ATTACHMENT I TO IPN-92-002

PROPOSED TECHNICAL SPECIFICATION CHANGES

RELATED TO

USE OF ZIRLO™ FUEL ASSEMBLIES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
DPR-64

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5.3 REACTOR

Applicability

Applies to the reactor core, and reactor coolant system.

Objective

To define those design features which are essential in providing for safe system operations.

A. Reactor Core

1. The reactor core contains approximately 87 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 or ZIRLO™ tubing to form fuel rods. The reactor core is made up of 193 fuel assemblies. Each fuel assembly contains 204 fuel rods,[1] except during Cycle 8 operation. For Cycle 8 operation only, fuel assembly T53 will contain two stainless steel filler rods in place of two fuel rods.

2. The average enrichment of the initial core was a nominal 2.8 weight percent of U-235. Three fuel enrichments were used in the initial core. The highest enrichment was a nominal 3.3 weight percent of U-235. [2]

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 4.5 weight percent of U-235.

4. Burnable poison rods were incorporated in the initial core. There were 1434 poison rods in the form of 8, 9, 12, 16, and 20-rod clusters, which are located in vacant rod cluster control guide tubes. [3] The burnable poison rods consist of borosilicate glass clad with stainless steel. [4] Burnable poison rods of an approved design may be used in reload cores for reactivity and/or power distribution control.

Amendment No. ~~61~~, ~~70~~, ~~90~~, ~~104~~,

2a. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary).
(Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

2b. T. M. Anderson to K. Kneil (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.
(Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

2c. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.
(Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

3a. WCAP-9220-P-A, Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL-1981 VERSION", February 1982 (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor.)

3b. WCAP-9561-P-A ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING W ECCS EVALUATION MODEL," July 1986 (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor.)

3c. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3d. WCAP-10054-P-A, "SMALL BREAK ECCS EVALUATION MODEL USING NOTRUMP CODE," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3e. WCAP-10079-P-A, "NOTRUMP NODAL TRANSIENT SMALL BREAK AND GENERAL NETWORK CODE," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

3f. WCAP-12610, "VANTAGE+ Fuel Assembly Report," (W Proprietary).
(Methodology for Specification 3.10.2 - Heat Flux Hot Channel Factor).

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety limits are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator-Region 1 within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification;

a. Sealed source leakage on excess of limits (Specification 3.9)

b. Inoperable Seismic Monitoring Instrumentation (Specification 4.10)

c. Seismic event analysis (Specification 4.10)

d. Inoperable plant vent sampling, main steam line radiation monitoring or effluent monitoring capability (Table 3.5-4, items 5, 6 and 7)

e. The complete results of the steam generator tube inservice inspection (Specification 4.9.C)

f. Inoperable fire protection and detection equipment (Specification 3.14)

g. Release of radioactive effluents in excess of limits (Appendix B Specifications 2.3, 2.4, 2.5, 2.6)

h. Inoperable containment high-range radiation monitors (Table 3.5-5, Item 24)

i. Radioactive environmental sampling results in excess of reporting levels (Appendix B Specification 2.7, 2.8, 2.9)

j. Operation of Overpressure Protection System (Specification 3.1.A.8.c)

k. Operation of Toxic Gas Monitoring Systems (Specification 3.3.H.3.)

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspection, repair and replacements of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all source material of record.

i. Records of reactor tests and experiments.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records of any drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g. Records of training and qualifications for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the SRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of secondary water sampling and water quality.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and records showing that these procedures were followed.

o. Records of service lives of all safety-related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION AND RESPIRATORY PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved maintained and adhered to for all operations involving personnel radiation exposure as to maintain exposures as far below the limits specified in 10 CFR Part 20 as reasonable achievable. Pursuant to 10 CFR 20.103 allowance shall be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1 of 10 CFR 20.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c) (2) of 10-CFR 20, each high radiation area in which the intensity of radiation is 1000 mrem/hr or less and 100 mrem/hr or greater shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit*. Any individual or group of individuals permitted to enter such areas shall be provided or accompanied by one or more or the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedure who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

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* Health Physics Personnel shall be exempt from the RWP issuance requirements for entries into high radiation areas during the performances of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the plant Radiological and Environmental Superintendent or his designee.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 Environmental qualification of electric equipment important to safety shall be in accordance with the provisions of 10 CFR 50.49. Pursuant to 10 CFR 50.49, Section 50.49 (d), the EQ Master List identifies electrical equipment requiring environmental qualification.

6.13.2 Complete and auditable records which describe the environmental qualification method used, for all electrical equipment identified in the EQ Master List, in sufficient detail to document the degree of compliance with the appropriate requirements of 10 CFR 50.49 shall be available and maintained at a central location. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

ATTACHMENT II TO IPN-92-002

SAFETY EVALUATION FOR TECHNICAL SPECIFICATION CHANGES

RELATED TO

USE OF ZIRLO^{TM} FUEL ASSEMBLIES

NEW YORK POWER AUTHORITY
INDIAN POINT 3 NUCLEAR POWER PLANT
DOCKET NO. 50-286
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Section I - Description of Changes

In order to use a new type of fuel cladding at IP3, the Power Authority, by this letter, is:

- proposing to revise the IP3 Technical Specifications, sections 5.3 and 6.9, to address the use of ZIRLO™, as well as Zircaloy-4, fuel rod cladding,
- requesting exemptions from the Code of Federal Regulations (CFR), specifically from 10 CFR 50.44, 10 CFR 50.46, and Appendix K to 10 CFR 50, since these regulations include specific references to fuel pellets with Zircaloy cladding, and
- providing an analysis, (required by 10 CFR 51.52 (b), because 10 CFR 51.52(a) includes a specific reference to fuel pellets encapsulated with Zircaloy) of the effects (on 10 CFR 51.52, Table S-4) of using ZIRLO™ fuel cladding.

The Technical Specification revisions will support the use of Westinghouse fuel assemblies clad with ZIRLO™ in the Indian Point 3 Cycle 9 core and future cores. (Pages 6-18 through 6-21 of the Technical Specifications are included with the proposed revision because of pagination changes.) The chemical composition of ZIRLO™ is slightly different than that of Zircaloy, and offers increased corrosion resistance.

Section II - Evaluation of Changes

Use of ZIRLO™ Fuel Cladding

Attachment III, "Safety Assessment For The Indian Point Unit 3 Fuel Assemblies With ZIRLO™ Clad Fuel Rods," describes the effects of using ZIRLO™ clad fuel rods in the Indian Point 3 core. Attachment III confirms that the fuel assemblies clad with ZIRLO™ will conform to the current fuel design bases and do not change the existing reload design and safety analysis limits.

To determine compliance with 10 CFR 50.46 for the use of ZIRLO™ cladding, the following methods were used. An IP3 specific large break LOCA analysis (single limiting break with a discharge coefficient of 0.4) was performed to determine a large break LOCA PCT. The large break LOCA PCT included the PCT results of the 1981 Westinghouse Evaluation Model with BART/BASH and all permanent and temporary PCT assessments that Westinghouse has applied to IP3. The small break LOCA evaluation was performed using a plant similar to IP3 to derive a ΔPCT for ZIRLO™ cladding, and this ΔPCT was then added to the current IP3 small break LOCA PCT. An IP3 specific small break LOCA evaluation will be performed and submitted to the NRC staff in February of 1992. The small break LOCA analysis was previously bounded by the large break analysis, and the Authority expects the new small break analysis to be bounded by the new large break analysis. For more detailed information on the evaluation methods used, see Attachment III.

The small break LOCA evaluation performed using a plant similar to IP3 resulted in a significant change in the peak cladding temperature (PCT) for the small break loss-of-coolant-accident (LOCA) analysis (ΔPCT = +73°F). The Authority recognizes the obligation to notify the NRC staff within 30 days, as required by 10 CFR 50.46(a)(3)(ii), should the plant specific small break LOCA analysis also result in a significant change in PCT.

Exemptions from 10 CFR 50.44, 50.46, and Appendix K to 10 CFR 50

Exemptions from the regulations listed are needed because those regulations include specific references to fuel pellets enclosed in Zircaloy tubes. The NRC may grant exemptions from the requirements of 10 CFR 50 when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security; and (2) when special circumstances are present. According to 10 CFR 50.12(a)(2)(ii), special circumstances are present when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule...."

The underlying purpose of 10 CFR 50.44 is to ensure an adequate means of controlling hydrogen gas following a postulated loss-of-coolant accident (LOCA). The hydrogen produced in a post-LOCA scenario comes from a metal-water reaction. The amount of hydrogen generated by the metal-water reaction in a ZIRLO™ core will be within the design basis. Since application of the regulation is not necessary to meet the underlying purpose of the rule, special circumstances exist.

The underlying purpose of 10 CFR 50.46 is to ensure that facilities have adequate acceptance criteria for emergency core cooling system (ECCS) performance. The effectiveness of the IP3 ECCS will not be affected by a change from Zircaloy to ZIRLO™ cladding. Due to similarities in the material properties of Zircaloy and ZIRLO™, the acceptability criteria for ECCS applied to reactors fueled with Zircaloy clad fuel are also applicable to the ECCS for the Indian Point 3 reactor fueled with ZIRLO™ clad fuel. Since application of the regulation is not necessary to meet the underlying purpose of the rule, special circumstances exist.

Paragraph I.A.5 of Appendix K to 10 CFR 50 states that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation. The intent of this part of the rule is to apply an equation that conservatively bounds all post-LOCA scenarios. The Baker-Just equation presumes the use of Zircaloy clad fuel, but, due to similarities in the composition of Zircaloy and ZIRLO™, applying the Baker-Just equation in the analysis of ZIRLO™ clad fuel will conservatively bound all post-LOCA scenarios. Since the Baker-Just equation presumes Zircaloy cladding, and failure to apply Baker-Just would defeat the purpose of Appendix K, paragraph I.A.5, special circumstances exist.

Analysis Required by Paragraph (b) of 10 CFR 51.52, "Environmental Effects of Transportation of Fuel and Waste - Table S-4"

10 CFR 51.52(a)(2) discusses fuel pellets encapsulated in Zircaloy rods. Using ZIRLO™ cladding necessitates (as required by 10 CFR 51.52(b)) performance of an analysis of the environmental impact of using ZIRLO™. Using ZIRLO™ cladding does not affect the conditions specified in 10 CFR 51.52, Table S-4. Listed below are the specific conditions of Table S-4, and how ZIRLO™ affects each entry.

1. Heat (per irradiated fuel cask in transit) - The use of ZIRLO™ cladding does not affect the heat transferred from an irradiated cask during transit because ZIRLO™ has essentially the same thermal/physical properties as Zircaloy-4.

2. Weight - The total core clad transit weight is not affected by the use of ZIRLO™ cladding because the density of ZIRLO™ is essentially the same as Zircaloy-4.

3. Traffic density - Using ZIRLO™ cladding does not affect traffic density because the fuel rod cladding has no relevance to the frequency of truck or rail transit.

4. Transportation workers (exposed population) - Cladding has an insignificant affect on the stated range of doses to exposed transportation workers because less than 1% of the radiation field is due to the fuel cladding. This will not change with the use of ZIRLO™ cladding material.

5. General public (exposed population) - As in item 4, this entry will not change with the use of ZIRLO™ cladding because less than 1% of the radiation field is due to the fuel cladding.

6. Radiological effects - The use of ZIRLO™ cladding does not increase the radiological effects or the environmental risk to the public because the source terms used in accident analyses are not affected by changing the cladding material.

7. Common (nonradiological) causes - The use of ZIRLO™ does not impact this entry because ZIRLO™ has essentially the same mechanical properties as Zircaloy-4, but with improved corrosion resistance and dimensional stability.

In conclusion, the use of ZIRLO™ cladding has no affect on Table S-4 of 10 CFR 51.52. Items 1, 2, 4, 5, and 6 are addressed in more detail in WCAP-12610, "VANTAGE+ Fuel Assembly Reference Core Report." The NRC reviewed WCAP-12610, and approved it on July 1, 1991.

Section III - No Significant Hazards Evaluation

Consistent with the requirements of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response:

The proposed amendment does not involve a significant increase in the probability or consequences of a previously-analyzed accident.

The VANTAGE 5 fuel assemblies containing ZIRLO™ clad fuel rods meet the same fuel assembly and fuel rod design bases as VANTAGE 5 fuel assemblies in other regions. The ZIRLO™ clad fuel rods meet the criteria of 10 CFR 50.46. Using ZIRLO™ clad fuel rods will not change the IP3 VANTAGE 5 reload design or safety analysis limits. The ZIRLO™ cladding is similar in chemical composition and has similar physical and mechanical properties as that of Zircaloy-4. Thus, the cladding and structural integrity are maintained. The ZIRLO™ cladding improves corrosion resistance and dimensional stability. The radiological consequences of accidents do not change because the dose rate predictions are not sensitive to cladding material changes.

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response:

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The fuel assemblies with ZIRLO^{TM} cladding will satisfy the same design bases as the fuel assemblies in other fuel regions, so the ZIRLO^{TM} clad rods will not initiate any new accident. Also, the use of ZIRLO^{TM} clad fuel assemblies does not involve any alteration to the plant equipment or procedures.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed amendment does not involve a significant reduction in a margin of safety.

The use of ZIRLO^{TM} clad fuel rods will not change the IP3 reload design or safety analysis limits (such as core physics peaking factors and average linear heat rate). In addition, the 10 CFR 50.46 criteria will be met for use of ZIRLO^{TM} clad fuel rods.

In the April 6, 1983 Federal Register, Vol. 048, No. 67, Page 14870, the NRC published a list of examples of amendments that are not likely to involve a significant hazards concern. Example (iii) of that list applies to the use of ZIRLO^{TM} clad fuel assemblies and states:

> For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that NRC has previously found such methods acceptable.

Section IV - Impact of Changes

This change will not adversely impact the following:

ALARA Program
Security and Fire Protection Programs
Emergency Plan
FSAR or SER Conclusions
Overall Plant Operations and the Environment

Section V - Conclusions

The incorporation of this change: a) will not increase the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the bases for any Technical Specification; d) does not constitute an unreviewed safety question; and e) involves no significant hazards considerations as defined in 10 CFR 50.92.

Section VI - References

a) IP-3 FSAR
b) IP-3 SER
c) WCAP-12610, "VANTAGE+ Fuel Assembly Reference Core Report."

ATTACHMENT III TO IPN-92-002

WESTINGHOUSE SAFETY ASSESSMENT FOR THE INDIAN POINT UNIT 3 FUEL ASSEMBLIES WITH ZIRLO^{TM} CLAD FUEL RODS

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