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September 6, 1976

Docket No.: 50-247

Consolidated Edison Company
of New York, Inc.
ATTN: Mr. William J. Cahill, Jr.
Vice President
4 Irving Place
New York, New York 10003

Gentlemen:

The Commission has issued the enclosed Amendment No. 20 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your requests dated July 9, 1975 and February 9, 1976, as amended and supplemented. As discussed with your staff, modifications have been made to your proposed changes to meet regulatory requirements.

This amendment revises the Technical Specifications to establish operating limits for Indian Point Unit No. 2 as reloaded for cycle 2 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR 50.46, and terminates the operating restrictions imposed by the Commission's December 27, 1974 Order for Modification of License.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 20
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Federal Register Notice

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

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. DPR-26

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

   A. The applications for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated July 9, 1975 and February 9, 1976, as amended and supplemented, 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.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 4, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
i and ii	i and ii
iv and v	iv and v
2.1-1 thru Figure 2.1-1	2.1-1 thru Figure 2.1-1
3.1-1 thru 3.1-4	3.1-1 thru 3.1-4
3.3-1, 3.3-2, 3.3-11 thru 3.3-14	3.3-1, 3.3-2, 3.3-11 thru 3.3-14
3.7-3 thru 3.7-5	3.7-3 thru 3.7-5
3.10-1 thru 3.10-7	3.10-1 thru 3.10-16
Figures 3.10-1 thru 3.10-4	Figures 3.10-1 thru 3.10-6
4.5-1 and 4.5-2	4.5-1 and 4.5-2
4.5-7	4.5-7

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2

Change No. 2

TAB_ OF CONTENTS (Continued)

LIST OF TABLES

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LIST OF FIGURES

2 SAFETY LIMTTS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperaure during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three-loop operation respectively (additional limitations on three-loop operation are described in section 3.1.A). The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

The Region 1 fuel residence time shall be limited to 21,000 effective full power hours (EFPH) under design operation conditions. The licensee may propose to operate individual assemblies from Region 1 in excess of 21,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operation conditions. Any such analysis, if proposed, shall be approved by the Regulatory Staff prior to operation in excess of 21.000 EFPH.

The following DNB related parameters pertain to four loop steady state operation at power levels greater than 98% of rated full power (in excess of 2703 MWt):

a. Reactor Coolant System $T_{avg} \leq 573.5^{o}F$
b. Pressurizer Pressure $\geq$ 2220 psia
c. Reactor Coolant System Total Flow Rate $\geq$ 358,800 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by operating the hot region of the core within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters: thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 DNB correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core locatiom to the local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.[1]

The curves of Figure 2.1-1 and 2.1-2 represent the loci of points of thermal power, coolant system pressure and average temperature for which the DNBR is no less than 1.30. The area where clad integrity is assured is below these lines.

The curves are based on the following nuclear hot channel factors:[2]

$$F_q^N = 3.12$$

$$F_{\Delta H}^N = 1.75$$

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion.[3] The control rod insertion limits are covered by Specification 3.10. Higher hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-3 insure that the DNBR is always greater at partial power than at full power. For three loop operation the insertion limits fo Figure 3.10-4 apply.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part-length rods that is allowed before the reactor trip set points are reduced and rod withdrawal block and load runback may be required.(2) Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than 1.30.(4)

The ranges on reactor coolant system temperature, pressure and loop coolant flow during steady-state, four-loop, power operation are specified to assure that the values assumed in the accident analyses are not exceeded during normal plant operation.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within it's range after each refueling cycle.

References

[1]FSAR Section 3.2.2

[2]FSAR Section 3.2.1

[3]FSAR Technical Specification 3.10

[4]FSAR Section 14.1.1

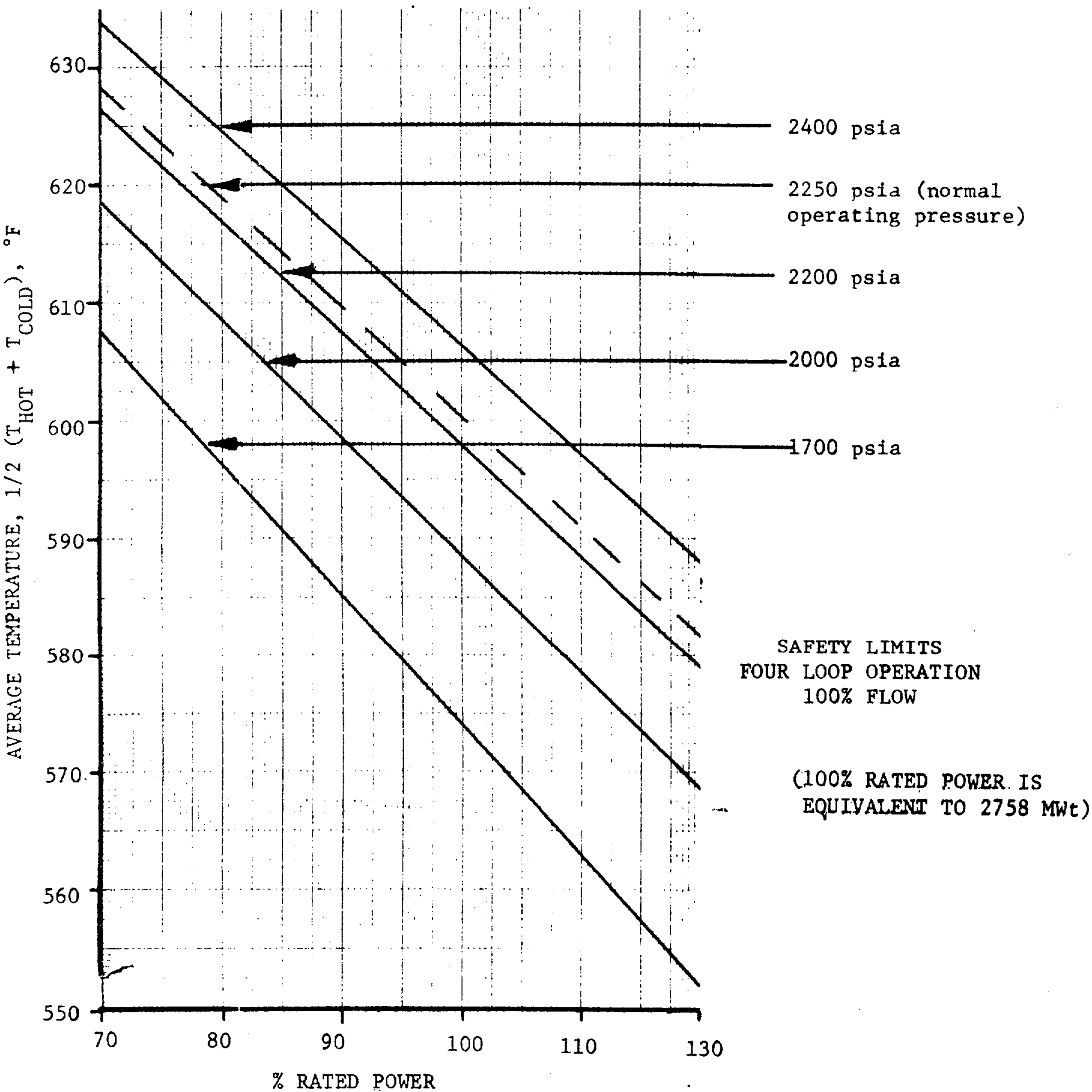

2400 psia
2250 psia (normal
operating pressure)
2200 psia
2000 psia
1700 psia
SAFETY LIMITS
FOUR LOOP OPERATION
100% FLOW
(100% RATED POWER IS
EQUIVALENT TO 2758 MWt)
AVERAGE TEMPERATURE, 1/2 (T_{HOT} + T_{COLD}), °F
630
620
610
600
590
580
570
560
550
70
80
90
100
110
130
% RATED POWER

FIGURE 2.1-1

3 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

   a. At least one reactor coolant pump or one residual heat removal pump in the Residual Heat Removal System when connected to the Reactor Coolant System shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is critical and above 2% rated power, except for natural circulation tests, at least two reactor coolant pumps shall be in operation.

   c. Reactor power shall not be increased above 60% of rated power with only three pumps in operation unless the overtemperature

ΔT trip setpoint for three loop operation has been set in accordance with specification 2.3.1.B-4.

d. Reactor operation with one of the four loops out of service will be permitted for up to 24 hours. If the fourth loop can not be returned to service within 24 hours, the reactor will be put in a hot shutdown condition using normal procedures.

2. Steam Generator

Two steam generators shall be capable of performing their heat transfer function whenever the reactor is critical and the average coolant temperature is above 350°F.

3. Safety Valves

a. At least one pressurizer code safety valve shall be operable whenever the reactor head is on the vessel except for hydrostatically testing the RCS in accordance with the ASME Section XI Boiler and Pressure Vessel Code.

b. All pressurizer code safety valves shall be operable whenever the reactor is critical.

c. The pressurizer code safety valve lift setting shall be set at 2485 psig with ± 1% allowance for error.

Basis

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one half hour. The pressurizer is of no concern because of the low pressurizer volume and because the pressurizer boron concentration will be higher than that of the rest of the reactor coolant.

Heat transfer analyses show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only[1]; hence, the specified upper limit of 2% rated power without operating pumps provides a substantial safety factor.

Three loop operation is allowed over a 24 hour period to permit corrective action to return the fourth loop to service and limit the number of unnecessary shutdown cycles. During these periods of three loop operation, the reactor coolant system parameters will be maintained within the limits described for three loop operation in Section 2.1 and 3.1 of the Technical Specifications.

Each of the pressurizer code safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. Below approximately 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure.[2]

If no residual heat were removed by the Residual Heat Removal System the amount of steam which could be generated at safety valve relief pressure would be less than half the capacity of a single valve. One valve therefore provides adequate protection for over-pressurization.

The combined capacity of the three pressurizer safety valves is greater than the maximum surge rate resulting from complete loss of load[3] without a direct reactor trip or any other control.

Two steam generators capable of performing their heat transfer function will provide sufficient heat removal capability to remove core decay heat after a reactor shutdown.

<u>Reference</u>

1) FSAR Section 14.1.6
2) FSAR Section 9.3.1
3) FSAR Section 14.1.10

B. HEATUP AND COOLDOWN

1. For the first two years of power operation (1.61×10^6 thermal megawatt days) the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

Heatup:

a. For indicated temperatures at or below 220°F the maximum indicated pressure shall not exceed 500 psig and the maximum heatup rate shall not exceed 50°F/hr, as shown by the dotted line on Figure 3.1-1.

b. For indicated temperatures above 220°F the heatup rate shall not exceed 100°F/hr.

Cooldown:

a. Allowable combinations of pressure and temperature for a specific cooldown rate for indicated temperature at or below 136°F are below and to the right of the solid limit lines for that rate as shown on Figure 3.1-2. Furthermore, the maximum indicated pressure shall not exceed 500 psig for indicated temperatures at or below 220°F as shown by the dotted limit line on Figure 3.1-2. The maximum cooldown rate shall not exceed 50°F/hr for indicated temperature at or below 220°F. The limit lines for cooling rates between those shown by the solid lines on Figure 3.1-2 may be obtained by interpolation.

b. For indicated temperatures above 220°F the rate shall not exceed 100°F/hr.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

3.3 ENGINEERED SAFETY FEATURES

Applicability

Applies to the operating status of the Engineered Safety Features.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, (3) to remove airborne iodine from the containment atmosphere following a Design Basis Accident, (4) to minimize containment leakage to the environment subsequent to a Design Basis Accident.

Specification

The following specifications apply except during low temperature physics tests.

A. Safety Injection and Residual Heat Removal Systems

1. The reactor shall not be made critical, except for low temperature physics tests, unless the following conditions are met:

   a. The refueling water tank contains not less than 350,000 gal. of water with a boron concentration of at least 2000 ppm.

   b. The boron injection tank contains not less than 1000 gal. of a 11 1/2% to 13% by weight (20,000 ppm to 22,500 ppm of boron) boric acid solution at a temperature of at least 145°F. Two channels of heat tracing, shall be available for the flow path. Valves 1821 and 1831 shall be open and valves 1822A and 1822B shall be closed, except during short period of time when they can be cycled to demonstrate their operability.

c. The four accumulators are pressurized to at least 600 psig and each contains a minimum of 734 ft^3 and a maximum of 749 ft^3 of water with a boron concentration of at least 2000 ppm. None of these four accumulators may be isolated.

d. Three safety injection pumps together with their associated piping and valves are operable.

e. Two residual heat removal pumps and heat exchangers together with their associated piping and valves are operable.

f. Two recirculation pumps together with the associated piping and valves are operable.

g. Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the RWST are de-energized in the open position.

h. Valves 856A, C, D and E, in the discharge header of the safety injection header are in the open position. Valves 856B and F, in the discharge header of the safety injection header are in the closed position. The hot leg valves (856B and F) shall be closed with their motor operators de-energized by locking out the circuit breakers at the Motor Control Centers.

i. The four accumulator isolation valves shall be open with their motor operators deenergized by locking out the circuit breakers at the Motor Control Centers.

j. Valve 1810 on the suction line of the high-head SI pumps and valves 882 and 744, respectively on the suction and discharge line of the residual heat removal pumps, shall be blocked open by de-energizing the valve-motor operators.

2. During power operation, the requirements of 3.3.A-1 may be modified to allow any one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 3.3.A-1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.A-1 are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

injection phase. The accumulator isolation valve motor operators are de-energized to prevent an extremely unlikely spurious closure of these valves from occuring when accumulator core cooling flow is required.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.(3) The measure of effectiveness of the Safety Injection System is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly where the core has been uncovered for postulated large area ruptures. The result of the performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met.(10) The range of core protection as a function of break diameter provided by the various components of the Safety Injection System is presented in Figure 6.2-6 of the FSAR.

The containment cooling and iodine removal functions are provided by two independent systems: (a) fan-coolers plus charcoal filters and (b) containment spray with sodium hydroxide addition. During normal power operation, the five fan-coolers are required to remove heat lost from equipment and piping within containment at design conditions (with a cooling water temperature of 85°F).(4) In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at a rate consistant with limiting off-site doses to acceptable values: (1) five fan-cooler units, (2) two containment spray pumps, (3) three fan-cooler units and one spray pump. Also in the event of a Design Basis Accident, three charcoal filters (and their associated recirculation fans) in operation, along with one containment spray pump and sodium hydroxide addition, will reduce airborne organic and molecular iodine activities sufficiently to limit off-site doses to acceptable values. These constitute the minimum safeguards for iodine removal, and are capable of being operated on emergency power with one diesel generator inoperable.

If off-site power is available or all diesel generators are operating to provide emergency power, the remaining installed iodine removal equipment (two charcoal filters and their associated fans, and one containment spray pump and sodium hydroxide addition) can be operated to provide iodine removal in excess of the minimum requirements. Adequate power for operation of the redundant containment heat removal systems (i.e., five fan-cooler units or two containment spray pumps) is assured by the availability of off-site power or operation of all emergency diesel generators.

One of the five fan cooler units is permitted to be inoperable during power operation. This is an abnormal operating situation, in that the normal plant operating procedures require that an inoperable fan-cooler be repaired as soon as practical.

However, because of the difficulty of access to make repairs, it is important on occasion to be able to operate temporarily without at least one fan-cooler. Compensation for this mode of operation, is provided by the high degree of redundancy of containment cooling systems during a Design Basis Accident.

The Component Cooling System is different from the system discussed above in that the pumps are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident.[6] During the recirculation phase following a loss-of-coolant accident, only one of the three component cooling pumps is required for minimum safeguards.[7]

A total of six service water pumps are installed, only two of the set of three service water pumps on the header designated the essential header are required immediately following a postulated loss-of-coolant accident.[8]

During the second phase of the accident, one additional service water pump on the non-essential header will be manually started to supply the minimum cooling water requirements for the component cooling loop.

The limits for the accumulators, and their pressure and volume assure the required amount of water injection following a loss-of-coolant accident. The values used for the accident analyses[9,10] are based on this accumulator water inventory plus the water within the connecting piping between the accumulator and first check valve.

Two full rated recombination systems are provided in order to control the hydrogen evolved in the containment following a loss-of-coolant accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each of the systems is separate from the other and is provided with redundant features. Power supplies for the blowers and ignitors and separate, so that loss of one power supply will not effect the remaining system. Hydrogen gas is used as the externally supplied fuel. Oxygen gas is added to the containment atmosphere through a separate containment feed to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The containment atmosphere sampling system consists of a sample line which originates in each of the containment fan cooler units. The fan and sampling pump head together are sufficient to pump containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment, and then through a second penetration to the sample termination inside the containment. The design hydrogen concentration for operating the recombiner is established at 2% by volume. Conservative calculations indicate that the hydrogen content within the containment will not reach 2% by volume until 13 days after a loss-of-coolant accident. There is therefore no need for immediate operation of the recombiner following an accident, and the quantity of hydrogen fuel stored at the site will be only for periodic testing of the recombiners.

The cable tunnel is equipped with two temperature controlled ventilation fans. Each fan has a capacity of 21,000 cfm and is connected to a 480v bus. One fan will start automatically when the temperature in the tunnel reaches 95°F. The second fan will start if the temperature in the tunnel reaches 100°F. Under the worst conditions, i.e. loss of outside power and all the Engineered Safety Features in operation, one ventilation fan is capable of maintaining : tunnel temperature below 104°. Under the same worst conditions, if no ventilation fans were operating, the natural air circulation through the tunnel would be sufficient to limit the gross tunnel temperature below a tolerable value of 140°F. However, in order to provide for ample tunnel ventilation capacity, the two ventilation fans are required to be operable when the reactor is made critical. If one ventilation fan is found inoperable, the daily testing of the other fan will ensure that cable tunnel ventilation is available.

Valves 856A, C, D and E are maintained in the open position during plant operation to assure a flow path for high-head safety injection during the injection phase of a loss-of-coolant accident. Valves 856B and F are maintained in the closed position during plant operation to prevent hot leg injection during the injection phase of a loss-of-coolant accident. As an additional assurance of preventing hot leg injection, the valve motor operators are de-energized to prevent spurious opening of these valves. Power will be restored to these valves at an appropriate time in accordance with plant operating procedures after a loss-of-coolant accident in order to establish hot leg recirculation.

Valves 842 and 843 in the mini-flow return line from the discharge of the safety injection pumps to the refueling water storage tank are de-energized in the open position to prevent an extremely unlikely spurious closure which would cause the safety injection pumps to overheat if the reactor coolant system pressure is above the shutoff head of the pumps.

References

(1) FSAR Section 9
(2) FSAR Section 6.2
(3) FSAR Section 6.2
(4) FSAR Section 6.3
(5) FSAR Section 14.3.5
(6) FSAR Section 1.2
(7) FSAR Section 8.2
(8) FSAR Section 9.6.1
(9) FSAR Section 14.3
(10) WCAP 8399 "ECCS Acceptance Criteria Analysis, Indian Point Nuclear Generating Station Unit No. 2" September 1974, Westinghouse Non-Proprietary Class 3

2. a. Establish 138 kv bus sections at Buchanan with at least 37 MW power (nameplate rating) from any combination of gas turbines at Buchanan and on-site.

   b. Two 138 kv lines to Buchanan energized from the gas turbines with breakers to Millwood and Orange and Rockland open.

   c. The 13.8 kv line to Buchanan operable and the 13.8/6.9 kv transformer available to supply 6.9 kv power.

   d. The 6.9 kv buses energized from the 138 kv source.

   e. The four 480-volt buses 2A, 3A, 5A and 6A energized and the bus tie breakers between buses 5A and 2A and between buses 3A and 6A open.

   f. Three diesel generators operable with on-site supply of 19,000 gallons of fuel available in the individual storage tanks and 22,000 gallons of fuel available on-site other than the normal supply tanks.

   g. Both batteries plus two chargers and the d.c. distribution system operable.

D. Whenever the reactor critical, the circuit breaker on the electrical feeder to emergency lighting panel 218 inside containment shall be locked open except when containment access is required.

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 in four buses. The 6900-volt equipment is supplied from six buses.

In addition to the unit transformer, three separate sources supply station service power to the plant.(1)

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from different sources. The charging pumps are supplied from the 480-volt buses Nos 3A, 5A, and 6A. The five containment fans are divided among the 480-volt buses. The two residual heat pumps are on separate 480-volt buses. Valves are supplied from separate motor control centers.

The station auxiliary transformer or the gas turbine is capable of providing sufficient power for plant startup. The station auxiliary transformer can supply the required plant auxiliary power during normal operation.

The bus arrangements specified for operation ensure that power is available to an adequate number of safeguards auxiliaries. With additional switching, more equipment could be out of service without infringing on safety.

Two diesel generators have sufficient capacity to start and run at design load the minimum required engineered safeguards equipment.(1) The minimum diesel fuel oil inventory at all times is maintained to assure the operation of two diesels carrying the load of the minimum required engineered safeguards equipment for at least eighty hours.(2) Additional fuel oil suitable for use in the diesel generators will be stored on site. The minimum storage of 22,000 gallons will assure operation of two diesels for ninety hours at the minimum load for engineered safeguards. Commercial oil supplies and trucking facilities exist to assure deliveries within one day's notice. One battery charger shall be in service on each battery so that the batteries will always be at full charge in anticipation of a loss-of-ac power incident. This insures that adequate d.c. power will be available for starting the emergency generators and other emergency uses.

The plant can be safely shutdown without the use of off-site power since all vital loads (safety systems, instruments, etc.) can be supplied from the emergency diesel generators.

Any two of three diesel generators, the station auxiliary transformer or the separate 13.8 to 6.9 kv transformer are each capable of supplying the minimum safeguards loads and therefore provide separate sources of power immediately available for operation of these loads. Thus, the power supply system meets the single failure criteria required of the safety systems.

Conditions of a system-wide blackout could result in a unit trip. Since normal off-site power supplies as required in Specification 3.7.A are not available for startup, it is desirable to be able to blackstart this unit with on-site power supplies as a first step in restoring the system to an operable status and restoring power to customers for essential service. Specification 3.7.C.1 provides for startup using the on-site gas turbine to supply the 6.9 kv loads and the diesels to supply the 480-volt loads. Tie breakers between the 6.9 kv and 480-volt systems are open so that the diesels would not be jeopardized in the event of any incident and would be able to to continue to supply 480-volt safeguards power. The scheme consists of starting two reactor coolant pumps, one condensate pump, 2 circulating water pumps and necessary auxiliaries to bring the unit up to approximately 10% power. At this point, loads can be assumed by the main generator and power supplied to the system in an orderly and routine manner.

This Specification (3.7.C.2) is identical with normal startup requirements as specified in 3.7.A except that off-site power is supplied exclusively from gas turbines with a minimum total power of 37 MW (nameplate rating) which is sufficient to carry out normal plant startup.

As a result of an investigation of the effect components that might become submerged following a LOCA may have an ECCS, containment isolation and other safety-related functions, a fuse and a locked open circuit breaker were provided on the electrical feeder to emergency lighting panel 218 inside containment. With the circuit breaker in the open position, containment electrical penetration H-70 is de-energized during the accident condition. Personnel access to containment may be required during power operation. Since it is highly improbable that a LOCA would occur during this short period of time, the circuit breaker may be closed during that time to provide emergency lighting inside containment for personnel safety.

References

1) FSAR - Section 8.2.1

2) FSAR - Section 8.2.3

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objectives:

To ensure:

1. Core subcriticality after reactor trip,
2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and
3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.32/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.64) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55 \, [1 + 0.2 \, (1-P)]$

where P is the fraction of full power at which the core is operating. K(Z) is the fraction given in Figure 3.10-2 and Z is the core height location of F_Q.

3.10.2.2 Following initial core loading, subsequent reloading and at regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this comparison,

3.10.2.2.1 The measurement of total peaking factor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

3.10.2.2.2 The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$ shall be increased by four percent to account for measurement error. If either measured hot channel factor exceeds its limit specified under Item 3.10.2.1, the reactor power and high neutron flux trip setpoint shall be reduced so as not to exceed a fraction of rated value equal to the ratio of the F_Q or $F_{\Delta H}^N$ limit to measured value, whichever is less. If subsequent in-core mapping cannot, within a 24 hour period, demonstrate that the hot channel factors are met, the reactor shall be brought to a hot shutdown condition with return to power authorized only for the purpose of physics testing.

3.10.2.3 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter. The target flux difference must be updated each effective full power month by linear interpolation using the most recent measured value and a value of approximately 0 percent at the end of the cycle life.

3.10.2.4 Except during physics tests, during excore calibration procedures and except as modified by Items 3.10.2.5 through 3.10.2.7 below, the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (defines the band on axial flux difference).

3.10.2.5 At a power level greater than 90% of rated power,

3.10.2.5.1 If the indicated axial flux difference deviates from its target band, the flux difference shall be returned to its target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

3.10.2.6 At a power level no greater than 90 percent of rated power,

3.10.2.6.1 The indicated axial flux difference may deviate from its ± 5% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -11 percent and +11 percent at 90% power and increasing by -1 percent and +1 percent for each 2 percent of rated power below 90% power.

3.10.2.6.2 If Item 3.10.2.6.1 is violated then the reactor power shall be reduced immediately to no greater than 50% power and the high neutron flux setpoint reduced to no greater than 55 percent of rated values.

3.10.2.6.3 A power increase to a level greater than 90 percent of rated power is contingent upon the indicated axial flux difference being within its target band.

3.10.2.7 At a power level no greater than 50 percent of rated power,

3.10.2.7.1 The indicated axial flux difference may deviate from its target band.

3.10.2.7.2 A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24 hour period. One half the time the indicated axial flux difference is out of its target band up to 50% of rated power is to be counted as contributing to the one hour cumulative maximum the flux difference may deviate from its target band power level $\leq$ 90% of rated power.

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 Part length rods shall not be permitted in the core except for low power physics tests and for axial offset calibration tests performed below 75% of rated power.

3.10.2.10 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 <u>Quadrant Power Tilt Limits</u>

3.10.3.1 Whenever 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 two percent of rated values 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 reset 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 ratio exceeds 1.09 and there is simultaneous indication of a misaligned control rod, restrict core power level two percent 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 misaligned control 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.

3.10.3.3 The rod position indicators shall be monitored and logged once each shift to verify rod position within each bank assignment.

3.10.3.4 The tilt deviation alarm shall be set to annunciate whenever the excore tilt ratio exceeds 1.02 except as modified in specification 3.10.10.

3.10.4 Rod Insertion Limits

3.10.4.1 The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality (i.e., the reactor is no longer subcritical by an amount equal to or greater than the shutdown margin in Figure 3.10-1).

3.10.4.2 When the reactor is critical, the control banks shall be limited in physical insertion to the insertion limits shown in Figure 3.10-3 or Figure 3.10-4.

3.10.4.3 Control bank insertion shall be further restricted if:

a. The measured control rod worth of all rods, less the worth of the most reactive rod (worst case stuck rod), is less than the reactivity required to provide the design value of available shutdown,
b. A rod is inoperable (Specification 3.10.7).

3.10.4.4 Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-1 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod inserted and part length rods fully withdrawn.

3.10.5 Rod Misalignment Limitations

3.10.5.1 If an indicated full-length or part-length rod cluster control assembly is misaligned from its bank demand position by more than 13 steps, then realign the rod or determine the core peaking factors within 2 hours and apply Specification 3.10.2.

3.10.5.2 If the restrictions of Specification 3.10.3 are determined not to apply and the core peaking factors have not been determined within two hours and the rod remains misaligned, the high reactor flux setpoint shall be reduced to 85% of its rated value.

3.10.5.3 If the misaligned rod cluster control is not realigned within 8 hours, the rod shall be declared inoperable.

3.10.6 Inoperable Rod Position Indicator Channels

3.10.6.1 If a rod position indicator channel is out of service then:

a. For operation between 50 percent and 100 percent of rating, the position of the rod cluster control shall be checked indirectly by core instrumentation (excore detectors and/or movable incore detectors) every shift, or subsequent to rod motion exceeding 24 steps, whichever occurs first.
b. During operation below 50 percent of rating, no special monitoring is required.

3.10.6.2 Not more than one rod position indicator channel per group nor two rod position indicator channels per bank shall be permitted to be inoperable at any time.

3.10.6.3 If a full length or part length 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 full length 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 and associated transient power distribution peaking factors shall be determined by analysis within 30 days. 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 of each full length rod cluster control shall be no greater than 1.8 seconds from loss of stationary gripper coil voltage to dashpot entry.

3.10.9 Rod Position Monitor

If the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.10 Quadrant Power Tilt Monitor

If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs shall be logged once per shift and after a load change greater than 10 percent of rated power.

3.10.11 Notification

Any event requiring plant shutdown or trip setpoint reduction because of Specification 3.10 shall be reported to the Nuclear Regulatory Commission within 30 days.

Basis

Design criteria have been chosen for normal operations, operational transients and those events analyzed in FSAR Section 14.1 which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above conditions, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution the following hot channel factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

An upper bound envelope of 2.32 times the normalized peaking factor axial dependence of Figure 3.10-2 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on 2.32 times the normalized envelope of Figure 3.10-2 indicate a peak clad temperature of 2115°F for the double-ended cold leg guillotine break with $C_D = 1.0$. This corresponds to a 85°F margin to the 2200°F limit. [1]

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

Measurements of the hot channel factors are required as part of startup physics tests, at least each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design basis including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design basis remain inviolate and identify operational anomolies which would, otherwise, affect these basis.

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration maximum instrumentation error.
2. Control Rod banks are sequenced with overlapping banks as described in Technical Specification 3.10.4.
3. The full length and part length control bank insertion limits are not violated.
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met. In Specification 3.10.2, F_Q is arbitrarily limited for $P \leq 0.5$ (except for low power physics tests).

The procedures for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that F_Q upper bound envelope of 2.32 times Figure 3.10-2 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occuring during these operations.

In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target bank when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target bank, however to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 to -14 percent (+11 percent to -11 percent indicated) increasing by ± 1 percent for each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ±5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with operating procedures is assumed as a precondition for condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Quadrant power tilt limits are based on the following considerations. Frequent power tilts are not anticipated during normal operation as this phenomenon is caused by some asymmetric perturbation, e.g. rod misalignment, or inlet temperature mismatch. A dropped or misaligned rod will easily be detected by the Rod Position Indication System or core instrumentation per Specification 3.10.6, and core limits are protected per Specification 3.10.5. A quadrant tilt by some other means would not appear instantaneously, but would build up over several hours and the quadrant tilt limits are met to protect against this situation. They also serve as a backup protection against the dropped or misaligned rod. Operational experience shows that normal power tilts are less than 1.01. Thus, sufficient time is available to recognize the presence of a tilt and correct the cause before a severe tilt could buildup. During startup and power escalation, however, a large tilt could be initiated. Therefore, the Technical Specification has been written so as to prevent escalation above 50 percent power if a large tilt is present. The numerical limits are set to be be commensurate with design and safety limits for DNB protection and linear heat generation rate as 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 cause of the tilt condition. The measurements would include a full core physics map utilizing the moveable detector system. For a tilt condition ≤ 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 two 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 ≤ 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 protection 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 (2% 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 worths. 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 13 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 13 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 4 week 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.

REFERENCE

1. "ECCS Acceptance Criteria Analysis, Major Reactor Coolant System Pipe Rupture, Indian Point Generating Station Unit 2", February 4, 1976.

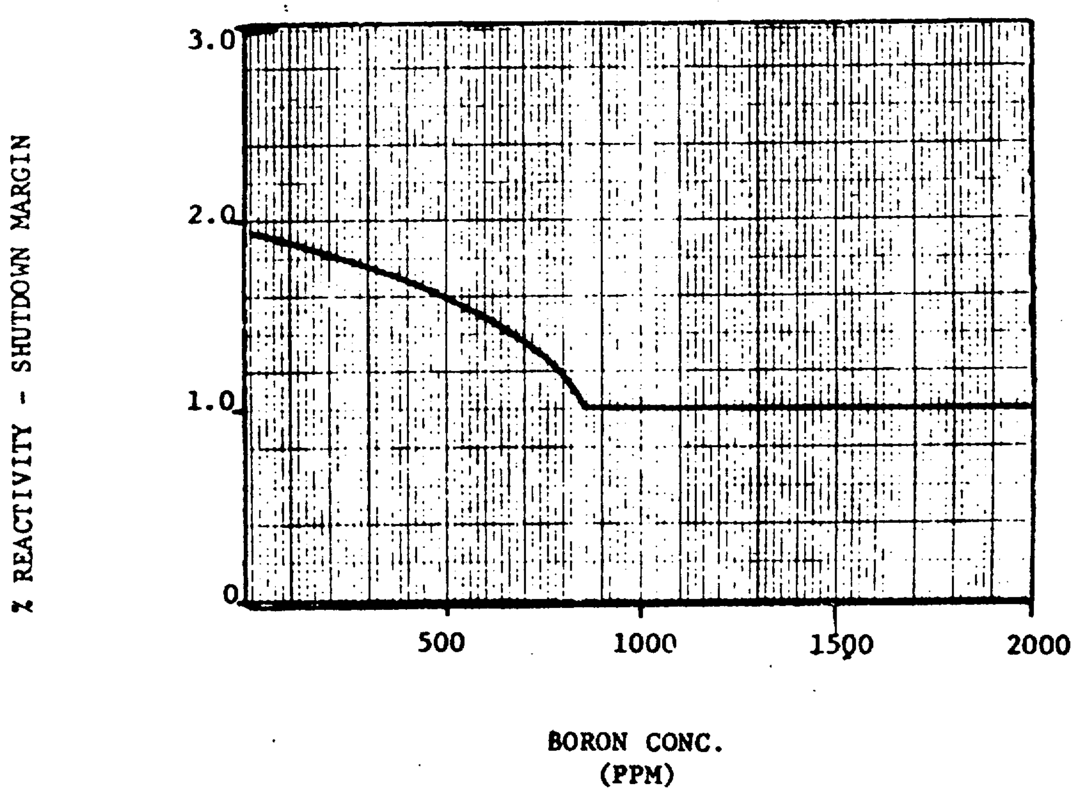


Figure 3.10-1

REQUIRED HOT SHUTDOWN MARGIN
vs
REACTOR COOLANT BORON CONCENTRATION

Figure 3.10-2

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

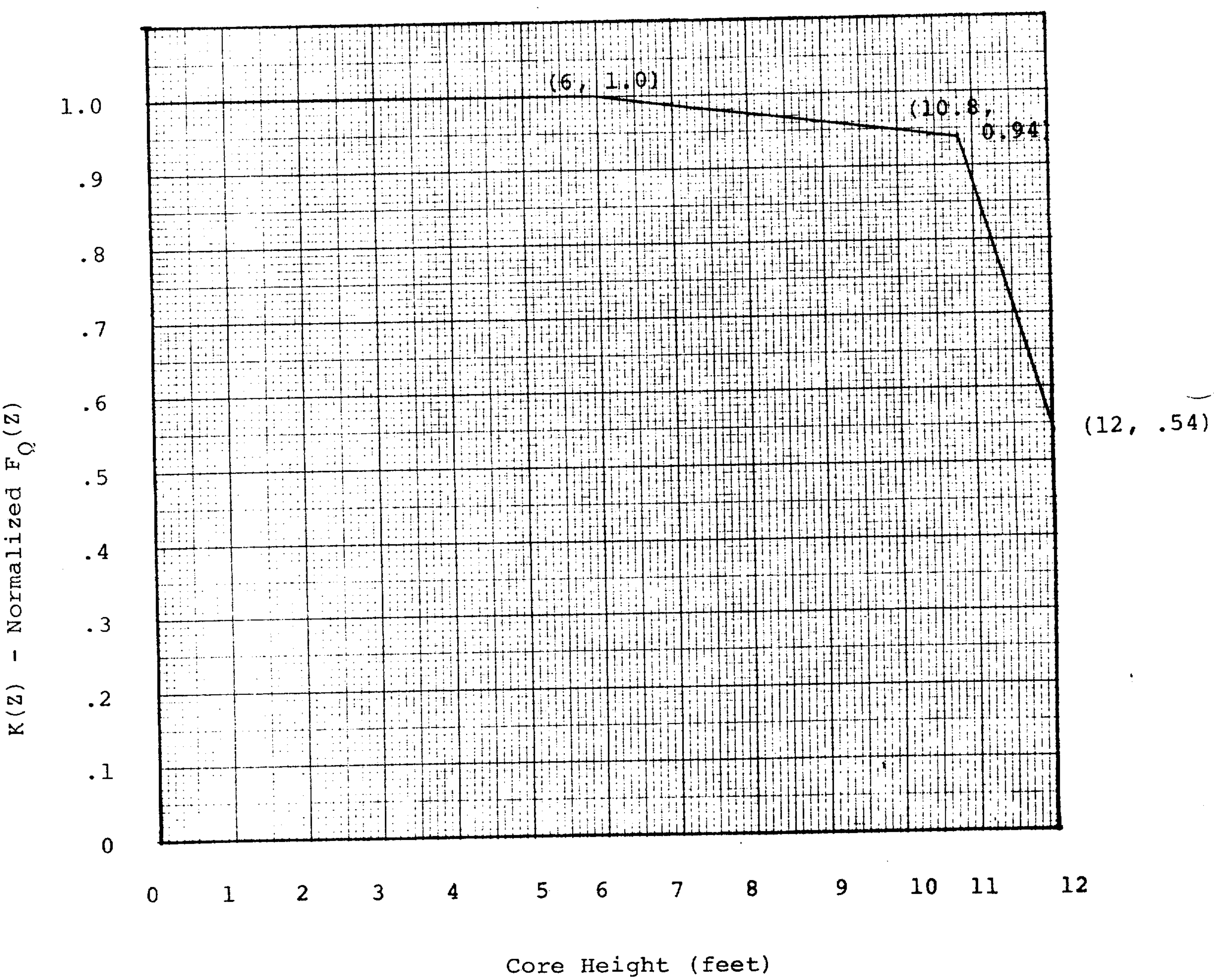
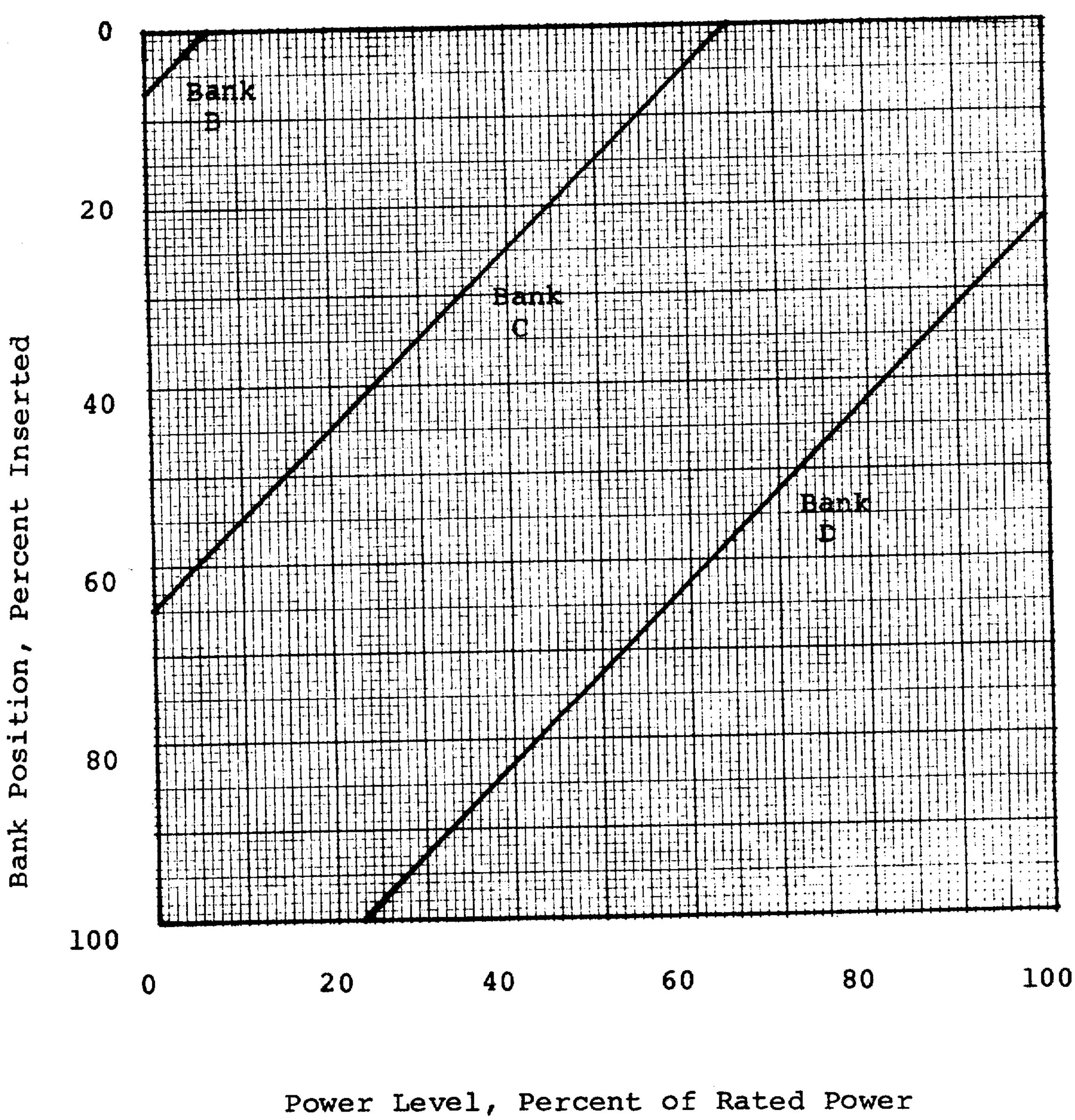


Figure 3.10-3

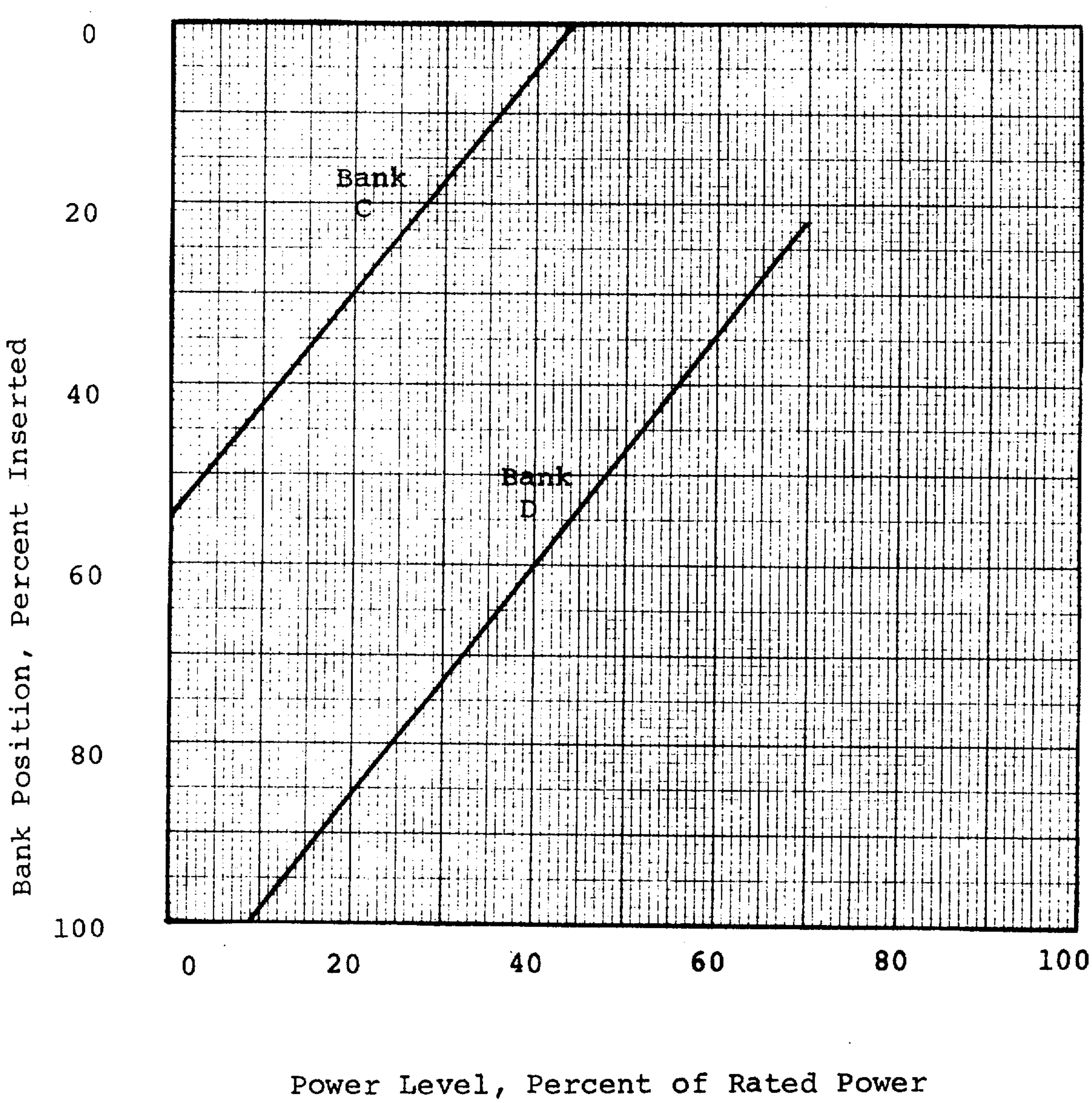
ROD BANK INSERTION LIMITS
(Four Loop Operation)
100 Step Overlay



Amendment No, 20

Figure 3.10-4

ROD BANK INSERTION LIMITS
(Three Loop Operation)
100 Step Overlap



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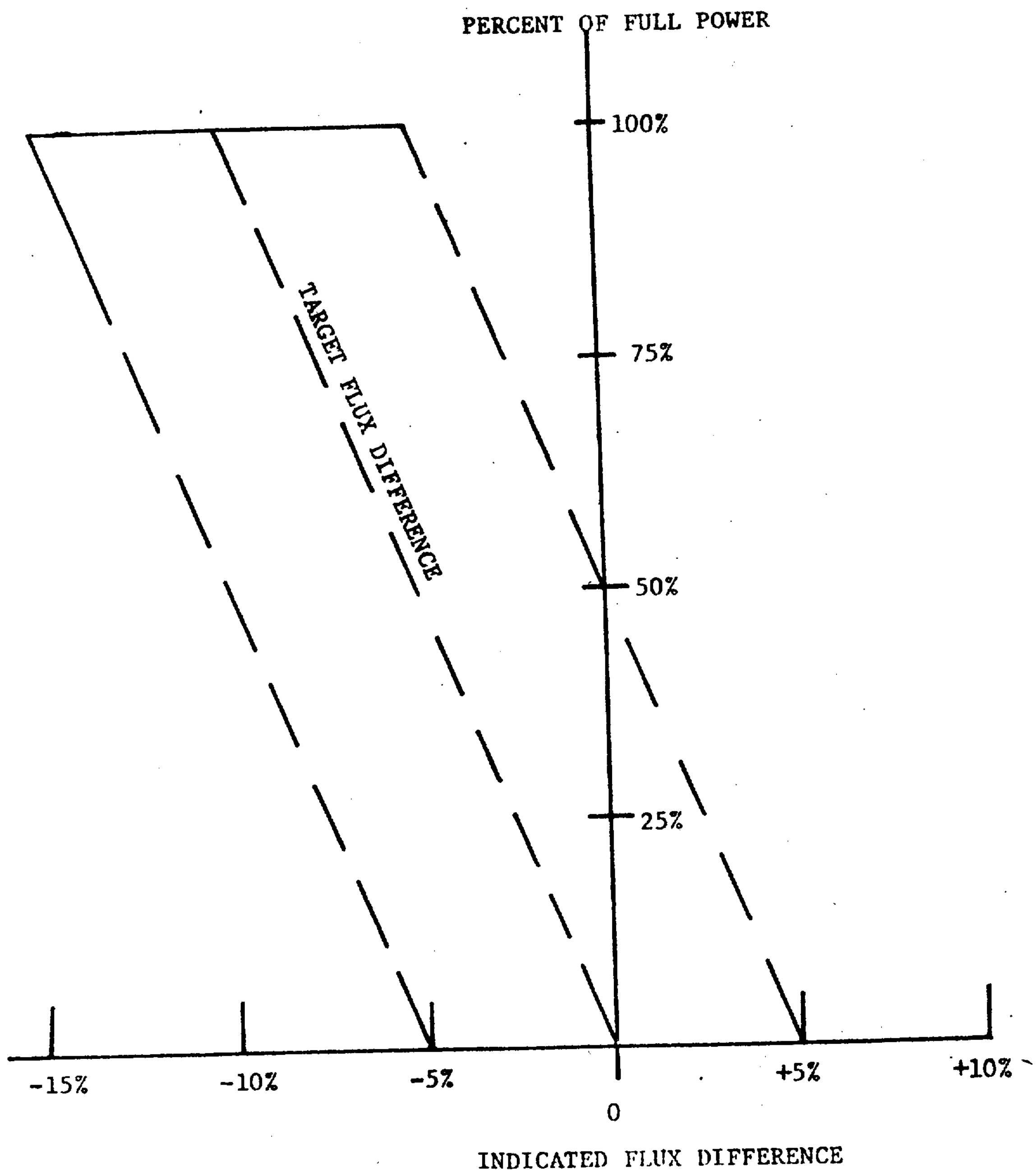

Figure 3.10-5 Target Band on Indicated Flux Difference as a Function of Operating Power Level (Typical for BOL)

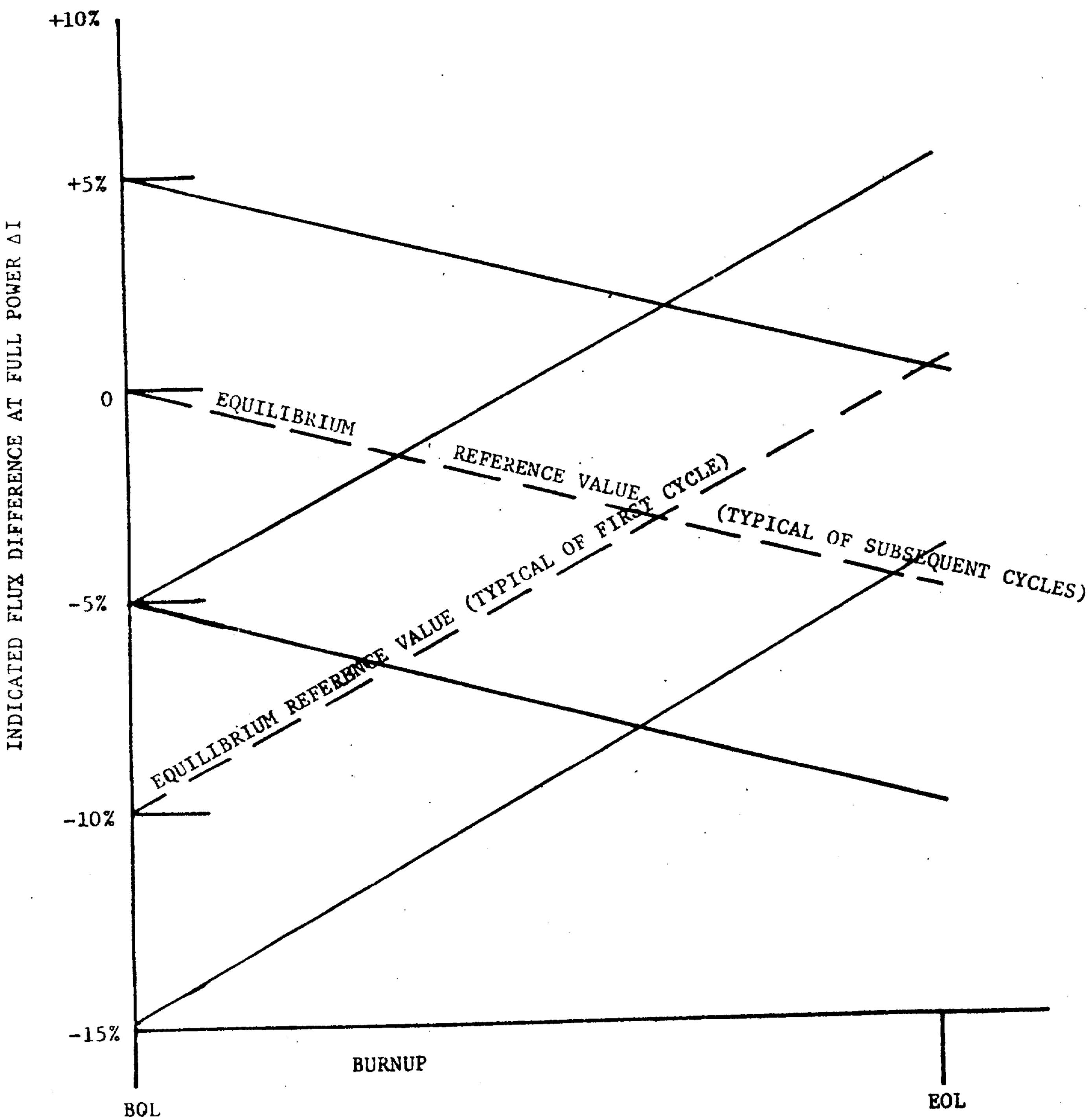

Figure 3.10-6 Permissible Operating Band on Indicated Flux Difference as a Function of Burnup (Typical)

4.5 ENGINEERED SAFETY FEATURES

Applicability

Applies to testing of the Safety Injection System, the Containment Spray System, the Hydrogen Recombiner System, and the Air Filtration System .

Objective

To verify that the subject systems will respond promptly and perform their design functions, if required.

Specification

A. SYSTEM TESTS

1. Safety Injection System

a. System tests shall be performed at each reactor refueling interval. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The safety injection and residual heat removal pumps are made inoperable for this test.

b. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, that is, the appropriate pump breakers shall have opened and closed, and the appropriate valves shall have completed their travel.

c. Conduct a flow test of the high head safety injection system after any modification is made to either its piping and/or valve arrangement.

d. Verify that the mechanical stops on Valves 856 A, C, D & E are set at the position measured and recorded during the most recent ECCS operational flow test or flow tests performed in accordance with (c) above. This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators and at a convenient outage if the position of the mechanical stops have not been verified in the preceding three months.

B. Containment Spray System

1. System tests shall be performed at each reactor refueling interval. The tests shall be performed with the isolation valves in the spray supply lines at the containment and the spray additive tank isolation valves blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation.

2. The spray nozzles shall be checked for proper functioning at least every five years.

3. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

C. Hydrogen Recombiner System

1. A complete recombiner system test shall be performed at each normal reactor refueling on each unit. The test shall include verification of ignition and attainment of normal operating temperature.

2. A complete control system test shall be performed at intervals not greater than six months on each unit. The test shall consist of a complete dry-run startup using artificially generated signals to simulate light off.

3. Containment atmosphere sampling system tests shall be performed at intervals no greater than six months. The test shall include drawing a sample from the fan cooler units and purging the sampling line.

be based upon containment atmosphere sample analysis. The complete functional tests of each unit at refueling shutdown will demotrate the proper operation of the recombiner system. More frequent tests of the recombiner control system and air-supply blowers will assure operability of the system. The biannual testing of the containment atmosphere sampling system will demonstrate the availability of this system.

For the four flow distribution valves (856 A, C, D & E), verification of the valve mechanical stop adjustments is performed periodically to provide assurance that the high head safety injection flow distribution is in accordance with flow values assumed in the core cooling analysis.

References

(1) FSAR Section 6.2

(2) FSAR Section 6.4

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING AMENDMENT NO. 20 TO FACILITY LICENSE NO. DPR-26 CONSOLIDATED EDISON COMPANY OF NEW YORK, INCORPORATED INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 DOCKET NO. 50-247

INTRODUCTION

By letter dated September 6, 1974, as supplemented by letters dated October 21, 1974, November 6, 1974, December 2 and 6, 1974, January 29, 1975, April 21 and 29, 1975, May 21, 1975, July 9 and 21, 1975, February 4, 9, and 19, 1976, April 22, 1976, May 27, 1976, June 14, 1976, and July 13 and 15, 1976, Consolidated Edison Company of New York (Con Ed) submitted an amendment to License No. DPR-26 pursuant to 50.46 and Appendix K of 10 CFR Part 50 (ECCS) and the Commission's Order for Modifications of License dated December 27, 1974. By letter dated February 9, 1976, as supplemented by letters dated May 27, 1976, July 13, 1976 and July 15, 1976, ConEd requested an amendment to License No. DPR-26 to permit operation of Indian Point Unit No. 2 as reloaded for cycle 2.

This safety evaluation is a combined evaluation of Con Ed's submittal with respect to Appendix K (ECCS) and our evaluation of their submittal with respect to cycle 2 for Indian Point Unit No. 2.

ECCS Analysis

The licensee submitted an evaluation of the ECCS performance analysis in References 1 and 2, pursuant to the requirements of the Commission's regulations 10 CFR 50.46. The analyses submitted were based on the approved Westinghouse ECCS evaluation model (References 3 & 6). Reference 2 addresses the analysis of the loss of reactor coolant from small ruptured pipes (small break LOCA). Reference 1 addresses the analysis of the major reactor coolant system pipe ruptures (large break LOCA).

The large break LOCA analysis submitted by the licensee was limited to a spectrum of three breaks which were specific for the Indian Point, Unit 2 design (Reference 1). To supplement this analysis, the licensee referenced WCAP-8356, "Westinghouse ECCS-Plant Sensitivity Studies", and WCAP-8558, "Westinghouse ECCS - Four Loop Plant (15x15) Sensitivity Studies" (References 7 and 8), which demonstrated that the guillotine breaks are the worst cases for this plant type.

The analyses submitted identified the worst break size as the double-ended cold leg guillotine break with a Moody multiplier of 1.0. The calculated peak clad temperature was 2115^{o}F; within the acceptable limit of 2200^{o}F (as specified in 10 CFR 50.46(b)). In addition, the maximum local metal/water reaction of 4.42% and a total core-wide metal/water reaction of less than 0.3% were well below the allowable limits of 17% and 1%, respectively. The analyses were performed based on an assumed total peaking factor of 2.32 at 102% of rated NSSS power level of 2758 MWt, with a peak linear power density of 13.4 kw/ft.

The licensee's submittal of a small break LOCA analyses by letter dated September 6, 1974, included a spectrum of three breaks which were specific for the Indian Point, Unit No. 3 design (Reference 2). In addition, the licensee submitted by reference a generic Westinghouse topical report which documented additional break analyses (Reference 7). The small break analysis, which identified the 6-inch pipe break as the limiting small break with a peak clad temperature of 1765^{o}F, demonstrates that the small break LOCA is not limiting.

An evaluation was not provided for ECCS performance during reactor operation with one primary loop out of service. Therefore, continuous reactor operation under such conditions is not authorized. The reactor may, however, operate for periods up to 24 hours with one primary loop out-of-service. This short time period permits corrective action to be taken and reduces the number of shutdowns and is consistent with other Technical Specifications. Appropriate modifications have been made to the proposed Technical Specifications to establish this restriction on three loop operation.

Due to the configuration of the Westinghouse reactor vessel design, a small portion of relatively cooler reactor inlet water is directed through several nozzles located on the periphery of the vessel to cool the upper portion of the head. Accordingly, upper head temperatures were assumed in evaluating ECCS performance to be equal to the reactor inlet water temperature. However, recent operating data gathered at the Connecticut Yankee facility has indicated that, contrary to this expectation, the temperature of the water in the upper head is warmer than the reactor inlet water temperature, by about some 60% of the reactor inlet - reactor outlet temperature differential. This increase in upper head water temperature over that used in ECCS performance calculations would have the effect of increasing the calculated peak clad temperature.

In a meeting with the staff on August 9, 1976, Westinghouse presented generic evaluations of the effect on calculated peak clad temperature for the worst break identified in previous calculations for each type of Westinghouse reactor and fuel design using an upper head water temperature exceeding reactor inlet water temperature by an amount equal to 75% of the reactor inlet - reactor outlet differential. On August 12, 1976, the staff directed the licensee to submit an analysis similar to the Westinghouse evaluation with the clearly conservative assumption of upper head water temperature equal to reactor outlet temperature (100% of the reactor inlet - reactor outlet differential) and to operate the facility in accordance with the results of this analysis. The results of the evaluation submitted for the Indian Point, Unit No. 2 reactor (Reference 13) indicated that with this modification of the upper head water temperature the calculated peak clad temperature for the worst case break (2185^{o}F) would not exceed the Commission's ECCS performance criteria.

Revised calculations fully conforming to the requirements of 10 CFR §50.46 are to be provided for the facility as directed by the Commission's Order dated August 27, 1976. Based on the Westinghouse sensitivity studies as applicable to IP-2, we expect that, when revised calculations are submitted they will demonstrate that operation within the present total nuclear peaking factor limit would conform to the criteria of 10 CFR §50.46(b). The revised calculations must use an approved evaluation model with correct input for upper head water temperature, or with the assumption that the upper head water temperature equals reactor vessel outlet water temperature.

ECCS Containment Pressure Evaluation

The ECCS containment pressure calculations for Indian Point, Unit No. 2 were performed using the Westinghouse ECCS evaluation model. We have reviewed Westinghouse's model and found it acceptable for ECCS evaluation. We required, however, that justification of the plant dependent input parameters used in the analyses be submitted for our review of each plant. This information was submitted for Indian Point, Unit No. 2 by letter dated July 9, 1975 (Reference 9) and in Reference 1. Consolidated Edison Company reevaluated the containment net free volume, the passive heat sinks, and, operation of the containment heat removal systems with regard to the conservatism for the ECCS analysis. This evaluation was based on equipment inventories and structural drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities. Minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant dependent information used for the ECCS containment pressure analysis for the Indian Point, Unit 2 plant is reasonably conservative and therefore, the calculated containment pressures are in accordance with Appendix K to 10 CFR 50 of the Commission's regulations.

Single Failure Criterion

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. The worst single failure which could minimize the ECCS available to cool the core and provide maximum containment cooling was identified by Westinghouse as the loss of a low pressure ECCS pump. As stated in Reference 6, we concluded that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

A review of the Indian Point, Unit 2 piping and instrumentation diagrams indicated that the inadvertent actuation of specific motor-operated valves could affect the appropriate single failure assumptions.

During the initial operating license review, we identified nine motor operated valves for which removal of electric power would satisfy the single failure criterion. These nine valves are covered in the present Technical Specifications. In this current review we determined that the removal of power to these nine valves was acceptable when implemented in accordance with Branch Technical Position EICSB-18 (Reference 11). An appropriate modification to the proposed Technical Specifications has been made. The licensee and the NRC have now identified several additional valves that did not meet the single failure criterion and should be modified in accordance with EICSB-18. The proposed Technical Specifications have been appropriately modified. The following is a complete list of these valves identified by the licensee and the NRC which are to be modified in accordance with EICSB-18 (All of the modifications will be completed during current refueling outage):

MOV #	Component Function	Failure Mode
** 856 B&F	Isolation hot leg injection lines	Opening of either valve during ECC injection and core reflood will allow injection into RCS hot legs and could cause steam binding.
1810	Isolates RWST from SI pumps	Closure of this valve results in loss of RWST to both SI pumps
882	Isolates RHR system from RWST	Closure of this valve results in loss of suction flow to both RHR pumps
744	RHR pump discharge to RCS cold leg loops, containment spray header and SI pumps	Closure of this valve could result in reduction of ECC flow from RHR pumps

894 A, B C & D	Accumulator isolation valves	Inadvertent closing of these valves would stop accumulator flow
*842 *843	Shutoff SI pumps	For a small break, closure of this valve prior to initiation of recirculation could cause damage or malfunction of SI pumps. However, these valves must be closed once the recirculation phase is initiated.
**856 A, C, D, E	Supply high head SI flow to RCS and balance flow split to cold legs	856A and D may be submerged following a LOCA

* Identified by the staff during present review

** Potentially submerged valves (856A, B, and D), addressed in later section

We have reevaluated the consequences of the failures of the above valves and have evaluated the licensee's proposals for required modifications. We have concluded that the following modifications are acceptable for complying with the single failure criterion:

(1) As proposed by the licensee, AC power will be disconnected for valves 856 B&F, 1810, 882, and 744 by racking out of breaker at the motor control center during power operation. Valves 1810, 882, and 744 will be locked in their open position. Valves 856 B&F will be locked in their closed position.

(2) During power operation AC power will be disconnected for valves 894 A, B, C, & D by racking out of breaker at the motor control center. These valves will be locked in the open position, since inadvertent actuation of any of these valves would stop accumulator flow.

(3) During power operation AC power will be removed from valves 842 and 843. These valves will be in their open position and power will be restored from the control room for switchover to recirculation mode operation. Necessity for this action is explained in the above table.

Section 3.3.A.1 of the plant Technical Specifications presently includes a listing of the electrically-operated valves addressed in (1) and (2), above, with their required positions. These specifications have been modified to include racking out of circuit breakers to effect de-energization in accordance with EICSB-18.

The licensee has proposed to remove power from valves 842 and 843 with the valves in their opened position and to provide the capability to restore power from within the control room for switchover to recirculation mode operation. The licensee has submitted a design modification for staff review which the staff has found acceptable. The licensee has informed us by letter dated July 15, 1976, that they will complete these modifications during the current refueling outage.

As noted in Table 3-1, valves 856 A, C, D and E are motor operated valves in the high pressure safety injection lines to the cold leg loops which are adjusted during preoperational flow tests to provide a balanced flow split to the RCS. Based on recent operating experience , we have modified the proposed Technical Specifications to establish additional surveillance requirements as follows:

For the four distribution valves (MOV-856, A, C, D, and E) verification of the valve mechanical stop adjustments will be performed periodically as noted below to provide assurance that the high pressure safety injection flow distribution is in accordance with the flow values specified in the Safety Analysis Report,

(a) by conducting a flow test of the HPSI system after any modification is made to either the HPSI piping and/or valve arrangement and,

(b) by verification that the mechanical stops are set at the position measured and recorded during the most recent of either the ECCS preoperational flow tests or flow tests performed in accordance with (a), above. This surveillance procedure shall be performed following any maintenance on these valves or their associated motor operators, and at any convenient outage if the position of the mechanical stops has not been verified within the previous three months.

Long-Term Boron Concentration Buildup

We have reviewed the proposed procedures and the system designed for preventing excessive boric acid buildups in the reactor vessel during the post-LOCA, long-term cooling period. The licensee has proposed that the switchover time from cold to simultaneous hot and cold leg injection should occur 24 hours after a LOCA. This time will assure that for cold leg breaks, the concentration of boric acid will not exceed 23.5 weight percent, which is 4 weight percent below the solubility limits at 212^{o}F. We have concluded that the required 4 weight percent safety margin will adequately compensate for potential variations in local concentrations of boric acid in the reactor vessel.

We have reviewed the piping and instrumentation diagrams and found that the system proposed by the licensee for long-term cooling can be operated in a manner complying with the single active failure criteria, provided that the system is modified to assure that potentially submerged motor operated valves which are required to function during this post-LOCA period, will perform their intended function. The licensee identified potentially submerged valves and proposed modifications to those valves important to post-LOCA cooling. These modifications are discussed below:

Submerged Valves

The licensee has submitted an analysis (Reference 5) which shows that following a LOCA the maximum water level inside containment will be at the 50 ft. - 1 inch elevation (containment floor elevation is 46 feet). The licensee has identified 25 motor-operated valves which may be submerged post-LOCA. These valves, whose motor operators are located below the flooded elevation, are listed below:

Potentially Submerged Motor Operated and Air Operated Valves

Valve No.	Description
856A	HPSI to loop 1 cold leg
856D	HPSI to loop 2 " "
856B	HPSI to loop 3 hot leg
123	Excess letdown control, CVCS
200B	Letdown orifice isolation, CVCS
200C	Letdown orifice isolation, CVCS
212	Auxiliary spray valve, CVCS
891B	Accumulator 22 N_2 fill valve
891D	" 24 " " "
896A	" 21 Drain Valve
896B	" 22 " "
896C	" 23 " "
896D	" 24 " "
955C	" 23 Sample Valve
955D	" 22 " "
955E	" 23 " "
955F	" 24 " "

Valve No.	Description
1003A	RCDT Level Control Valve
1003B	" " " "
1163	Condensate Weir Drain Valve - WDS
1164	" " " " "
1165	" " " " "
1166	" " " " "
1167	" " " " "
1609	PRT Drain Valve

The majority of these valves either have no ECCS function (valves 123, 200B, 200C, 212, 1003A, 1003B, 1163, 1164, 1165, 1166, 1167, 1609) or, their malfunction due to submergence following a LOCA would have no impact on the required ECCS function (896A, 896B, 896C, 896D, 955C, 955D, 955E, 955F).

The licensee has submitted an analysis of the remaining submerged valves. During the injection phase following a postulated LOCA, all of the cold leg injection valves must be opened (valves 856 A, C, D and E) and the hot leg injection valves must be closed (856 B and F) to assure that minimum required ECCS flow is delivered assuming a single active failure. After approximately 24 hours, the ECCS will be transferred to the hot leg recirculation mode of operation to prevent excessive boric acid buildup in the reactor vessel. The Indian Point, Unit 2, emergency operating procedures specify that hot leg recirculation be established in each of the HPSI trains by closing 1 of the 2 cold leg injection valves and opening the hot leg injection valve. To establish this realignment, valves 856A, B, C, D, E, and F must be operable. To assure that these valves are properly aligned and operable as required the licensee has proposed to:

(a) Relocate the motor operators on valves 856 A, B, and D by elevating them such that the bottom of the lowest valve operator is above the maximum calculated water level.

(b) Modify the valve control logic in each HPSI train to assure that each hot leg injection valve is prevented from being opened until a cold leg valve has been closed to prevent the HPSI pumps from exceeding their maximum flow limit (runout).

(c) Provide redundant position indication for all deenergized valves in the control room to assure that valve status is in the proper safeguards position for the depowered condition. The licensee has agreed to adhere to Branch Technical Position, EICSB-18 when removing power to the valves 856 B & F.

We have reviewed the licensee's submittal and concluded that the proposed modifications to the 856-Series valves are acceptable.

By letter dated May 27, 1976, the licensee stated that modifications would be accomplished in accordance with the Indian Point Unit No. 2 seismic class 1 criteria.

The licensee has also identified (Reference 10) air operated valves 891 B and D, containment isolation valves for the accumulator Nitrogen fill lines, as potentially submerged post-LOCA. To assure proper actuation, the licensee proposed to raise the solenoids for the air operators above the maximum flood level during the present refueling outage. We find this acceptable.

By letter dated July 15, 1976, the licensee stated that all proposed modifications to potentially submerged valves will be completed during the current refueling outage.

Rod Bow

Recent generic information provided by Westinghouse indicates that the effects of rod-to-rod bowing on DNBR and local power spike should be considered in evaluating the thermal hydraulic design and ECCS performance.

The licensee has stated that Indian Point, Unit No. 2, Cycle 2, will utilize a high-parasitic nine grid (HIPAR) fuel assembly skeleton. The amount of fuel rod bowing has been found to be significantly less than that of fuel rods in low parasitic Westinghouse fuel assemblies.

Westinghouse, the fuel supplier for Indian Point Unit No. 2, has recently conducted DNB (departure from nucleate boiling) experiments which showed a significant decrease in the DNB ratio (DNBR) when bowing takes place in the presence of unheated rods such as thimbles. (Reference 12) In these tests the rods were bowed to contact. Data taken by the licensee on the amount of fuel rod bowing with HIPAR fuel (Reference 13) shows that the maximum amount of bowing observed is approximately 33% clearance reduction. Thus, the effect of rod bowing on DNBR would be less than that predicted in the Westinghouse experiments. In addition, the licensee has used an $F\Delta H$ of 1.65 for thermal calculations while the Indian Point Unit No. 2 technical specifications require an $F\Delta H$ of 1.55. This provides more than adequate margin to assure staying above a DNBR of 1.3 for all anticipated transients. Therefore, we conclude that the licensee has adequately accounted for the effects of rod bow on thermal hydraulic performance for 15x15 high parasitic fuel and that no additional operational penalties need be applied to Indian Point, Unit No. 2, Cycle 2.

Rod bowing also affects the magnitude of the power in a fuel rod and thus affects F_Q^N. This effect has also been accounted for by the licensee in a manner acceptable to the staff.

Submerged Electrical Equipment

The licensee has identified the following electrical components located below the maximum calculated LOCA flood level inside containment (50 ft - 1 inch elevation):

All Electrical Penetrations on Lowest Level: H58, H59, H60, H61, H63, H64, H66, H67, H68, H69, H70.

Pump Motors: Reactor Coolant Drain Tank Pump #21 and Junction Box;
Reactor Coolant Drain Tank Pump #22 and Junction Box;
Containment Sump Pump #29 and level switch; and
Containment Sump Pump #210 and level switch.

Zone #24 - Rack 14 - Weld Channel and Penetration Pressurizer System Junction Box

LT - 1003 (Reactor Coolant Drain Tank (RCDT) Level Transmitter)

TT - 1058 (RCDT Temperature Transmitter)

TE - 122 (Excess Letdown Line Temperature Element)

TE - 126 (Charging Line Temperature Element)

LT - 1133 (ΔP cell for Fan Cooler Weir #21)

LT - 1134 (ΔP cell for Fan Cooler Weir #22)

LT - 1135 (ΔP cell for Fan Cooler Weir #23)

LT - 1136 (ΔP cell for Fan Cooler Weir #24)

LT - 1137 (ΔP cell for Fan Cooler Weir #25)

FT - 925 (High Head Line to Cold Leg #3 - Flow Transmitter)

FT - 926 (High Head Line to Cold Leg #4 - Flow Transmitter)

FT - 946B (Low Head Line to Cold Leg #3 - Flow Transmitter)

LT - 938 (Recirculation Sump Level Transmitter)

Of these components only four have been identified by the licensee as safety-related. These components and the licensee's proposed corrective action to be completed during current refueling outage are as follows:

1. Flow transmitter FT-925 for the high head injection line to cold leg #3 will be relocated above the maximum flood level.
2. Flow transmitter FT-926 for the high head injection line to cold leg #4 will be relocated above the maximum flood level.
3. Flow transmitter FT-946B for the low head-RHR injection line to cold leg #3 will be relocated above the maximum flood level.
4. Level transmitter LT-938 is one of two redundant transmitters which provide level indication in the central control room for the recirculation sump. Design changes to this level transmitter are not necessary since it has been designed and qualified for submerged service in borated water at 295°F at a pressure of 69 psig. These conditions are more severe than the post-LOCA design conditions. This qualification has previously been documented in the Indian Point Unit No. 2 FSAR.

In addition to these changes, the licensee has completed modifications to two circuits whose respective electrical containment penetrations are submerged.

(1) Penetration H-69 contains the electrical feed to a 480 volt motor control center inside containment. Although this non-safety related power supply is de-energized by a safety injection signal or undervoltage signal, the licensee has installed fuses in series with the existing circuit breaker as an added measure of circuit protection.

(2) Penetration H-70 contains the emergency lighting feed to a lighting panel within containment and is fed by emergency DC power. The licensee has installed a new circuit breaker in series with new fuses in order to provide redundant protection for the emergency DC bus. This modification assures that no single component failure will compromise a safety bus. In addition, the licensee will modify his operating procedures such that the circuit breaker will be locked open during normal plant operation and under accident conditions. This procedural change will assure that penetration H-70 is de-energized whenever access to containment is not required.

The licensee has provided the results of an evaluation of the effects of submerged electrical component failures on ECCS performance, containment isolation, and other safety-related functions. In each case, primary and backup electrical protection for the circuits of submerged components was considered. The following two modifications have been implemented to provide adequate breaker and fuse coordination.

(1) The circuitry for non-safety related valves 1163, 1164, 1165, 1166, and 1167, which control letdown through the Containment Recirculation Fan Condensate Measuring System, has been modified by installing fuses in series with the existing circuit breaker to provide redundant fault protection and render these circuit protection devices consistent with others.

(2) Level transmitter LT-1003 circuitry has been modified by installing a fuse in series with the existing circuit breaker in order to provide the same redundant fault protection.

We find the foregoing modifications (Items 1 and 2) acceptable.

Engineered Safeguards

1 Safety Injection System

The licensee has stated that the sensing logic for the Safety Injection System (SIS) of Unit No. 2 is identical to that of Unit No. 3. The differences between Units 2 and 3 safety injection systems that require design changes are addressed below:

(a) Safety Injection Block Switch

The licensee identified a single safety injection block switch whose failure may affect the redundant logic trains of low pressurizer pressure/level. To preclude the effects of this failure, the licensee has provided a second independent and redundant switch. The circuit has been modified such that each switch is associated with only one of the two redundant logic trains. This meets single failure criterion and is acceptable.

(b) Bypass of Redundant Engineered Safety Feature Logic Trains

The relay logic scheme that provides the signal to actuate safety injection equipment is tested for continuity manually by use of an ohmmeter rather than by the use of test relay and test light combinations typical of Unit No. 3. In order to perform this test, individual logic matrices on a single train will be bypassed, one at a time, by operation of any one of the associated logic relay test switches. Since the safety function associated with this matrix on one train is inoperative, the same function on the redundant train must remain operative. To preclude bypassing the second train of this ECCS function, the licensee has installed separate annunciation devices which alarm whenever either train is bypassed. Administrative procedures do not permit the placement of the opposite safeguard train in test.

With these modifications and procedural changes to be completed during current refueling outage, there is added assurance that an operator error will not compromise redundant trains. We find this acceptable.

(c) Redundant Train Actuations

Both trains of the SIS of Unit No. 2 are used to actuate each electrically operated component of the Emergency Core Cooling System. Although this design provides for redundancy and flexibility of equipment operation, we requested that the licensee determine if any single failure could compromise redundant trains. The licensee provided information on the Westinghouse relays which are used to actuate the ECCS equipment. The licensee has stated that the construction of these relays is such that it is virtually impossible to short circuit the coil to any individual contact or to short circuit between adjacent contacts. In addition, two relay failures in redundant safeguards cabinets would be required to compromise redundant trains of actuated equipment. The design of this portion of the actuation system is therefore in conformance with the single failure criterion and is acceptable.

2 Residual Heat Removal (RHR) Electrical Interlocks

A single reactor coolant system pressure transmitter provides an interlocking signal which prevents manual opening of two series connected high pressure to low pressure RHR valves (Valves 730 and 731) whenever the reactor coolant system (RCS) is above a predetermined pressure. To meet the single failure criterion, a second independent pressure transmitter will be installed to provide a separate, independent interlocking signal to one of the two valves. The existing pressure transmitter will provide its signal to the other valve. This redundant circuit design will assure that malfunction of a single pressure interlock will not allow the opening of both RHR valves at high RCS pressure. Until this modification is completed, valves 730 and 731 will be de-energized in the closed position whenever the RCS pressure is above the RHR system design pressure (letter from ConEd, dated May 27, 1976).

The electrical interlock between safety injection valves 888A and 888B and RHR valves 730 and 731 will be changed such that valve 730 will be interlocked with valve 888A and valve 731 will be interlocked with valve 888B. This modification to be completed during current refueling outage will assure the availability of a path for delivery of recirculated fluid to the suction side of the safety injection pumps in the event of a single failure.

We find that this design is acceptable because it meets the single failure criterion for high pressure to low pressure isolation.

Switchover from the Injection Mode to the Recirculation Mode of ECCS Operation

We have identified certain switches which, if not properly positioned, could prevent automatic initiation of redundant safety equipment. In addition to switch position indication, the licensee has added contacts, which open upon safety injection actuation, in series with the following switches or interposing relay contacts:

(1) Switch 3

"43/RS-3" trip to each RHR pump

(2) Switch 6

"43/RS-6" open signal to valves 888A and B

"43/RS-6" close signal to valves 746 and 747

(3) Switch 7

"43/RS-7" trip to each SI pump

These changes will defeat the recirculation switch function during the first phase of safety injection operation. However, two minutes after SI initiation the recirculation switches may manually be made functional. Since redundant switches are available for each actuated system, this change conforms with the single failure criterion.

4. <u>Miniflow Bypass Valves for RHR Pumps</u>

The miniflow bypass valves provide a recirculation path through heat exchangers from the discharge to the suction side of the RHR pumps whenever the pumps are dead-headed.

To assure that a postulated single failure of the RHR miniflow bypass valves 743 or 1870 would not cause interruption of RHR pump miniflow, these two valves will be made passive by having their electric power physically disconnected and locked in the open position. This will assure that these valves will no longer be electrically operable. We find this proposal acceptable.

Emergency Onsite Power System

1. Diesel Generators

The Onsite Power System consists of three independent diesel generators. Two diesels feed independent buses. The third diesel feeds two buses through separate switchgear. Two of the three diesel generators are required to mitigate the worst case consequences of a Loss of Coolant Accident. The design is such that any single electrical failure will not prevent the required engineered safety feature performance under accident conditions. In addition, the design does not allow redundant safeguard buses to be automatically tied together. We find that this design is acceptable.

2. Automatic Transfer Devices

The licensee has identified seven automatic transfer circuits used with engineered safeguards. Three automatic transfer circuits provide redundant 125 VDC control power to the three diesel generators. The remaining four transfer circuits provide redundant power to the 480V diesel generator switchgear. Each transfer device receives its 125V DC power from the same two emergency battery buses. The licensee has provided two circuit interrupting devices between the auto transfer device and each DC bus. The licensee has stated and we have verified that no single failure in the transfer device circuitry would cause the loss of either DC bus. Although it is possible to connect redundant power sources in parallel considering an undetected failure, two separate short circuits to ground (or a line to line short) and the failure to function of four overcurrent protection devices would be required to compromise redundant DC buses.

The licensee uses ground detectors as an integral part of the Westinghouse Battery Chargers. If a ground is present on a DC bus, a ground indicating light will go out. In addition, a "battery charger trouble" alarm will annunciate in the Central Control Room. The circuit grounding problem will then be isolated and corrected.

The licensee has also agreed to incorporate a test procedure for the ground detection system in the periodic battery testing program.

This design, with the above periodic testing, meets the single failure criterion and is acceptable.

3. <u>118V AC Instrument Buses</u>

Two 118V AC Instrument buses are fed from independent DC buses through separate inverters. A third instrument bus is fed from a 480V AC emergency bus through a transformer and solatron. The fourth 118V AC instrument bus is fed from offsite power through a 480V to 120V transformer and solatron. The instrument bus design is fail safe (trip) on loss of power, except for containment spray pump initiation logic. Power supplies for the containment spray actuation logic are supplied by safeguards power sources such that minimum requirements for system redundancy are satisfied.

Based on the above evaluation, we conclude that the design of the Onsite Power System meets the single failure criterion and is acceptable.

Environmental Qualification of Equipment Inside Containment

The licensee has identified all equipment and components located in the primary containment which are required to be operable during and after a LOCA. Included in this list are valve motors, fan cooler motors, recirculation pump motors, cable, and all instrumentation required to withstand the worst case effects of a LOCA. Qualification parameters include containment pressure, temperature, radiation, humidity, and chemistry. The environmental qualification status was reviewed at the FSAR review stage and based on the conclusions in Section 8.6 of the Safety Evaluation Report, we find the results of the environmental testing program acceptable.

Conclusions (Electrical, Instrumentation and Control Systems)

The licensee has identified design differences in the Electrical, Instrumentation and Control Systems on Unit No. 2 from those on Unit No. 3. This evaluation consists of a review of these differences. Those portions of the ECCS which are identical to those of Indian Point Unit No. 3 and found acceptable in the Unit 3 evaluation, were also accepted on Unit No. 2.

On this basis, we find that the proposed modifications and procedural changes allow the ECCS to withstand any single electrical failure without loss of function, and are acceptable.

Reload Cycle 2

Introduction

The licensee, by letter dated February 9, 1976 (Reference 11) indicated that he planned to reload Indian Point, Unit 2, with 72 fresh fuel assemblies and to replace 65 Region 1 and 7 Region 2 assemblies with 74 Region 4 assemblies. The analyses performed for this reload were done by Westinghouse using calculational methods which had been previously accepted by NRC. This evaluation showed that the important safety related parameters were enveloped by those used in previous analyses or that the accepted criteria of previous analyses were not exceeded. As a result, the licensee concluded that this reload did not involve any unreviewed safety question.

The licensee reviewed the FSAR analyses of accidents to determine which accidents could potentially be affected by cycle 2 conditions. A reanalysis of those accidents potentially affected by cycle 2 conditions was submitted in addition to justification of the applicability of other FSAR accident analyses.

The staff evaluation is based on a review of the Cycle 2 Reload submittal dated February 9, 1976, as supplemented by letters dated May 26, 1976 and July 13, 1976. The nominal design parameters for Cycle 2 are 2758 MWt core power, 2250 psia system operating pressure and core average linear power of 5.8 kw/ft.

Fuel System Design, Cycle 2

The mechanical design of the Region 4 fuel is nearly identical to that of Regions 2 and 3. The maximum irradiation expected at the end of Cycle 2 is 21,500 EFPH. Since clad flattening is not expected to occur for less than 30,000 EFPH for Regions 2 and 3, we agree with the licensee that clad fattening would not occur during Cycle 2 operation.

Nuclear Design, Cycle 2

1. Core Characteristics

   The Cycle 2 core loading will consist of 57 fuel assemblies (17,800 MWD/MTU Burnup) of Region 2, 64 fuel assemblies (12,700 MWD/MTU Burnup) of Region 3, and 72 fresh fuel assemblies of Region 4. Depleted burnable poison rods will be inserted in twenty-eight Region 4 and four Region 3 assemblies to reduce the radial peaking factor. Two of the Region 3 assemblies contain secondary source rods and their associated burnable poison and will be symmetrically loaded. The two Region 3 assemblies, symmetric (90^0) to the secondary source rod assemblies, have matching burnable poison inserted to preserve core symmetry.

   For the Cycle 2 core loading, the maximum differential worth of two control rod banks moving together in their highest worth region is equal to or less than 80 PCM/sec; the current limit for Cycle 1 is 80 PCM/sec. The total

Cycle 2 trip reactivity insertion rate is unchanged from Cycle 1. However, the Cycle 2 trip reactivity insertion rate for the upper third of the core is more rapid than in Cycle 1, but slower for the remainder of the core. The axial flux shape is normally distributed evenly with constant axial offset control. The change in trip reactivity versus rod insertion will not affect slow system transients since they are relatively insensitive to changes in trip reactivity insertion rates. Fast transients, such as rod ejection and rod withdrawal from subcritical conditions, will be unaffected by the change in trip reactivity insertion rates since the transients would be turned around due to Doppler feedback before rod insertion would begin. The licensee evaluated the effect on minimum DNB ratio of variations in trip reactivity insertion rates for the rod withdrawal at power incidents. Since the minimum DNB ratio for this transient occurs at low reactivity insertion rates, there was no affect on the overall minimum DNB ratio. The loss of flow transient, which is also sensitive to the rate of trip reactivity insertion, was not reanalyzed since the calculated Cycle 2 insertion rate is more rapid in the upper third of the core and would reduce core power earlier in the transient than for Cycle 1, and therefore be more conservative.

The Cycle 2 beginning of life delayed neutron fraction is lower than the values used for Cycle 1 analyses, and, as a result the rod ejection transient cases were reanalyzed using the reduced delayed neutron fraction.

The Cycle 2 Doppler coefficient is more negative than the Cycle 1 values. Accidents which are potentially affected, such as loss of flow, loss of load and locked rotor, were reevaluated. Only the loss of load transient has been reanalyzed, with the more negative Doppler coefficient, since the loss of flow and locked rotor transients in terms of minimum DNB ratio were found to be relatively insensitive. The results of these analyses are discussed below in the accident analyses section. All other transients and postulated accidents are bounded by the previous Cycle 1 analyses. The small break LOCA was reanalyzed only to provide additional operational flexibility during Cycle 2.

2. <u>Power Distribution</u>

The Licensee has provided predictions of the maximum peaking factor, $F_Q(Z)$, as a function of core axial height for Cycle 2 core characteristics. The $F_Q(Z)$ calculations were performed using constant axial offset control (CAOC) procedures with a $\pm 5\%$ ΔI band. The predictions considered various load following maneuvers as a function of extremes in possible depletion modes of the reactor, control strategies and magnitude of load follow. The maximum

$F_Q(Z)$ calculated was compared with the $F_Q(Z)$ limit, which must be maintained to avoid exceeding the linear power density used for the LOCA analysis.

For Indian Point Unit 2 Cycle 2 the results of the calculations indicate that the $F_Q(Z)$ limit (2.32) will not be exceeded using the proposed constant axial offset control Technical Specification target band of $\pm 5\%$. To assure that the Cycle 2 power control maneuvers, allowed by the Technical Specifications will satisfy these operating limits, the licensee proposed a modification to the Cycle 1 Technical Specifications to increase the power distribution limits in the upper portion of the core. To justify this broader operating limit, the small break LOCA was reanalyzed and was found to satisfy the FAC criteria of 10 CFR 50.46 as noted in the accident analysis sections, below.

Accident and Transient Analyses, Cycle 2

1. Loss of Load Transient

   The Cycle 2 Doppler coefficient, as noted above, was more negative than the Cycle 1 values. The loss of load transient was reanalyzed using the Cycle 2

Doppler Coefficient. The analysis was performed using the methods and assumptions previously employed and accepted for Cycle 1. The results of this reanalysis show that the minimum DNB ratio did not drop below 1.3 and that the pressurizer and steam generator safety valves are more than adequate to limit the maximum reactor coolant and main steam system pressures to within acceptable limits. We have concluded that the results of these analyses are acceptable.

2. Rod Ejection Transient

The rod ejection transient was reanalyzed to account for Cycle 2 beginning of life (BOL) delayed neutron fractions being lower than these used for Cycle 1 analyses. The hot zero power (HZP) end of life (EOL) transient was reanalyzed to account for the increased peaking factor for Cycle 2. In addition, the licensee reanalyzed the hot full power (HFP) EOL case due to an increase in average fuel temperature for Cycle 2 resulting from a change in the methodology of representing the hot spot linear power density. For the HFP-EOL case, the licensee conservatively assumed a value of 0.67% for $\Delta\rho$ and 15.3 for the maximum F_Q^N; the more limiting conditions for Cycles 1 and 2. For the remaining three cases (1) HFP-BOL, (2) HZP-BOL, and (3) HFP-EOL, the values of $\Delta\rho$ and maximum F_Q^N used in the Cycle 1 analysis were also used for Cycle 2, analysis. This is a conservative approach since Cycle 1 analysis assumed the use of part length rods to push power toward the top of the core which would maximize the worth of the ejected rod. Since part length rods will not be used in Cycle 2 and the rod insertion limits for Cycle 2 have been changed to provide additional shutdown margin with respect to Cycle 1 we have concluded the consequences of an ejected rod are acceptable.

3. Small Break LOCA

The licensee proposed to change K(Z), the power distribution limit to gain additional operational flexibility for Indian Point Unit 2 Cycle 2. Since the peak cladding temperature developed during small break LOCA is a function of power in the upper region of the core, the small break was reanalyzed to justify the licensee's proposed increase in the K(Z) limit. Presently referenced on the IP-2 docket is a conservative small break LOCA analysis which was performed at the more restrictive IP-3 operating conditions (see section on ECCS Analysis, pages 2 and 3). The licensee reanalyzed the 4, 6, 8, and 9.57 inch diameter break sizes specifically for the IP-2 plant using the approved versions of the WFLASH and LOCTA-IV computer codes in accordance with Appendix K to 10 CFR 50. A core power distribution was assumed that was skewed toward the top of the core to maximize the calculated peak cladding temperature. The results of this analysis, when compared to the IP-3 results referenced on the IP-2 docket, showed a calculated peak cladding temperature of 1380°F for the limiting 8 inch break size which was lower than the IP-3 result of 1765°F. The licensee has submitted this analysis by letter dated July 13, 1976. We have concluded that the peak cladding temperatures and oxidation associated with the small break conform with the criteria of 10 CFR 50.46, and that operation of the plant within the proposed Technical Specification limits on power distribution is acceptable.

The licensee, in response to our request has submitted Technical Specification limits to assure that the thermal margin is maintained during steady state operation and during operational transients. These limits are:

Pressurizer Pressure ≥2220 psia

Core Average Temperature ≤573.5°F

RCS Flow ≥358,800 gallons per minute

The pressurizer pressure and core average temperature limits take measurement uncertainties into account. The core flow value is nominal flow and assures coreflow in excess of that assumed for transient and accident analyses.

The licensee estimates the flow measurement uncertainty to be three to five percent. The value of the flow proposed by the licensee and given above is the value of the flow used by the licensee in the steady state analyses aud in the analyses of anticipated transients and accidents. We have concluded that the proposed Technical Specifications would provide assurance that the thermal margin would be maintained during steady state operations, anticipated transients, and accidents.

Conclusions

Based on our review, we have determined that:

1) The LOCA analyses that were performed are in accordance with the requirements of Appendix K to 10 CFR 50.

2) The ECCS cooling performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR 50.46.

3) ECCS cooling performance will be adequate despite any postulated failure of a single active component.

4) Adequate systems and procedures exist to provide long term cooling to the reactor vessel.

5) The proposed modifications and procedural changes make the ECCS capable of withstanding any single electrical failure without loss of function, and are acceptable.

6) The proposed core reload will not adversely affect the safety of the plant and that it is acceptable for the licensee to proceed with Cycle 2 operation in the manner proposed.

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

Dated: September 4, 1976

REFERENCES

1. "ECCS Acceptance Criteria Analysis: Major Reactor Coolant System Pipe Rupture, Indian Point Generating Station Unit 2," Submitted by letter from Consolidated Edison Company of New York to B. C. Rusche, dated February 4, 1976, Docket No. 50-247.

2. WCAP - 8399 - "ECCS Acceptance Criteria Analysis, Indian Point Nuclear Generating Station, Unit 2, September, 1974.

3. Letter to C. Eicheldinger from D. B. Vassallo, "NRC Staff Review of the Westinghouse ECCS Evaluation Model," dated May 30, 1975.

4. Letter to G. Lear from W. J. Cahill, Transmittal of Emergency Operating Procedure E-2, dated April 29, 1975.

5. Letter to R. W. Reid from W. J. Cahill, "Potential for Submerged Valves within Containment Following a LOCA," dated February 19, 1976.

6. "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," October 15, 1974.

7. WCAP - 8356 - "Westinghouse ECCS - Plant Sensitivity Studies," July, 1974.

8. WCAP - 8558 - "Westinghouse ECCS - Four Loop Plant (15x15) Sensitivity Studies," October 10, 1975.

9. "Analysis of the ECCS in Accordance with the Acceptance Criteria of 10 CFR 50.46 and Appendix K of 10 CFR 50, "Submitted by letter dated July 9, 1975.

10. Letter to Robert W. Reid from W. J. Cahill, dated April 22, 1976.

11. Letter to Ben C. Rusche from LeBoeuf, Lamb, Leeby and MacRae, dated February 9, 1976, Transmitting Reload Safety Evaluation for Indian Point Nuclear Plant, Unit 2, Cycle 2.

12. Letter to V. Stello from Westinghouse Electric Corporation dated August 17, 1976, Proprietary.

13. Letter to J. O'Reilly from W. J. Cahill, dated August 17, 1976.

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.20 TO LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

I. Description of Proposed Action

By letters dated September 6, 1974, October 21, 1974, November 6, 1974, December 2 and 6, 1974, January 29, 1975, April 21 and 29, 1975, May 21, 1975, July 9 and 21, 1975, February 4, 9, and 19, 1976, April 22, 1976, May 27, 1976, June 14, 1976, July 13 and 15, 1976, and August 17, 1976, the Consolidated Edison Company (ConEd) provided information and supportive analysis relative to a proposed change in the Appendix A Technical Specifications of Facility License No. DPR-26. The proposed change concerns revisions to the limiting conditions for operation to the Indian Point Nuclear Generating Unit No. 2 as a result of the implementation of the Acceptance Criteria for the Emergency Core Cooling System (ECCS) and reload for cycle 2 operation.

II. Environmental Impacts of Proposed Action

The NRC has evaluated the potential environmental impacts associated with this proposed license amendment as required by the NEPA and Section 51.7 of Part 51 CFR.

The potential NEPA concerns associated with the implementation of the ECCS Criteria for Indian Point Unit No. 2 and cycle 2 operation can be defined as:

1. Changes in benefits accruing from plant operation due to revisions to reactor power limits. (There are no changes in reactor power limits in this amendment and no change in planned power.)

2. Variation in environmental impacts resulting from changes in non-radiological effluent releases. (There are no changes in the non-radiological effluent limits or the potential for release of non-radiological effluents as a result of this amendment to the Technical Specifications.)

3. Variation in environmental impacts resulting from changes in radiological effluent releases. (There are no changes in radiological effluent limits or the potential for release of radiological effluents as a result of this amendment to the Technical Specifications.)

Since this amendment will not result in modified power levels and no changes in radiological or non-radiological effluents, there are no changes in the Cost/Benefit balance. Fuel characteristics, cooling water flow, thermal effluents, chemical effluents, and radiological source term during operation and postulated accident conditions will not be revised as a result of the implementation of the ECCS Acceptance Criteria or cycle 2 operation. The only changes to the facility involve improvements to meet ECCS criteria.

III. Conclusions and Basis for Negative Declaration

On the basis discussed above, we conclude that the amendment to the Technical Specifications to implement the ECCS Acceptance Criteria for Indian Point Unit No. 2 or the amendment to the Technical Specifications for cycle 2 operations does not involve a significant change in the types or significant increase in the amounts of effluents or a significant increase in the potential for accidental releases. Therefore, we conclude that there will be no significant environmental impact attributable to this licensing action.

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for the proposed license amendment and that a Negative Declaration shall be issued to this effect.

Dated: September 4, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-247

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 20 to Facility Operating License No. DPR-26, issued to The Consolidated Edison Company of New York, Inc. (the licensee), which revised Technical Specifications for operation of the Indian Point Nuclear Generating Unit No. 2 (the facility), located in Buchanan, Westchester County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to establish operating limits for Indian Point Unit No. 2 as reloaded for cycle 2 operation based upon an acceptable Emergency Core Cooling System evaluation model conforming to the requirements of 10 CFR 50.46, and terminates the operating restrictions imposed by the Commission's December 27, 1974, Order for Modification of License.

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 rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notices of

Proposed Issuance of Amendment to Facility Operating License in connection with this action were published in the FEDERAL REGISTER on September 26, 1975 (40 F.R. 44362) and April 15, 1976 (41 F.R. 15917). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action and that a negative declaration to this effect is appropriate.

For further details with respect to this action, see (1) the applications for amendment dated July 9, 1975 and February 9, 1976, as amended and supplemented, (2) Amendment No. 20 to License No. DPR-26, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Impact Appraisal. 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 Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, New York.

A copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of September 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors