POWER AUTHORITY OF THE STATE OF NEW YORK

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February 16, 1979
IPN-79-4

Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. Albert Schwencer, Chief
Operating Reactor Branch No. 1
Division of Operating Reactors

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
ACRS Concerns

Dear Sir:

In our letter of October 13, 1978 (IPO-162), the Authority offered to provide documentation of the extent to which the IP3 facility satisfies the ACRS resolved generic items.

Attached please find ten (10) copies of a document entitled "Comparison of the Indian Point 3 Nuclear Power Plant to the ACRS Resolved Generic Items Contained in ACRS Report No. 6 to the NRC". This document satisfies our commitment in Item 5 of the above mentioned letter. If you have any questions on this document, please contact us.

Very truly yours,

Paul J. Early
Assistant Chief Engineer-
Projects

Att.

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DOCKET NO. 50-286 ACC# 7902220124
COMPARISON OF INDIAN PT 3 NUCLEAR PWR PLANT TO ACRS RESOLVED GENERIC ITEMS CONTAINED IN ACRS RPT 6 TO NRC.
rec'd w/ltr dtd 2/16/79

COMPARISON OF
INDIAN POINT 3 NUCLEAR POWER PLANT
TO ACRS RESOLVED GENERIC ITEMS
CONTAINED IN ACRS REPORT NO. 6 TO THE NRC
(RESPONSES TO GROUPS I, IA, IB, IC, ID, and IE)

Power Authority of the State of New York
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

February, 1979

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ITEM NO.: I-1

SUBJECT: Net Positive Suction Head for Emergency Core Cooling System Pumps

Conformance to Reg. Guide 1.1 is documented by NRC staff review and approval of response to FSAR Question 6.4.

ITEM NO.: I-2

SUBJECT: Emergency Power

In the NRC Safety Evaluation Report, 9/21/73, staff concluded that IP3 design of the onsite A-C power system is in conformance with Reg. Guides 1.6 and 1.9 and with IEEE Std 308. All modifications to the plant following issuance of FOL DPR-64 were and are being made to conform to the latest revision to IEEE 308 in effect at the time of the modification.

The intent of Reg. Guide 1.32 was satisfied as follows:

1. Technical Specification §3.7: two immediate access circuits from the transmission network are provided.

2. The battery charger supply capacity is based on the largest combined demands for the various steady-state loads and the charging capacity to restore the battery from the minimum-charge state to the full-charge state.

3. Battery performance discharge and battery service tests are conducted in a manner which satisfies the requirements of IEEE Std 450.

4. Physical independence has been incorporated in the design as indicated in the basis to Technical Specification §3.7.

5. The intent of the guidance presented in Reg. Guide 1.75, "Physical Independence of Electric Systems", (Revision 1, 2/74) has been followed for the connection of Non-Class IE equipment to Class IE equipment.

ITEM NO.: I-3

SUBJECT: Hydrogen Control After A Loss-of-Coolant-Accident

This matter was resolved at the time of issuance of FOL DPR-64; on the basis of responses to FSAR Questions 6.17 and 6.25. The response to FSAR Q6.17 discussed the compliance of the IP3 design with Reg. Guide 1.7. The response to FSAR Q6.25 indicated that an analysis of hydrogen production and accumulation in the containment following the Design Basis Accident based on the parameter values of Reg. Guide 1.7 had been performed.

NRC staff in the Safety Evaluation Report (9/21/73) §6.2.4, pg. 6-12 concluded that the combustible gas control systems met the recommendations of Reg. Guide 1.7.

ITEM NO.: I-4

SUBJECT: Instrument Lines Penetrating Containment

The NRC staff in Safety Evaluation Report (9/21/73) §6.2.3 pgs. 6-9 to 6-10 concluded that the containment isolation provision for instrument lines penetrating the containment meets the intent of General Design Criterion 56.

ITEM NO.: I-5

SUBJECT: Strong Motion Seismic Instrumentation

Response to FSAR Questions 5.13, 5.38 and 5.42 provide details of seismic instrumentation. NRC staff concluded in the Safety Evaluation Report, 9/21/73, that strong motion seismic instrumentation at IP3 correspond to the recommendations of Reg. Guide 1.12.

ITEM NO.: I-6

SUBJECT: Fuel Storage Pool Design Bases

Response to FSAR Q9.14 evaluates the Fuel Handling System design with respect to Regulatory Guide 1.13. Where the requirements of Reg. Guide 1.13 have not been incorporated in the design, adminstrative controls have been instituted (Technical Specification §3.8). The staff Safety Evaluation Report, 9/21/73, §9.1, pg. 9-7, concludes that the IP3 FHS is acceptable.

On September 1, 1977 an application was filed proposing Technical Specification amendments and requesting staff review of a proposed modification to the IP3 Spent Fuel Pool design. The staff found the modification acceptable and in conformance with the guidance of Standard Review Plan, and Reg. Guide 1.13. On March 22, 1978 NRC issued Amendment No. 13 to Facility Operating License DPR-64 authorizing modification of the pool and revising the Technical Specification.

ITEM NO.: I-7

SUBJECT: Protection of Primary System and Engineered Safety Features Against Pump Flywheel Missiles

Response to FSAR Q4.7 discusses conformance of the IP3 design to the intent of Reg. Guide 1.14. The Safety Evaluation Report (9/21/73) §3.5, pg. 3-5 concludes that measures which have been taken to provide protection against internally generated missiles are acceptable.

Inservice Inspection Program requirements, as identified in Technical Specification §4.2, Table 4.2-1 include examination for RCP flywheel integrity in conformance with Reg. Guide 1.14.

ITEM NO.: I-8

SUBJECT: Protection Against Industrial Sabotage

The NRC staff Safety Evaluation Report corresponding to Amendment No. 12 to FOL DPR-64, dated 3/8/78, concludes that the IP3 Interim Security Plan, through Revision 7, contains procedural requirements of 10 CFR 73.55. A Modified Amended Security Plan (MASP) has been submitted to the USNRC Office of Safeguards (NRR) which is in full compliance with 10 CFR 73.55. It is anticipated that approval and implementation of this plan will be made by February 23, 1979.

ITEM No.: I-9

SUBJECT: Vibration Monitoring of Reactor Internals and Primary System

NRC staff SER, (9/21/73), §3.9.1, pg. 3-10, concluded that on the basis of the applicability of the IP2 tests and conformance to Regulatory Guide 1.20, the vibration test program proposed for IP3 is acceptable on the condition that a confirmatory preoperational vibration test, in accordance with Reg. Guide 1.20 be conducted at IP3.

Extensive testing was carried out by Westinghouse and Con Ed using a neutron noise monitoring system during the preoperational startup tests of IP3. (See Westinghouse WCAP 9103; Nuclear Noise Monitoring of Indian Point Unit 3 (INT) Plant Startup).

ITEM NO.: I-10

SUBJECT: Inservice Inspection of Reactor Coolant Pressure Boundary

Technical Specification §4.2 approved by the staff indicates that the inspection program is in compliance with Section XI of the January 1970 edition of the ASME B&PVC for In-Service Inspection of Nuclear Reactor Coolant Systems.

By September 1, 1979 the Authority, in conformance with the requirements of 10 CFR 50.55a(g) is scheduled to submit an updated ISI Program for IP3. The updated program will reflect the requirements of the 1974 edition of the ASME B&PVC, Section XI, up to and including the summer 1975 Addenda, as well as requirements of Regulatory Guide 1.26. Further, any applicable portions of Regulatory Guide 1.65 will also be included.

This ISI program and related Technical Specification changes will be submitted to NRC for approval prior to implementation.

ITEM NO.: I-11

SUBJECT: Quality Assurance During Design, Construction and Operation

The March 16, 1977 submittal by the Power Authority to NRC, requesting an amendment to FOL DPR-64 authorizing the Authority to assume sole responsibility for the operation of IP3, included a QA Program for operation which was reviewed, revised and approved by the staff on March 10, 1978.

The program complies with the guidance of WASH Documents No. 1283, 1309, and 1284. The program also complies, where applicable, with the related US NRC Regulatory Guides referred in the WASH Documents or an acceptable alternative.

These regulatory guides include:

1. (1.8), Personnel Selection and Training (3/71).
2. (1.28), QA Program Requirements (Design and Construction) (6/72).
3. (1.30), QA Requirements for Installation, Inspection and Testing of Instrumentation and Electric Equipment (8/11/72).
4. (1.33), QA Program Requirements (Operation) (11/72).
5. (1.37), QA Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73).
6. (1.38), QA Requirements for Packing, Shipping, Receiving Storage and Handling of Items for Water-Cooled Nuclear Power Plants (Applicable for Operation Phase) (3/73).
7. (1.39), Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3/73).
8. (1.54), QA Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (6/73).
9. (1.58), Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel (8/73).
10. (1.64), QA Requirements for the Design of Nuclear Power Plants (10/73).

ITEM NO.: I-12

SUBJECT: Inspection of BWR Steam Lines Beyond Isolation Valves

Not Applicable

ITEM NO.: I-13

SUBJECT: Independent Check of Primary System Stress Analysis

The primary system stress analyses were performed by Westinghouse Corporation. Westinghouse has an internal policy that requires that all calculations be checked for both method and numerical accuracy by other qualified engineers within the company. This is in keeping with the requirements of the ASME III code even though the code was not adopted unitl after the IP3 primary system had been designed.

ITEM NO.: I-14

SUBJECT: Operational Stability of Jet Pumps

Not Applicable

ITEM NO.: I-15

SUBJECT: Pressure Vessel Surveillance of Fluence and NDT Shift

Compliance with the ASTM Std. E-185 is outlined in responses to FSAR Q4.3 and Q4.4. The program for accurately predicting the Nil Ducility Transition (NDT) Shift, beyond the requirements of ASME Code, is detailed in the response to FSAR Q4.2.

Technical Specification §4.2, Table 4.2-1 establishes the irradiation specimen schedule for testing in compliance with 10CFR50, App. A and H, and their references.

ITEM NO.: I-16

SUBJECT: Nil Ductility Properties of Pressure Vessel Materials

Refer to Item I-15. In addition, Technical Specification Bases for §3.18, identifies compliance with Section III of ASME B & PVC and with ASTM Std. E-185.

In Supplement 1, to NRC staff SER, dated January 16, 1975, staff concludes that the limits on pressure and temperature during heatup and cooldown given in the IP3 Technical Specification are in compliance with 10 CFR50, Appendix G.

On March 31, 1978 the Power Authority transmitted to NRC WCAP-8475, Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, dated January 1975.

ITEM NO.: I-17

SUBJECT: Operation of Reactor with Less than all Loops in Service

Operation of IP3 with less than all four loops in service is specifically disallowed by a provision of the operating license. Administrative procedures have been issued concerning this provision.

ITEM NO.: I-18

SUBJECT: Criteria for Preoperational Testing

Response to FSAR Q13.5 and Q13.7 indicate conformance of the IP3 initial test program with the requirements of applicable guides. NRC SER (9/21/72), §14.0, indicates conformance with the appropriate guides.

ITEM NO.: I-19

SUBJECT: Diesel Fuel Capacity

As indicated in Technical Specification Bases, §3.7, the minimum storage allowed will assure continued operation of two diesel generators at the minimum engineered safeguards load for a total of seven days. Capability for operating for longer periods of time is also provided.

Technical Specification Basis, §4.6, indicates that the fuel supply is monitored in accordance with the provisions of Table 4.1-3.

ITEM NO.: I-20

SUBJECT: Capability of Biological Shield to Withstand Double-Ended Pipe Break at Safe Ends

Covered by ACRS-NRC Regulatory Staff position cited in several letters that such failure should have no unacceptable consequences.

ITEM NO.: I-21

SUBJECT: Operating One Plant While Other(s) is (are) Under Construction

Not Applicable

ITEM NO.: I-22

SUBJECT: Seismic Design of Steam Lines

Response to FSAR Q4.22 indicates that the emergency conditions stress limits are applied to systems outside of the reactor coolant pressure boundary (under the load combination: normal plus DBE), including the steam lines inside the containment, up to and including the isolation valves outside the containment.

NRC SER (9/21/73), §3.9.2, pg. 3-11, concludes that stress criteria utilized provide an acceptable margin of safety for the appropriate systems and components outside of the RCPB which may be subject to seismic loading, which includes the steam lines.

ITEM NO.: I-23

SUBJECT: Quality Group Classifications for Pressure Retaining Components

The design, fabrication, erection and testing of pressure retaining components important to safety for Indian Point 3 was done in accordance with the 1968 Edition of the ASME Code, and not Regulatory Guide 1.26 (Revision 0, March 23, 1972). However, the Inservice Inspection Program Plan for the next service interval is being updated utilizing Regulatory Guide 1.26 (Revision 3, February 1976) as the basis for system classification.

ITEM NO.: I-24

SUBJECT: Ultimate Heat Sink

Response to FSAR Q9.15.2 provides an evaluation of the service water system design with respect to Regulatory Guide 1.27. NRC SER (9/21/73), §9.2.2, pg. 9-10, concludes that the station's service water system design is adequate for long-term accident cooling.

ITEM NO.: I-25

SUBJECT: Instrumentation to Detect Stress in Containment Walls

Response to FSAR Q5.40 indicates the extent to which Reg. Guide 1.18 has been followed. Appropriate surveillance actions were taken at IP3.

ITEM NO.: IA-1

SUBJECT: Use of Furnace Sensitized Stainless Steel

Response to FSAR Q4.5 describes the plans that were followed to avoid partial or local severe sensitivization of austenitic stainless steel for core structural load bearing members and component parts of the RCPB. Response to Q4.6 provides justification that some material will not be susceptible to stress corrosion cracking under severely sensitized conditions. Response to Q4.27 indicates compliance with additional criteria.

The IP3 design meets the intent of the Reg. Guide 1.44.

ITEM NO.: IA-2

SUBJECT: Primary System Detection and Location of Leaks

NRC SER (9/21/73) §5.4, pg. 5-7 concludes that the IP3 systems have the capability to detect leakage even for small through-wall flow in the RCPB. The IP3 leakage detection equipment complies with the applicable requirements of IEEE Std 279-1971 as indicated in the SER, pp 7-1 and 8-4.

ITEM NO.: IA-3

SUBJECT: Protection Against Pipe Whip

Response to FSAR Q6.10 provides a detailed description of the measures that have been used to assure that essential equipment within the containment have been adequately protected against blowdown jet forces and pipe-whip.

NRC SER (9/21/73), pg 3-5, concludes that IP3 design meets the criteria of Reg. Guide 1.46.

ITEM NO.: IA-4

SUBJECT: Anticipated Transients Without Scram

To date Westinghouse has prepared a number of generic analysis on this subject which are applicable to IP3, including the following:

WCAP 7306 & 7706	"Susceptibility of Reactor Protection Systems to Common Mode Failure"
WCAP 7486	"An Evaluation of Anticipated Operational Transients in Westinghouse PWR's"
WCAP 8330	"Westinghouse ATWT Analysis"
WCAP 8404	"ATWT for Westinghouse PWR's with Series 44 Steam Generators"

A site specific ATWS reviews or IP3 has not nor will be initiated until after the NRC publishes its final rule.

ITEM NO.: IA-5

SUBJECT: ECCS Capability of Current and Older Plants

NRC SER Supplement No. 3, dated 4/5/76, §6.3.3, pg. 6-8, concluded that the results of the ECCS analysis for IP3 were acceptable and that the analyses were performed with an acceptable evaluation model pursuant to the requirements of 10 CFR §50.46.

In April 1978, Westinghouse Electric Corporation informaed the Power Authority of a calculational error in their ECCS evaluation model. By letters to NRC, dated April 12 and April 13, 1978, the Power Authority notified NRC that no peaking factor penalties needed to be imposed on IP3 based on a revision of the October 1975 version of an approved ECCS evaluation model.

On May 19, 1978 the Power Authority submitted to the NRC results of a revised ECCS analysis in comformance with Appendix K to 10 CFR 50. This reanalysis included the correction for the zirc-water reaction modeling error of which the Authority had been informed in April 1978. No other corrections were made or any credit taken for any other model changes. The resultant total peaking factor limit was calculated to be 2.17.

An 18 case final acceptance criteria (FAC) analysis confirmed that the unit could be operated throughout Cycle 2 without any power penalty and still meet the F_q limit of 2.17.

The reanalysis submitted on May 19, 1978 was performed prior to NRC review of some additional modification in the Westinghouse ECCS evaluation model which would result in a peaking factor limit somewhat higher than 2.17. Since the 18 case FAC analysis showed that for Cycle 2 Operation a value of 2.17 value is adequate, further reanalysis with the latest model is not necessary at this time.

ITEM NO.: IB-1

SUBJECT: Positive Moderator Coefficient

Technical Specification §3.1.C.1. dictates that "...except during low power physics tests the reactor shall not be made critical at any temperature above which the moderator temperature coefficient is positive". Thus, at all times in the power operating range the moderator coefficient is negative.

Furthermore, the IP3 core is designed to maintain a negative moderator coefficient at all times. When necessary burnable poison assemblies are inserted to accomplish this purpose. The moderator coefficient is subject to administrative controls.

ITEM NO. IB-2

SUBJECT: Fixed Incore Detectors on High Power PWRs

Fixed incore detectors are not required for PWRs since reviews of potential power distribution anomalies have not revealed a clear need for continuous incore monitoring.

ITEM NO.: IB-3

SUBJECT: Performance of Critical Components in Post-LOCA Environment

Response to FSAR Q6.9 provides a reference to the design bases of RPS and ESFAS, and to the identification and discussion of safety related equipment and components and the qualificaton and testing of these components.

NRC, SER, 9/21/73, §7.8, pg. 7-8, indicated that the design criterion for safety-related equipment installed inside the containment is that the equipment shall be capable of functioning under the post-accident temperature pressure and humidity, and radiation conditions for the time periods required. The staff concluded that the type of test performed to demonstrate conformance with this design criterion are adequate and that the environmental and radiations qualification program is acceptable.

NRC IE Circular 78-08 (5/31/78) requires that IP3 confirm the environmental qualification of safety related electrical equipment to function under postualted accident conditions. An investigation is being conducted to locate documentation of the qualifications of this effected equipment. The results of the investigation will be made available, upon request, to future NRC inspectors as specified by this circular.

ITEM NO.: IB-4

SUBJECT: Vacuum Relief Valves Controlling Bypass Paths on BWR Pressure Suppresion Containments

Not Applicable

ITEM NO.: IB-5

SUBJECT: Emergency Power for Two or More Reactors at the Same Site

Not applicable since the emergency and shutdown systems of IP3 are independent of the other units on the site.

ITEM NO.: IB-6

SUBJECT: Effluents from Light-Water-Cooled Nuclear Reactors

NRC staff evaluated the radwaste systems for all three Indian Point units in accordance with the staff's design objectives proposed in Docket No. RM-50-2 which are considerably more stringent than those of the current Appendix I, and has determined that effluents from the units would meet the Commission's "as low as practicable" guidelines (§11.9 of SER dated 9/21/73).

ITEM NO.: IB-7

SUBJECT: Control Rod Ejection Accident

IP3 FSAR §14.2.6, indicates that the plant's design meets the requirements of Regulatory Guide 1.77.

ITEM NO.: IC-1

SUBJECT: Main Steam Isolation Valve Leakage of BWRs

Not Applicable

ITEM NO.: IC-2

SUBJECT: Fuel Densification

The effects of fuel densification on the operation of IP3 were originally presented in a preliminary report filed 4/2/73. NRC SER Supplement No. 3 (4/5/76), pg. 6-11, indicated that the final fuel densification report (WCAP-8146) had shown satisfactory behavior of the fuel for the original design peaking F_q, of 2.56.

As a result of a calculational error in the Westinghouse ECCS evaluation model, the Authority submitted a revised analysis on May 19, 1978, (see response to Item IA-5). This re-analysis included the effects of fuel densification in the same manner as the prior analysis.

The total peaking factor to be used during operation was determined to be 2.17.

ITEM NO.: IC-3

SUBJECT: Rod Sequence Control Systems

Not Applicable

ITEM NO.: IC-4

SUBJECT: Seismic Category I Requirement for Auxiliary Systems

The seismic design requirements for auxiliary systems for IP3 was reviewed by the NRC in the Safety Evaluation Reports dated 9/21/73, 11/16/75 and 12/12/75 and found to be acceptable.

ITEM NO.: ID-1

SUBJECT: Instruments to Detect (Limited) Fuel Failures

At IP3, fuel failures are detected through:

1. The Gross Failed Fuel Detector (GFFD), continuously monitors delayed neutron activity from fission products in the primary coolant after having allowed for the decay of N-17 neutron activity.

2. Implementation of Technical Specification §4.1 (Table 4.1-2), which requires gross activity analysis of the primary coolant on a periodic basis.

ITEM NO.: ID-2

SUBJECT: Instrumentation to Follow the Course of an Accident

NRC SER (9/21/73), §7.5, pg 7-7, concluded that the IP3 instrumentation systems that provide information to enable the operator to perform required safety functions throughout all operating conditions of the plant and to monitor the course of accidents is acceptable.

On October 13, 1978 the Power Authority committed to perform a survey to identify the capability of the existing IP3 fixed and portable instrumentation to monitor the course of accidents that extend beyond conditions assumed in the design of the plant. This survey will be completed and results submitted to the NRC by July 9, 1979.

ITEM NO.: ID-3

SUBJECT: Pressure in Containment Following LOCA

NRC SER (9/21/73), §6.2, pg 6-5, concludes that the analyses of containment pressure transients and material structures differential pressures, as reported in the FSAR Q14.6 and Q14.7 are adequate; and that design pressures of containment and of reactor cavity and steam generator compartments are acceptable.

ITEM NO.: ID-4

SUBJECT: Fire Protection

In December 1976 and in a later update in April 1977, NRC received the review of the IP3 Fire Protection Program for conformance to BTP APCSB 9.5-1, Appendix A. On June 29, 1978 the Power Authority submitted responses to additional questions from the NRC on the April 1977 submittal.

During the week of September 4, 1978 an NRC Fire Protection Review Team visited the IP3 site. On October 23, 1978 and February 6, 1979 the Power Authority submitted responses to first and second round requests for additional information on the questions/positions resulting from the Fire Protection site visit. An additional submittal is scheduled for April 16, 1979 to complete the responses to the site visit questions/ positions. On February 13, 1979, the Power Authority submitted its responses to the NRC questions of December 12, 1978 on the Administrative Controls for plant fire protection.

The NRC's review of the Indian Point 3 Fire Protection Program is continuing.

ITEM NO.: IE-1

SUBJECT: Control Rod Drop Accident (BWRs)

Not Applicable

ITEM NO.: IE-2

SUBJECT: Rupture of High Pressure Lines Outside Containment

The NRC Standard Review Plan §3.6.1 and 3.6.2 specifies that design of plants for which CP applications were tendered before July 1, 1973 and OL's were issued after July 1, 1975 should follow the guidance provided in the December 1972 letter from A. Giambusso, Deputy Director of Reactor Projects, Directorate of Licensing, to all applicants and licensees and in its attachment entitled: "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment".

NRC SER (9/21/73) §10.3, concludes that the IP3 design satisfactorily considers the consequences of postulated pipe failures outside of containment.

ITEM NO.: IE-3

SUBJECT: Isolation of Low Pressure from High Pressure Systems

NRC SER Supplement No. 1 (1/16/75), §18.1, pg. 15, discusses the isolation of low-pressure from high-pressure systems for IP3, and indicated that a procedure for testing the position of check valves identified for this purpose will be written and reviewed by NRC.

NRC SER Supplement No. 2 (12/12/75), §18.1, pg. 18-1, concludes that this matter is acceptably resolved.