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New York Power Authority

John C. Brons
Executive Vice President
Nuclear Generation

JPN-90-051
IPN-90-036
July 3 , 1990

U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Status of Implementation of Generic Safety Issues

Reference: Generic Letter 90-04, dated April 25, 1990, "Request for Information on the Status of Licensee Implementation of Generic Safety Issues Resolved with Imposition of Requirements or Corrective Actions"

Dear Sir:

Generic Letter 90-04 requested that the Authority review and provide documentation of the current implementation status of generic safety issues (GSIs) applicable to the Authority's two nuclear power plants. Attachments I and II provides the status of these GSIs for the James A. FitzPatrick and Indian Point 3 Nuclear Power Plants respectively.

If you should have any questions regarding this information, please contact Mr. J. Ellmers or Mr. P. Kokolakis of my staff.

Very truly yours,

John C. Brons
John C. Brons
Executive Vice President
Nuclear Generation

cc: see next page

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cc: U. S. Nuclear Regulatory Commission
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Attachment I

Generic Safety Issues (GSIs) Licensing Status Mark-up

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

Docket No. 50-333

FACILITY NAME: James A. FitzPatrick Nuclear Power Plant
DOCKET NO.: 50-333
LICENSEE: New York Power Authority

STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES

RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**	
40 (B065)	Safety Concerns Associated With Pipe Breaks In The BWR Scram System	All BWRs	C	JPN-82-010 JPN-83-039	1-15-82 5-3-83
41 (B058)	BWR Scram Discharge Volume Systems	All BWRs	C	JPN-83-063	7-7-83
43 (B107)	Reliability Of Air Systems	All Plants	I	See note 1	
51 (L913)	Improving the Reliability of Open-Cycle Service Water Systems	All Plants	I	See note 2	
67.3.3 (A017)	Improved Accident Monitoring	All Plants	C	See note 3	
75** (B076)	Item 1.1 - Post-Trip Review (Program Description and Procedure)	All Plants	C	JPN-83-092 JPN-84-042	11-9-83 6-29-84
75 (B085)	Item 1.2 - Post-Trip Review - Data and Information Capability	All Plants	C	JPN-84-042 JPN-87-041	6-29-84 8-4-87

* Status codes are defined by the NRC Attachment to Enclosure 1 to Generic Letter 90-04 as follows: C - Completed, date of completion; I - Incomplete, expected date of completion; NA - Not applicable; E - Response to Generic Letter not completed, expected response date; NC - No changes were necessary.

** The letters listed in the comments column identify correspondence from the Power Authority to the NRC. These letters do not certify completion, but provide detailed information which in most cases was the basis for an associated NRC Safety Evaluation Report.

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS **
75 (B077)	Item 2.1 - Equipment Classification and Vendor Interface (Reactor Trip System Components)	All Plants	C	JPN-84-042 6-29-84 JPN-85-055 7-2-85
75 (B086)	Item 2.2.1 - Equipment Classification for Safety-Related Components	All Plants	C	See note 4
75 (L003)	Item 2.2.2 - Vendor Interface for Safety-Related Components	All Plants	E	9-26-90
75 (B078)	Items 3.1.1 & 3.1.2 - Post - Maintenance Testing (Reactor Trip System Components)	All Plants	C	JPN-84-042 6-29-84
75 (B079)	Item 3.1.3 - Post-Maintenance Testing-Changes to Test Requirements (Reactor Trip System Components)	All Plants	C	JPN-84-042 6-29-84
75 (B087)	Items 3.2.1 & 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components)	All Plants	C	JPN-84-042 6-29-84
75 (B088)	Item 3.2.3 - Post-Maintenance Testing-Changes to Test Requirements (All Other Safety-Related Components)	All Plants	C	JPN-84-042 6-29-84
75 (B080)	Item 4.1 - Reactor Trip System Reliability (Vendor-Related Modifications)	All Plants	NA	

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
75 (B081)	Items 4.2.1 & 4.2.2 - Reactor Trip System Reliability- Maintenance and Testing (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)	All PWRs	NA	
75 (B082)	Item 4.3 - Reactor Trip System Reliability - Design Modifications (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)	All W and B&W Plants	NA	
75 (B090)	Item 4.3 - Reactor Trip System Reliability - Tech Spec Changes (Automatic Actuation of Shunt Trip Attachment For Westinghouse and B&W Plants)	All W & B&W Plants	NA	
75 (B091)	Item 4.4 - Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)	All B&W Plants	NA	

GSI/(MPA No.)	TITLE	APPLICABILITY		STATUS*	COMMENTS**	
75 (B092)	Item 4.5.1 - Reactor Trip System Reliability-Diverse Trip Features (System Functional Testing)	All Plants		NC	See note 5	
75 (B093)	Items 4.5.2 & 4.5.3 - Reactor Trip System Reliability - Test Alternatives and Intervals (System Functional Testing)	All Plants	4.5.2	C	JPN-84-042	6-29-84
			4.5.3	C	JPN-89-067	10-23-89
86 (B084)	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	All BWRs		C	JPN-88-041	8-16-88
93 (B098)	Steam Binding of Auxiliary Feedwater Pumps	All PWRs		NA		
99 (L817)	RCS/RHR Suction Line Valve Interlock on PWRs	All PWRs		NA		
124	Auxiliary Feedwater System Reliability	ANO-1&2, Rancho Seco, Prairie Island 1&2, Crystal River-3 Ft. Calhoun		NA		
A-13 (B017)	Snubber Operability Assurance - Hydraulic Snubbers	All Plants		C	JPN-84-051	7-27-84

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**	
A-13 (B022)	Snubber Operability Assurance - Mechanical Snubbers	All Plants	C	JPN-84-051	7-27-84
A-16 (D012)	Steam Effects on BWR Core Spray Distribution	Oyster Creek & NMP-1	NA		
A-35 (B023)	Adequacy of Offsite Power Systems	All Plants	C	JPN-82-054	6-22-82
B-10	Behavior of BWR Mark III Containments	All BWR Mark III Plants	NA		
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems	All Plants with OL Applications After 4/1/80	NA		
B-63 (B045)	Isolation of Low Pressure Systems Connected to the Reactor Coolant System Pressure Boundary	All Plants	NA		

Notes:

1. The Authority has completed a review requested by Generic Letter 88-14, "Instrument Air Supply Systems Problems Affecting Safety-Related Equipment," and forwarded a response to the NRC by letters JPN-89-007, dated February 17, 1989 and JPN-89-061, dated September 21, 1989. Completion of the GSI is scheduled for the 1991 FitzPatrick plant maintenance outage.

2. The Authority's response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," was completed and forwarded to the NRC by letter JPN-90-015, dated February 13, 1990. The Authority's Implementation Plan (Attachment I to JPN-90-015) has a projected completion date of January 31, 1992.

3. The complete response to the Generic Safety Issue concerning Improved Accident Monitoring (Regulatory Guide 1.97, Revision 2) consists of six letters sent to the NRC:

   JPN-84-077, November 30, 1984
   JPN-85-053, June 28, 1985
   JPN-85-091, December 24, 1985
   JPN-86-006, February 25, 1986
   JPN-87-033, June 9, 1987
   JPN-87-055, November 11,1987

4. The complete response to Generic Letter 83-28,"Required Actions Based on Generic Implications of Salem ATWS Events," Item 2.2.1 - Equipment Classification for Safety-Related Components, consists of six letters sent to the NRC:

   JPN-83-092, November 9, 1983
   JPN-84-042, June 29, 1984
   JPN-85-055, July 2, 1985
   JPN-85-093, December 31,1985
   JPN-87-015, March 20, 1987
   JPN-89-066, October 16, 1980

5. The Authority's response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," Item 4.5.1 - Reactor Trip System Reliability-Diverse Trip Features (System Functional Testing), was sent to the NRC in NYPA letter JPN-84-042, dated June 29, 1984. The Authority did not agree with this item of the Generic Letter and forwarded justifications in its June 29, 1984 response. The justification was found acceptable by the NRC and approved in NRC letter and SER, dated August 6, 1986.

Attachment II

Generic Safety Issues (GSIs) Licensing Status Mark-up

New York Power Authority

INDIAN POINT 3 NUCLEAR POWER PLANT

Docket No. 50-286

FACILITY NAME: Indian Point 3 Nuclear Power Plant
DOCKET NO.: 50-286
LICENSEE: New York Power Authority

STATUS OF LICENSEE IMPLEMENTATION OF GENERIC SAFETY ISSUES

RESOLVED WITH IMPOSITION OF REQUIREMENTS OR CORRECTIVE ACTIONS

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
40 (B065)	Safety Concerns Associated With Pipe Breaks In The BWR Scram System	All BWRs	N/A	
41 (B058)	BWR Scram Discharge Volume Systems	All BWRs	N/A	
43 (B107)	Reliability Of Air Systems	All Plants	I	See Note 1
51 (L913)	Improving the Reliability of Open-Cycle Service Water Systems	All Plants	C	IPN-90-004 2-6-90
67.3.3 (A017)	Improved Accident Monitoring	All Plants	C	IPN-84-020 6-29-84 IPN-86-005 1-7-86
75 (B076)	Item 1.1 - Post-Trip Review (Program Description and Procedure)	All Plants	C	IPN-83-091 11-7-83 IPN-85-027 6-5-85
75 (B085)	Item 1.2 - Post-Trip Review - Data and Information Capability	All Plants	C	IPN-83-091 11-7-83

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** The letters listed in the comments column identify correspondence from the Power Authority to the NRC. These letters do not certify completion, but provide detailed information which in most cases was the basis for an associated NRC Safety Evaluation Report.

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
75 (B077)	Item 2.1 - Equipment Classification and Vendor Interface (Reactor Trip System Components)	All Plants	C	IPN-83-091 11-7-83 IPN-84-022 7-3-84 IPN-86-040 8-25-86
75 (B086)	Item 2.2.1 - Equipment Classification for Safety-Related Components	All Plants	C	IPN-83-091 11-7-83 IPN-85-026 5-17-85 IPN-86-040 8-25-86
75 (L003)	Item 2.2.2 - Vendor Interface for Safety-Related Components	All Plants	E	9-26-90
75 (B078)	Items 3.1.1 & 3.1.2 - Post - Maintenance Testing (Reactor Trip System Components)	All Plants	C	IPN-83-091 11-7-83 IPN-84-022 7-3-84
75 (B079)	Item 3.1.3 - Post-Maintenance Testing-Changes to Test Requirements (Reactor Trip System Components)	All Plants	C	IPN-85-026 5-17-85
75 (B087)	Items 3.2.1 & 3.2.2 - Post-Maintenance Testing (All Other Safety-Related Components)	All Plants	C	IPN-83-091 11-7-83 IPN-84-022 7-3-84
75 (B088)	Item 3.2.3 - Post-Maintenance Testing-Changes to Test Requirements (All Other Safety-Related Components)	All Plants	C	IPN-85-026 5-17-85
75 (B080)	Item 4.1 - Reactor Trip System Reliability (Vendor-Related Modifications)	All Plants	C	IPN-83-091 11-7-83

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
75 (B081)	Items 4.2.1 & 4.2.2 - Reactor Trip System Reliability-Maintenance and Testing (Preventative Maintenance and Surveillance Program for Reactor Trip Breakers)	All PWRs	C	IPN-83-091 11-7-83 IPN-84-065 12-28-84
75 (B082)	Item 4.3 - Reactor Trip System Reliability - Design Modifications (Automatic Actuation of Shunt Trip Attachment for Westinghouse and B&W Plants)	All W and B&W Plants	C	IPN-83-091 11-7-83 IPN-85-018 4-16-85
75 (B090)	Item 4.3 - Reactor Trip System Reliability - Tech Spec Changes (Automatic Actuation of Shunt Trip Attachment For Westinghouse and B&W Plants)	All W & B&W Plants	C	IPN-87-009 3-19-87
75 (B091)	Item 4.4 - Reactor Trip System Reliability (Improvements in Maintenance and Test Procedures for B&W Plants)	All B&W Plants	N/A	

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
75 (B092)	Item 4.5.1 - Reactor Trip System Reliability-Diverse Trip Features (System Functional Testing)	All Plants	C	IPN-83-091 11-7-83
75 (B093)	Items 4.5.2 & 4.5.3 - Reactor Trip System Reliability - Test Alternatives and Intervals (System Functional Testing)	All Plants	C	IPN-83-091 11-7-83 IPN-85-026 5-17-85
86 (B084)	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	All BWRs	N/A	
93 (B098)	Steam Binding of Auxiliary Feedwater Pumps	All PWRs	C	IP3-WAJ-010Z 2-18-86 IPN-88-012 4-18-88
99 (L817)	RCS/RHR Suction Line Valve Interlock on PWRs	All PWRs	I	See Note 2
124	Auxiliary Feedwater System Reliability	ANO-1&2, Rancho Seco, Prairie Island 1&2, Crystal River-3 Ft. Calhoun	N/A	
A-13 (B017)	Snubber Operability Assurance - Hydraulic Snubbers	All Plants	C	IPN-81-071 9-25-81 IPN-81-081 11-21-81 IPN-81-095 11-24-81 IPN-83-086 10-7-83 IPN-84-009 3-12-84 IPN-84-030 8-13-84 IPN-87-060 12-21-87 IPN-88-039 8-23-88

GSI/(MPA No.)	TITLE	APPLICABILITY	STATUS*	COMMENTS**
A-13 (B022)	Snubber Operability Assurance - Mechanical Snubbers	All Plants	C	IPN-84-020 8-13-84
A-16 (D012)	Steam Effects on BWR Core Spray Distribution	Oyster Creek & NMP-1	N/A	
A-35 (B023)	Adequacy of Offsite Power Systems	All Plants		
B-10	Behavior of BWR Mark III Containments	All BWR Mark III Plants	N/A	
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems	All Plants with OL Applications After 4/1/80	N/A	
B-63 (B045)	Isolation of Low Pressure Systems Connected to the Reactor Coolant System Pressure Boundary	All Plants	C	IPN-80-031 3-13-80

Notes:

1. In letter IPN-89-012, dated February 17, 1989, the Authority provided a 180-day response to Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment." This letter stated that the Authority plans to complete the requested verification, including any testing determined necessary, prior to startup from the Cycle 7/8 refueling outage.

2. In letters IPN-89-001 and IPN-89-008, dated January 3, 1989 and February 7, 1989, respectively, the Authority responded to Generic Letter 88-17, "Loss of Decay Heat Removal." IPN-89-008 stated that programmed enhancements requiring hardware changes or significant plant testing will be implemented prior to start-up following the Cycle 7/8 refueling outage.