-10
3.3-13	3.3-13
3.3-15	3.3-15
3.6-1	3.6-1
3.7-2	3.7-2
3.7-3	3.7-3
	3.7-3a
3.8-2	3.8-2
3.8-3	3.8-3
3.8-4	3.8-4
3.10-4	3.10-4
3.10-7	3.10-7
3.10-14	3.10-14
3.10-15	3.10-15
3.10-16	3.10-16
3.11-1	3.11-1
3.12-1	3.12-1
4.4-3	4.4-3
5.3-2	5.3-2
5.4-1	5.4-1

1.9.2 Instrument Channel Functional Test

Injection of a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

1.9.3 Instrument Channel Calibration

Adjustment of channel output such that it responds, with acceptable range and accuracy, to know values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the channel functional test.

1.9.4 Logic Channel Functional Test

The operation of relays or switch contacts, in all the combinations required, to produce the required output.

1.10 CONTAINMENT INTEGRITY

Containment integrity is defined to exist when:

1.10.1 All non-automatic containment isolation valves which are not required to be open during accident conditions, except those required to be open for normal plant operation or testing as identified in Table 3.6-1, are closed and blind flanges are installed where required.

1.10.2 The equipment door is properly closed.

1.10.3 Both doors in each personnel air lock are properly closed unless being used for entry, egress or maintenance, at which time at least one air lock door shall be closed.

1.10.4 All automatic containment isolation valves are either operable or in the closed position, or isolated by a closed manual valve or flange that meets the same design criteria as the isolation valve.

1.11 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio shall be the ratio of the maximum upper excore dector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

1.12 SURVEILLANCE INTERVAL

Each Surveillance Requirement, with the exception of those with shift and daily frequencies, shall be performed within the specified time interval with:

a. A maximum allowable extension not to exceed 25% of the surveillance interval; and

b. A total maximum combined interval time for any three (3) consecutive surveillance interval not to exceed 3.25 times the specified surveillance interval.

1.13 OPERATION IN A DEGRADED MODE

The plant is said to be operating in a degraded mode when it is operating with one or more systems listed herein inoperable as permitted by the Technical Specifications. If inoperable components or systems are subsequently made operable, the action statements requiring plant shutdown no longer apply.

1.14 $\bar{E}$- AVERAGE DISINTEGRATION ENERGY

Noble gas $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 10 minutes, making up at least 95% of the total activity in the coolant.

1.15 DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

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C. MINIMUM CONDITIONS FOR CRITICALITY

1. Except during low power physics test, the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive.

2. The reactor shall not be brought to a critical condition until the pressure temperature state is to the right of the criticality limit line shown in Figure 3.1-1.

3. At all times during critical operation, T_{avg} should be no lower than $450^{O}F$.

4. The reactor shall be maintained subcritical by at least $1\%\frac{\Delta k}{k}$ until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.(1)(2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.(1)(2) Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of an increase in moderator temperature. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken.

The requirement that the reactor is not to be made critical except in accordance with Figure 3.1-1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1% subcriticality will assure that the Reactor Coolant not be solid when criticality is achieved.

References:

L. FSAR Table 3.2.1-1

2. FSAR Figure 3.2.1-9

F. LEAKAGE OF REACTOR COOLANT

Specification

1. If leakage of reactor coolant is indicated by the means available such as water inventory balance, monitoring equipment or direct observation a follow-up evaluation of the safety implications shall be initiated as practicable but no later than within 4 hours. Any indicated leak shall be considered to be a real leak until it is determined that the indicated leak cannot be substantiated by direct observation or other indication.

2. If the leakage rate, excluding controlled leakage sources such as the Reactor Coolant Pump Controlled Leakage Seals and Leakage into Closed Systems, exceeds 1 gpm and the source of leakage is not identified, reduce the leakage rate to within limits within four hours or be in hot shutdown within the next six hours and in cold shutdown within the following 30 hours.

3. If the sources of leakage are identified and the results of the evaluation are that continued operation is safe, operation of the reactor with a total leakage, other than from controlled sources or into closed systems, not exceeding 10 gpm shall be permitted except as specified in 3.1.F.4 below.

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Measurement of the leakage rate to the containment atmosphere in also possible through humidity detection and condensation collection and measurement. However, it is expected that the containment activity method will give the initial indication of coolant leakage. The other methods will be employed primarily to confirm that leakage exists, to indicate the location of the leakage sources, and to measure the leakage rate.

As described above, the four reactor coolant leak detection systems are based on three different principles, i.e., activity, humidity and condensate flow measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the containment.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Four hours is allowed from the time of leakage detection to identify the leakage source and to measure the leakage rate. This time period is required since identification and quantification of leakage sources of less than ten gallons per minute require a careful gathering and evaluation of data and/or a visual inspection of the reactor coolant system.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system. Cracks having a primary-to-secondary leakage less than 500 gallons per day during operation will have an

c. One residual heat removal pump and heat exchanger together with the associated piping and valves operable.

d. One recirculation pump together with its associated piping and valves operable.

2. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.1 within 1 hour . the reactor shall be in the cold shutdown condition within the next 20 hours.

3. The reactor coolant system T_{avg} shall not exceed 350°F unless the following requirements are met:

a. The refueling water storage tank contains a minimum of 346,870 gallons of water at a boron concentration of at least 2000 ppm.

b. The boron injection tank contains 900 gallons of a boric acid solution of 11-1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) at a temperature of at least 145°F. Two channels of heat tracing shall be operable for that portion of the flow path bounded by the boron injection tank inlet and outlet motor operated valves and the recirculation flow path to and from the boric acid tanks.

c. The four accumulators are pressurized between 600 and 700 psig and each contains a minimum of 800 ft^3 and a maximum of 815 ft^3 of water at a boron concentration of at least 2000 ppm. Accumulator isolation valves 894A, B, C, and D shall be open and their power supplies de-energized whenever the reactor coolant system pressure is above 1000 psig.

5. If the Safety Injection and Residual Heat Removal Systems are not restored to meet the requirements of 3.3.A.3 within the time periods specified in 3.3.A.4; then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and the cold shutdown condition within the following 24 hours.

   b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25^{o}F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown procedures. The shutdown shall start no later than the end of the 48 hour period.

B. Containment Cooling and Iodine Removal Systems

1. The reactor shall not be brought above the cold shutdwon condition unless the following requirements are met:

   a. The spray additive tank contains a minimum of 4000 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

   b. The five fan cooler-charcoal filter units and the two spray pumps, with their associated valves and piping, are operable.

2. The requirements of 3.3.B.1 may be modified to allow any one of the following components to be inoperable at one time:

a. Fan cooler unit 32, 34, or 35 or the flow path for fan coolant unit 32, 34, or 35 may be out os service for a period not to exceed 24 hours, provided both containment spray pumps are demonstrated to be operable.

OR

Fan cooler unit 31 or 33, or the flow path for fan cooler unit 31 or 33 may be out of service for a period not to exceed 7 days provided both containment spray pumps are demonstrated daily to be operable.

b. One containment spray pump may be out of service for a period not to exceed 24 hours, provided the five fan cooler units are operable and the remaining containment spray pump is demonstrated to be operable.

c. Any valves required for the functioning of the system during and following accident conditions may be inoperable provided it is restored to an operable status within 24 hours and all valves in the system that provide the duplicated function are demonstrated to be operable.

3. If the Containment Cooling and Iodine Removal are not restored to meet the requirements of 3.3.B.1 within the time period specified in 3.3.B.2, then:

a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and in the cold shutdown condition within the following 24 hours.

b. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respective, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

C. Isolation Valve Seal Water System (VSWS)

1. The reactor shall not be brought above cold shutdown unless the following requirements are met:

   a. The IVSWS shall be operable.

   b. The IVSW tank shall be maintained at a minimum pressure of 45 psig and contain a minimum of 144 gallons of water.

2. The requirements of 3.3.C.1 may be modified to allow any one of the following components to be inoperable at any one time:

   a. Any one header of the IVSWS may be inoperable for a period not to exceed 4 consecutive days.

   b. Any valve required for the functioning of the system during and following accident conditions provided it is restored to an operable status within 4 days and all valves in the system that provide a duplicate function are demonstrated to be operable.

3. If the IVSW System is not restored to an operable status within the time period specified, then:

   a. If the reactor is critical, it shall be brought to the hot shutdown condition utilizing normal operating procedures. The shutdown shall start no later than at the end of the specified time period.

3. If the Component Cooling System is not restored to meet the requirements of 3.3.E.1 within the time periods specified in 3.3.E.2, then:

   a. If the reactor is critical, it shall be in the hot shutdown condition within four hours and in the cold shutdown condition within the following 24 hours.

   b. If the reactor is subscritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values. If the requirements of 3.3.A.3 are not satisfied within an additional 48 hours, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures. The shutdown shall start no later than the end of the 48 hour period.

F. Service Water System

1. The reactor shall not be brought above cold shutdown unless three service water pumps on the designated essential header and a minimum of two service water pumps on the designated non-essential header together with their associated piping and valves are operable.

2. When the reactor is above cold shutdown and if the requirements of 3.3.F.1 cannot be met within twelve hours, the reactor shall be brought to the cold shutdown condition, starting no later than the end of the twelve hour period, utilizing normal operating procedures.

2. The requirements of 3.3.H.1 may be modified as follows:

   a. The control room ventilation system may be inoperable for a period not to exceed seventy-two hours. At the end of this period if the mal-condition in the control room ventilation system has not been corrected, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If after an additional 48 hours the mal-condition still exists, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of startup, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation, and, therefore, the minimum required engineered safeguards and auxiliary cooling systems are required to be operable.

The probability of sustaining both a major accident and a simultaneous failure of a safeguards component to operate as designed is necessarily very small. Thus, operation with the reactor above the cold shutdown condition with minimum safeguards operable for a limited period does not significantly increase the probability of an accident having consequences which are more severe than the Design Basis Accident.

The operable status of the various systems and components is demonstrated by periodic tests defined by Specification 4.5. A large fraction of

Assuming the reactor has been operating at full rated power, the magnitude of the decay heat decreases after initiating hot shutdown. Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 1 hour of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case the reactor is to be put into the cold shutdown condition.

The limits for the Boron Injection Tank, Refueling Water Storage Tank, and the accumulators insure the required amount of water with the proper boron concentration for injection into the reactor coolant system following a loss-of-coolant accident is available. These limits are based on values used in the accident analysis.[9] [13]

The specified quantities of water for the RWST include unavailable water (4687 gals) in the tank bottom, inaccuracies (1406 gals) in the alarm setpoints, and minimum quantities required during injection (246,000 gals)[3] and recirculation phases (80,000 gals).[4] The minimum RWST (e.g., 346,870 gals) provides approximately 13,370 gallons margin.

The four accumulator isolation valves (894 A. B, C, D) are maintained in the open position when the reactor coolant pressure is above 1000 psig to assure flow passage from the accumulators will be available during the injection phase of a loss-of-coolant accident. Indication is also provided on the monitor light panel, should any of these valves not be in the full open position even with the valve operator de-energized. The 1000 psig limit is derived from the minimum pressure requirements of the accumulators combined with instrument error and an operational band and is based upon avoiding inadvertent injection into the reactor coolant system. The accumulator isolation valve motor operators are de-energized to prevent an extremely

3.6 CONTAINMENT SYSTEM

Applicability

Applies to the integrity of reactor containment.

Objective

To define the operating status of the reactor containment for plant operation.

Specification

A. Containment Integrity

1. The containment integrity (as defined in 1.10) shall not be violated unless the reactor is in the cold shutdown condition. However, those non-automatic valves listed in Table 3.6-1, may be opened if necessary for plant operation and only as long as necessary to perform the intended function.
2. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is sufficient to maintain the shutdown margin $\geq 10\% \frac{\Delta k}{k}$.
3. If the containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within one hour or the reactor shall be in the hot shutdown condition within six hours and in cold shutdown condition within the next 30 hours.

B. Internal Pressure

If the internal pressure exceeds 2.5 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected or the reactor shutdown.

and is in addition to the fuel requirements for other nuclear units on the site.

6. Three batteries plus three chargers and the D. C. distribution systems operable.

7. No more than one 120 volt A. C. Instrument Bus in the backup lighting supply.

B. The requirements of 3.7.A may be modified to allow any one of the following power supplies to be inoperable at any one time.

1. One diesel or any diesel fuel oil system or a diesel and its associated fuel oil system may be inoperable for up to 72 hours provided the 138 KV and the 13.8 KV sources of offsite power are available and the remaining diesel generators are tested daily to ensure operability and the engineered safety features associated with these diesel generator buses are operable.

2. The 138 KV or the 138 KV sources of power may be inoperable for 48 hours provided the three diesel generators are operable. This operation may be extended beyond 48 hours provided the failure is reported to the NRC within the 48 hour period with an outline of the plans for restoration of offsite power and NRC approval is granted.

3. One battery may be inoperable for 2 hours provided the other batteries and the three battery chargers remain operable with one battery charger carrying the D. C. load of the failed battery supply system.

C. If the electrical distribution system is not restored to meet the requirements of 3.7.A within the time periods specified in 3.7.B, then:

1. If the reactor is critical, it shall be in the hot shutdown condition within six hours and in the cold shutdown condition within the following 30 hours.

2. If the reactor is subcritical, the reactor coolant system temperature and pressure shall not be increased more than 25°F and 100 psi, respectively, over existing values.

D. The requirements of Specification 3.7.A.1 may be modified during an emergency system-wide blackout condition as follows:

Two of the three 13.8 KV feeders (13W92, 13W93 and/or 13W94) to the Buchanan Substation 138 KV buses operable with at least 37 MV power from any combination of gas turbines (nameplate rating at 80°F) at the Buchanan Substation and onsite and onsite available for exclusive use on Indian Point Unit No. 3.

E. Whenever the reactor critical, the circuit breaker on the electrical feeder to emergency lighting panel 318 inside containment shall be locked open except when containment access is required.

F. As a minimum, under all conditions including cold shutdown, the following A.C. electrical power sources shall be operable:

1. One transmission circuit to Buchanan Substation, except for testing.

2. Either:

   a. 6.9 KV buses 5 or 6 energized from the 138 KV feeder 95331 or 95332,

      or

   b. 13.8 KV feeder 13W92 or 13W93 and its associated 13.8/6.9 KV transformer available to supply 6.9 power,

3. Two of the four 480-volt buses 2A, 3A, 5A and 6A energized.

4. Two operable diesel generators together with total underground storage containing a minimum of 5676 gallons of fuel.

Basis

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt equipment is arranged on 4 buses. The 6900-volt equipment is supplied from 6 buses.

The Buchanan Substation has both 345 KV and 138 KV transmission circuits which are capable of supplying startup, normal operation, shutdown and/or engineered safeguards loads.

The 138 KV supplies or the gas turbines are capable of providing sufficient power for plant startup. Power via the station auxiliary transformer can supply all the required plant auxiliaries during normal operation, if required.

In addition to the unit transformer, four separate sources supply station service power to the plant.[1]

7. The containment vent and purge system, including the radiation monitors which initiate isolation, shall be tested and verified to be operable within 100 hours prior to refueling operations.

8. No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 120 hours. In addition, movement of fuel in the reactor before the reactor has been subcritical for equal to or greater than 365 hours will necessitate operation of the Containment Building Vent and Purge System through the HEPA filters and charcoal adsorbers. For this case operability of the Containment Building Vent and Purge System shall be established in accordance with Section 4.13 of the Technical Specifications. In the event that more than one region of fuel (72 assemblies) is to be dischargeed from the reactor, those assemblies in excess of one region shall not be discharged before an interval of 400 hours has elapsed after shutdown.

9. Whenever movement of irradiated fuel is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of the reactor pressure vessel flange.

10. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than maximum load to be assumed by the hoists or cranes during the refueling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test and prior to fuel handling. A test of interlocks shall also be performed.

11. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the period of inoperability.

B. If any of the specified limiting condition for refueling are not met, refueling shall cease until the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

C. During fuel handling and storage operations, the following conditions shall be met.

1. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therin. If the monitor is inoperable, a portable monitor may be used.

2. The spent fuel cask shall not be moved over any region of the spent fuel pit which contains irradiated fuel. Additionally, if the spent fuel pit contains irradiated fuel, no loads in excess of 2,000 pounds shall be moved over any region of the spent fuel pit.

3. During periods of spent fuel cask or fuel storage building cask crane movement over the spent fuel pit, or during periods of spent fuel movement in the spent fuel pit, when the pit contains irradiated fuel, the pit shall be filled with borated water at a concentration of $>$1000 ppm.

4. Whenever movement of irradiated fuel in the spent fuel pit is being made, the minimum water level in the area of movement shall be maintained 23 feet over the top of irradiated fuel assemblies seated in the storage rack.

5. Hoists or cranes utilized in handling irradiated fuel shall be dead-load tested before fuel movement begins. The load assumed by the hoists or cranes for this test must be equal to or greater than the maximum load to be assumed by the hoists or cranes during the fuel handling operation. A thorough visual inspection of the hoists or cranes shall be made after the dead-load test prior to fuel handling. Inspection of fuel handling equipment is to include testing of overload cutoff devices on the manipulator.

5. The fuel storage building emergency ventilation system shall be operable whenever irradiated fuel is being handled within the fuel storage building. The emergency ventilation system may be inoperable when irradiated fuel is in the fuel storage building, provided irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are moved over the spent fuel pit during the periods of inoperability.

Basis

The equipment and general procedures to be utilized during refueling, fuel handling, and storage are discussed in the FSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling, fuel handling, reactor maintenance or storage operations that would result in a hazard to public health and safety.(1) Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat removal pump is used to maintain a uniform boron concentration.

The shutdown margin indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling the reactor refueling cavity is filled with approximately 342,000 gallons of water from the refueling water storage tank with a boron concentration of 2000 ppm. A shutdown margin of 10% $\Delta K/K$ in the cold condition with all rods inserted will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration and residual heat removal pump operation insure the proper shutdown margin. The requirement for direct communications allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accomodate only one fuel assembly at a time.

The 120-hour decay time following the subcritical condition and the 23 feet of water above the top of the reactor pressure vessel flange is consistent with the assumptions used in the dose calculation for the fuel-handling accident.

The waiting time of 400 hours required following plant shutdown before unloading more than one region of fuel from the reactor assures that the maximum pool water temperature will be within design objectives as stated in the FSAR.

3.10.2.8 Alarms are provided to indicate non-conformance with the flux difference requirements of 3.10.2.5.1 and the flux difference-time requirements of 3.10.2.6.1. If the alarms are temporarily out of service, conformance with the applicable limit shall be demonstrated by logging the flux difference at hourly intervals for the first 24 hours and half-hourly thereafter.

3.10.2.9 If the core is operating above 75% power with one excore nuclear channel out of service, then core quadrant power balance shall be determined once a day using movable incore detectors (at least two thimbles per quadrant).

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 When ever the indicated quadrant power tilt ratio exceeds 1.02, except for physics tests, within two hours the tilt condition shall be eliminated or the following actions shall be taken:

a) Restrict core power level and reset the power range high flux setpoint three percent of rated value for every percent of indicated power tilt ratio exceeding 1.0, and

b) If the tilt condition is not eliminated after 24 hours, the power range nuclear instrumentation setpoint shall be rest to 55% of allowed power. Subsequent reactor operation is permitted up to 50% for the purpose of measurement, testing and corrective action.

3.10.3.2 Except for physics tests, if the indicated quadrant power tilt ration exceeds 1.09 and there is simultaneous indication of a misaligned control rod, restrict core power level 3% of rated value for every percent of indicated power tilt ratio exceeding 1.0 and realign the rod within two hours. If the rod is not realigned within two hours or if there is no simultaneous indication of a mis-alighned rod, the reactor shall be brought to the hot shutdown condition within 4 hours If the reactor is shut down, subsequent testing up to 50% of rated power shall be permitted to determine the cause of the tilt.

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3.10.6.3 If a control rod having a rod position indicator channel out of service, is found to be misaligned from 3.10.6.1a. above, then Specification 3.10.5 will be applied.

3.10.7 Inoperable Rod Limitations

3.10.7.1 An inoperable rod is a rod which does not trip or which is declared inoperable under Specification 3.10.5 or fails to meet the requirements of 3.10.8.

3.10.7.2 Not more than one inoperable control rod shall be allowed any time the reactor is critical except during physics tests requiring intentional rod misalignment. Otherwise, the plant shall be brought to the hot shutdown condition.

3.10.7.3 If any rod has been declared inoperable, then the potential ejected rod worth, associated transient power distribution peaking factors and the accidents listed in Table 3.10-1 shall be analyzed within 5 days, or the reactor brought to the hot shutdown condition using normal operating procedures. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the plant power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis.

3.10.8 Rod Drop Time

At operating temperature and full flow, the drop time to each control rod shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry.

described below.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses. It is not intended that reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This asymmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level.

The two hour time interval in this specification is considered ample to identify a dropped or misaligned rod and complete realignment procedures to eliminate the tilt. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the casue of the tilt condition. The measurements would include a full core physics map utilizing the moveable detector system. For a tilt condition $\leq$ 1.09, an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of three percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two to one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment.

In the event a tilt condition of $\leq$ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for low power physics testing. To avoid reset of a large number of portection setpoints, the power range nuclear instrumentation would be reset to cause an automatic reactor trip at 55% of allowed power. A reactor trip at this power has been selected to prevent, with margin, exceeding core safety limits even with a nine percent tilt condition.

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a hot shutdown condition for investigation. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (3% for each one percent the tilt ratio exceeds 1.0) for two hours to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. One percent shutdown is adequate except for steam break analysis, which requires more shutdown if the boron concentration is low. Figure 3.10-1 is drawn accordingly.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequence of a hypothetical rod ejection accident. The available control rod reactivity, or excess beyond needs, decreases with decreasing boron concentration because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.4) is to measure the worth of all rods less the worth of the worst case for an assumed stuck rod, that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest such as end of life cooldown, or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worth. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during these tests.

The rod position indicator channel is sufficiently accurate to detect a rod ±7 inches away from its demand position. An indicated misalignment less than 12 steps does not exceed the power peaking factor limits. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable incore detectors, will be used to verify power distribution symmetry. These indirect measurements do not have the same resolution if the bank is near either end of the core, because a 12 step misalignment would have no effect on power distribution. Therefore, it is necessary to apply the indirect checks following significant rod motion.

One inoperable control rod is acceptable provided that the power distribution limits are met, trip shutdown capability is available, and provided the potential hypothetical ejection of the inoperable rod is not worse than the cases analyzed in the safety analysis report. The rod ejection accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 5 day period is short compared with the time interval required to achieve a significant, non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

<u>REFERENCE</u>

1. WCAP-8576, "Augmented Startup and Cycle 1 Physics Program:, August 1975

2. FSAR Appendix 14C

TABLE 3.10-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss Of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation system, for the recalibration of the excore axial off-set detection system.

Specification

A. A minimum of 2 thimbles per quadrant and sufficient movable in-core detectors shall be operable during re-calibration of the excore axial off-set detection system.

B. Power shall be limited to 90% of rated power for 4 loop or 65% of rated power for 3 loop operation if re-calibration requirements for excore axial off-set detection system, identified in Table 4.1-1. are not met.

C. During the incore/excore calibration procedure, all full core flux maps will be made only when 75% of the movable detector guide thimbles are operable.

Basis

The Movable In-core Instrumentation System (1) has six drives, six detectors, and 50 thimbles in the core. Each detector can be routed to sixteen or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the ex-core detectors.

To calibrate the excore detectors system, it is only necessary that the Movable In-core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

3.12 RIVER LEVEL

Applicability

Applies to water elevation of the Hudson River as measured at the Indian Point Unit No. 3 intake structure.

Objective

To specify the maximum water elevation of the Hudson River for safe operation of the reactor.

Specification

When the Hudson River water elevation as measured at the Indian Point Unit Unit No. 3 intake structure reaches 11'-0" above mean sea level, sandbagging the service water pumps will be initiated. If the Hudson River water elevation reaches 12'-5" above mean sea level at the Indian Point Unit No. 3 intake structure, the reactor will be in the hot shutdown condition within six hours and in the cold shutdown condition within the following 30 hours.

Basis

Analyses have been performed wich indicate that the river water elevation would have to reach 15'-3" above mean sea level before it would seep into the lowest floor elevation of any of the buildings hoursing equipment vital for safe shutdown of the reactor.[1] Monitoring of the Hudson River water elevation will not be required until there is a flood warning notice dis-eminated by the New York City National Oceanographic and Atmosphere Administration (NOAA) office.

References:

(1) FSAR, Section 2.5

b. If repairs are not completed and conformance to the acceptance criterion is not demonstrated within 7 days, the reactor shall be shut down until repairs are effected and the continuous leakage meets the acceptance criterion.

C. Sensitive Leakage Rate

1. Test

A sensitive leakage rate test shall be conducted with the containment penetrations, weld channels, and certain double gasketed seals and isolation valve interspaces at a minimum pressure of 41 psig and with the containment building at atmospheric pressure.

2. Acceptance Criteria

The test shall be considered satisfactory if the leak rate for the containment penetrations, weld channel and other pressurized zones is equal to or less than 0.2% of the containment free volume per day.

3. Frequency

A sensitive leakage rate test shall be performed at a frequency of at least every other refueling but in no case at intervals greater than 3 years.

D. Air Lock Tests

1. The containment air locks shall be tested at a minimum pressure of 40.6 psig and at a frequency of every 6-months. The acceptance criteria is included in E.2a. The equipment hatch is to be leak rate tested after every reinsertion prior to requiring containment integrity.

2. Whenever containment integrity is required, verification shall be made of proper repressurization to at least 41 psig of the double-gasket air lock door seal upon closing an air lock door.

5. There are 53 control rods in the reactor core. The control rods contain 142 inch lengths of silver-indium-cadmium alloy clad with the stainless steel.[5]

B. Reactor Coolant System

1. The design of the reactor coolant system complies with the code requirements.[6] Design values for system temperature and pressure are 571.5^{o}F and 2250 psig respectively.

2. All piping, components and supporting structures of the reactor coolant system are designed to Class I requirements, and have been designed to withstand the maximum potential seismic ground acceleration, 0.15g, acting in the horizontal and 0.10g acting in the vertical planes simultaneously with no loss of function.

3. The total liquid volume of the reactor coolant system, at rated operating conditions, is 11,522 cubic feet.

References

(1) FSAR Section 3.2.2

(2) FSAR Section 3.2.1

(3) FSAR Section 3.2.1

(4) FSAR Section 3.2.3

(5) FSAR Sections 3.2.1 & 3.2.3

(6) FSAR Table 4.1-10

5.4 FUEL STORAGE

Applica:ility

Applies to the capacity and storage arrays of new and spent fuel.

Objective

To define those aspects of fuel storage relating to prevention of criticality in fuel storage areas.

Specification

1. The spent fuel pit structure is designed to withstand the anticipated earthquake loadings as a Class I structure. The spent fuel pit has a stanless steel liner to insure against loss of water.

2. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than an array of vertical fuel assemblies with the sufficient center-to-center distance between assemblies to assure $k_{eff} \leq 0.95$. The nominal center-to-center spacing of the storage racks is 1 foot. The capacity of the spent fuel pit is 840 assemblies and the new fuel area is 72 assemblies.

3. Whenever there is fuel in the pit (except in the initial core loading), the spent fuel storage pit is filled and borated to the concentration to match that used in the reactor cavity and refueling canal during refueling operations.

4. Fuel assemblies that contain more than 44.46 grams of uranium -235, or equivalent, per axial centimeter of fuel assembly shall not be stored in the spent fuel pit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 34 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

Introduction

By letter of July 7, 1979, we identified sections of the Indian Point 3 Technical Specifications that needed upgrading to be at least as stringent as the Standard Technical Specifications for Westinghouse plants. We undertook this effort in accordance with Item F.1(f)(3) of the Zion/Indian Point Task Action Plan (Enclosure 4 of our April 5, 1980 letter). By letter dated September 29, 1980, the Power Authority of the State of New York (the licensee) proposed Technical Specifications to satisfy parts of our July 7, 1979 letter.

By letter dated October 31, 1980, the licensee proposed a change to the Technical Specification definition of "surveillance interval." Also, in a second letter dated October 31, 1980 the licensee proposed a change to the Technical Specifications to assure a minimum of 23 feet of water over the top of the reactor pressure vessel flange during movement of fuel assemblies. This proposal was in response to our letter of August 15, 1980 to "All Westinghouse Pressurized Water Reactor Licensees."

By letter dated November 7, 1980, the licensee proposed a change to the Technical Specifications to require valves 1870 and 743 be open and the power supplies de-energized. This proposal is in response to our letter of September 22, 1980.

Finally, the license condition which deals with steam generator inspection is being revised to eliminate the requirement for a mid-cycle inspection of one steam generator. This inspection was conducted in October 1980 and approved by our letter dated November 20, 1980.

Evaluation

The proposed Technical Specifications submitted by letter of July 7, 1980 are, in all cases, more restrictive than the current Technical Specifications. Our objective to upgrade the Indian Point 3 Technical Specifications, as discussed in Item F.1(f)(3) of the Zion/Indian Point Task Action Plan, has been met in these proposed Technical Specifications and we therefore find them acceptable.

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However, the licensee still has to propose changes to the following Sections of the Technical Specifications to completely satisfy our July 7, 1979 letter: 3.1.A.1.b, 3.1.A.1.c, 3.5.3.6.c, 4.5.A.4, 4.5.A.5, 4.5.A.6, 4.6.B, 4.6.X, 4.8.1, and appropriate Limiting Conditions for Operation and Surveillance Requirements for remote shutdown instrumentation, post-accident instrumentation, chlorine detection instrumentation, turbine overspeed, and electrical equipment protection devices.

One of the October 31, 1980 letters proposes to change the definition of "surveillance interval" to that which is in the Standard Technical Specifications. We find the change acceptable.

The other letter dated October 31, 1980 was sent in response to our August 15, 1980 letter. The proposed change assures a minimum of 23 feet over the top of the reactor pressure vessel flange during movement of fuel assemblies. This, in turn, assures that the minimum level of water over the fuel assemblies is approximately 33 feet which is sufficient depth to prevent inadvertent exposure of a fuel assembly during transfer. We find this proposed change acceptable.

The November 7, 1980 letter proposes a change to the Technical Specifications which keeps valves 1870 and 743 open and their power supplies de-energized. This is required to prevent a spurious closing of either of these valves following a safety injection signal. These valves are in the Residual Heat Removal (RHR) System miniflow line, and a spurious closing of either valve could result in damage to the RHR pumps if the reactor coolant system pressure is still high.

Finally, the updated license condition for the steam generator inspection requires the inspection of all four steam generators within twelve equivalent months of operation from the mid-cycle inspection. The mid-cycle inspection of #32 Steam Generator was performed after slightly more than eight equivalent months of operation. Thus, with the new license condition, the mid-cycle inspection was performed when at least 40% of the Cycle 3 operation was completed. Therefore, our conclusion in the November 20, 1980 letter that the plant can operate for the duration of Cycle 3 without additional inspection of the steam generators remains valid.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: January 15, 1981

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-286

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 34 to Facility Operating License No. DPR-64, issued to the Power Authority of the State of New York (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 3 (the facility) located in Buchanan, Westchester County, New York. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications in several areas to make them more consistent with the Standard Technical Specifications, revises the Technical Specifications to assure at least 23 feet of water over the top of the reactor pressure vessel flange during movement of fuel assemblies, revises the Technical Specification to require that two valves in the miniflow line for the Residual Heat Removal Pumps be kept open, and revises the license condition dealing with steam generator inspection.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's

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rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated July 7, 1979, November 7, 1980 and two applications dated October 31, 1980, (2) Amendment No. 34 to License No. DPR-64, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the White Plains Public Library, 100 Martine Avenue, White Plains, New York. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 15 day of January 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing