

May 23, 1989

Docket No. 50-247

LICENSEE: Consolidated Edison Company of New York, Inc.

FACILITY: Indian Point Nuclear Generating Unit No. 2

SUBJECT: MEETING MINUTES REGARDING MAY 16, 1989 MEETING TO DISCUSS
PROPOSED LICENSE AMENDMENT FOR AUTHORIZING AN INCREASE IN
LICENSED POWER FOR INDIAN POINT UNIT 2 (TAC NO. 69542)

A meeting was held in the NRC One White Flint offices in Rockville, Maryland, with Consolidated Edison and NRC staff representatives to discuss the licensee's proposal for increasing the licensed thermal power of Indian Point Unit 2 from 2758 MWT to 3071.4 MWT. The meeting was requested by the licensee. Enclosure 1 is a list of the meeting attendees.

The licensee submitted a request for the proposed license amendment on September 30, 1988. Supplemental information supporting this amendment request was submitted on January 10, 1989, March 30, 1989, and April 14, 1989. Enclosure 2 is a copy of the licensee's handout that was discussed at the meeting. Staff representatives asked several questions regarding the submittals. The licensee provided clarification for the staff concerns. The staff will continue its review of the proposed license amendment.

Original signed by

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects I/II

Enclosures:
As stated

cc: See next page

[5/16/89 IP-2 MTG SUMMARY]

OFC	:PDI-1	:PDI-1	:PDI-1	:	:	:	:
NAME	:CVogan	:DBrinkman	:vr:RCapra	:	:	:	:
DATE	:5/23/89	:5/23/89	:5/23/89	:	:	:	:

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DISTRIBUTION FOR MEETING SUMMARY DATED: May 23, 1989

Docket file

NRC PDR

Local PDR

JSniezek

PDI-1 Rdg

RCapra

DBrinkman

OGC

EJordan

BGrimes

ACRS (10)

H. B. Clayton

PSwetland, RI

CVogan

AToalston

CLiang

MHum

CSellers

JMinns

SRhow

JGuo

ALee

MChatterton

Mr. Stephen B. Bram
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station 1/2

cc:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, New York 10511

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Mr. Jude Del Percio
Manager of Regulatory Affairs
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, New York 10511

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Post Office Box 38
Buchanan, New York 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Mr. Peter Kokolakis, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, New York 10003

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

MEETING ATTENDEES

May 16, 1989 meeting to discuss Licensee's Proposal to Increase Licensed Power for Indian Point 2.

<u>NAME</u>	<u>TITLE</u>	<u>ORGANIZATION</u>
D. S. Brinkman	Senior Project Manager	NRC/NRR
A. Toalston	Electrical Engineer	NRC/NRR
Chu-yu Liang	Sr. Nuclear Engineer	NRC/NRR/SRXB
Martin Hum	Materials Engineer	NRC/NRR/EMTB
C. D. Sellers	Sr. Materials Engineer	NRC/NRR/EMTB
Steve Kline	Sr. License Engineer	Bechtel Serch
Alex Ball Jr.	Sr. License Engineer	Westinghouse
		Nuclear Safety
B. Samardzich	Senior Engineer	Westinghouse
		Nuclear Fuels
Louis Liberatori	Mgr. Safety Assessment	Con Edison
Roy F. Kim, Jr.	Senior Engineer	Westinghouse
		Sys. Engr.
M. P. Osborne	Mgr. Transient Analysis	Westinghouse
		Nuclear Safety
Gary G. Ament	Engineer	Westinghouse
		Nuclear Safety
J. J. Akers	Engineer	Westinghouse
		Nuclear Safety
J L. Minns	Health Physicist	NRC/DRPEP/NRR
R. H. McFetridge	Mgr. Systems Integration	Westinghouse
		Sys. Engr.
Getachew Tesfaye	Engineer	Con Edison
Paul Malik	Manager	Con Edison
Sang Rhow		NRC/ICSB/NRR
J. S. Guo	Reactor Systems Engineer	NRC/SPLB/NRR
Arnold Lee	Mechanical Engineer	NRC/EMEB/NRR
Jude Del Percio	Mgr. Regulatory Affairs & Safety Assess.	Con Edison
M. Chatterton	Nuclear Engineer	NRC/SRXB/NRR

INDIAN POINT UNIT 2
STRETCH RATING
PRESENTATION
TO
NRC

MAY 16, 1989

AGENDA

INTRODUCTION & OVERVIEW

J. DELPERCIO

STRETCH PROGRAM RESULTS

P. MALIK

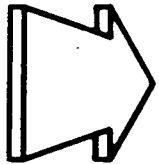
STRETCH PROGRAM SUBMITTALS

G. TESFAYE

CLOSING

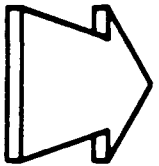
J. DELPERCIO

INDIAN POINT UNIT 2 STRETCH RATING *OVERVIEW*



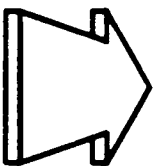
PROVIDES NUMEROUS BENEFITS

- 3083.4 MWt
- IMPROVE OPERATING FLEXIBILITY
- INTEGRATED WITH OPTIMIZED FUEL PROGRAM
- INCORPORATES 25% TUBE PLUGGING MARGIN



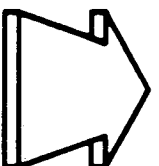
THE PROGRAM EVALUATED ESSENTIALLY ALL FACETS OF THE NSSS & BOP

- | | |
|-------------|--------------|
| ● SYSTEMS | ● COMPONENTS |
| ● ACCIDENTS | ● OPERATIONS |



REVIEW PROCESS IN PROGRESS

- LIMITING ANALYSES SUBMITTED SEPT. 30, 1988
- FINAL ANALYSIS AND DOCUMENTATION
SUBMITTED MARCH 30, 1989
- OPERATE AT 3083.4 (12/25/89)



NRC - CON EDISON COOPERATION
IS ESSENTIAL

INDIAN POINT UNIT 2 STRETCH RATING PROGRAM

MAIN FEATURES AND CONSERVATISMS:

- ~ 30° TEMPERATURE RANGE
- REDUCED THERMAL DESIGN FLOW (90%)
- ANALYSES JUSTIFY 3083.4 MWt NSSS POWER
- 25% S/G TUBE PLUGGING
- CORE PEAKING FACTORS
 - $F_Q = 2.32$ (SAME AS PREVIOUS)
 - $F_{\Delta H} = 1.62$ (INCREASED FROM 1.55)
- UPGRADE ANALYSES TO 1988 TECHNOLOGY
 - ITDP/WRB-1
 - BASH
 - NOTRUMP
- CONTAINMENT LOCA ANALYSIS/OFFSITE DOSE AT 3216 MWt

INDIAN POINT UNIT 2 PARAMETERS

STRETCH RATING RANGE
 STRETCH RATING RANGE
 CURRENT OPERATION
 DESIGN BASIS

NSSS POWER (MWt)	2770	2770	3083.4	3083.4
REACTOR FLOW (GPM/LOOP)	89,700	89,700	80,700	80,700
T _{COLD} (°F)	543.0	521.8	515.8	547.7
T _{AVG} (°F)	569.5	549.0	549.0	579.7
T _{HOT} (°F)	596.0	578.8	582.2	611.7
T No Load (°F)	547.0	547.0	547.0	547.0
ΔT (°F)	53.0	57.0	66.4	64.0
PRIMARY SIDE PRESSURE (PSIA)	2250	2250	2250	2250
STEAM PRESSURE (PSIA)	776	700	650	768
STEAM FLOW (LB.M/HR. x 10 ⁶)	11.66	11.66	13.25	13.31
ASSUMED S/G TUBE PLUGGING (%)	0	7	25	25

INDIAN POINT UNIT 2 STRETCH RATING PROGRAM

NSSS SAFETY AND OPERATIONS CONSIDERATIONS

- PRIMARY AND SECONDARY PLANT PARAMETERS
- LOSS OF COOLANT ANALYSIS
- NON-LOCA ANALYSES
- STEAM GENERATOR TUBE RUPTURE
- FUEL DESIGN
- MASS AND ENERGY RELEASES
- LOCA HYDRAULIC FORCING FUNCTIONS
- TECHNICAL SPECIFICATIONS
- CONTROL AND PROTECTION SYSTEM SETPOINTS
AND ALARMS
- CONTROL SYSTEMS OPERATING MARGINS
- PLANT PROCEDURES
- OPERATOR TRAINING

INDIAN POINT UNIT 2

STRETCH RATING ANALYSES - FUEL

- FUEL FEATURES

- 15X15 OPTIMIZED FUEL (ZIRC MIXING VANE GRIDS)
- EXTENDED BURNUP FEATURES
- THIMBLE PLUG DELETION

SUBMITTED SEPTEMBER 30, 1988, "APPLICATION FOR AMENDMENT TO LICENSE TO INCORPORATE WESTINGHOUSE OPTIMIZED FUEL ASSEMBLIES (OFA)", AS SUPPLEMENTED BY LETTER DATED DECEMBER 30, 1989 AND CLARIFIED BY LETTERS DATED JANUARY 20, FEBRUARY 7, MARCH 3, AND APRIL 14, 1989.

- WORK TO SUPPORT FUEL DESIGN CHANGE PERFORMED @
STRETCH RATED CONDITIONS

- COVERS BOTH EXISTING AND NEW FUEL
- USE IMPROVED THERMAL DESIGN PROCEDURE FOR
DNB EVALUATIONS
- PAD 3.4 FOR FUEL ROD DESIGN (WCAP-11873-A)
- 3071.4 MWt

Indian Point Unit 2

FSAR Licensing Basis Non-LOCA Accident Analyses for Stretch Rating

<u>Accident</u>	<u>Analysis</u>	<u>Acceptance Criteria</u>
A. Uncontrolled Control Rod Withdrawal from a Subcritical Condition	OFA	DNB limit (non-ITDP)
B. Uncontrolled Rod Cluster Control Assembly Withdrawal at Power	OFA/Stretch	DNB limit (ITDP)
C. Rod Cluster Control Assembly Drop	OFA	DNB limit (ITDP)
D. Chemical and Volume Control System Malfunction	OFA	Operator action time - Refueling: 30 min. - Startup: 15 min. - At Power: 15 min.
E. Loss of Reactor Coolant Flow / Locked Rotor	OFA/Stretch	DNB limit (ITDP) RCS pressure % Fuel rods in DNB Clad temperature
F. Startup of an Inactive Reactor Coolant Loop	N/A	
G. Loss of External Electrical Load	Stretch	DNB limit (ITDP) RCS pressure
H. Loss of Normal Feedwater	Stretch	No water relief from pressurizer
I. Reduction in Feedwater Enthalpy	Stretch	DNB limit (ITDP)
J. Excessive Load Increase Incident	Stretch	DNB limit (ITDP)
K. Loss of All AC Power to the Station Auxiliaries	Stretch	No water relief from pressurizer
L. Rupture of a Steam Pipe (Core Response) (M&E release to Containment)	Stretch	DNB limit (non-ITDP) Containment press.
M. Rupture of a Control Rod Drive Mechanism Housing	OFA	Clad temp., % fuel pellet melt, fuel pellet enthalpy

LARGE BREAK LOCA ANALYSIS WITH BASH

CORE POWER: 3071.4 MW_T

STEAM GENERATOR TUBE PLUGGING: 25%

RANGE OF OPERATING TEMPERATURES:

$$579.7^{\circ}\text{F} \leq T_{\text{AVG}} \leq 549^{\circ}\text{F}$$

MODEL 44 OR 44F SG

15X15 OFA FUEL

90% THERMAL DESIGN FLOW

$$F_Q = 2.32$$

$$F\text{-DELTA-H} = 1.62$$

RESULTANT PEAK CLAD TEMPERATURE: 2039°F

SMALL BREAK LOCA ANALYSIS WITH NOTRUMP

CORE POWER: 3071.4 MW_T

STEAM GENERATOR TUBE PLUGGING: 25%

RANGE OF OPERATING TEMPERATURES:

$$579.7^{\circ}\text{F} \leq T_{\text{AVG}} \leq 549^{\circ}\text{F}$$

MODEL 44 OR 44F SG

15X15 OFA FUEL

90% THERMAL DESIGN FLOW

$$F_Q = 2.32$$

$$F\text{-DELTA-H} = 1.65$$

RESULTANT PEAK CLAD TEMPERATURE: 1218.5°F

INDIAN POINT UNIT 2 STRETCH RATING PROGRAM

NSSS SYSTEMS CONSIDERATIONS

- PRESSURIZER & S/G SAFETY VALVE SYSTEM
- CHEMICAL & VOLUME CONTROL SYSTEM
- RESIDUAL HEAT REMOVAL SYSTEM
- EMERGENCY CORE COOLING SYSTEM
- CONTAINMENT COOLING SYSTEM
- SERVICE WATER AND COMPONENT COOLING SYSTEM

INDIAN POINT UNIT 2 STRETCH RATING PROGRAM

NSSS EQUIPMENT DESIGN CONSIDERATIONS

- COMPONENT DESIGN TRANSIENTS
- REACTOR VESSEL
- REACTOR VESSEL INTERNALS
- CONTROL ROD DRIVE MECHANISMS
- STEAM GENERATORS
- PRESSURIZER
- REACTOR COOLANT PUMP
- REACTOR COOLANT SYSTEM PIPING
- AUXILIARY EQUIPMENT

INDIAN POINT UNIT 2 STRETCH RATING PROGRAM

BOP CONSIDERATIONS

- PLANT DESIGN FOR 3216
- BOP SYSTEM OPTIMIZED AND GUARANTEED
AT 3083.4 MWt
- FI ROTOR
- NEW ELECTRICAL GENERATOR
- ALL BOP SYSTEMS WERE REVIEWED
 - FEEDWATER SYSTEM
 - STEAM SYSTEM (INCLUDES SAFETY SYSTEM)
 - CONDENSATE SYSTEM

RERATINGS ARE NOT
NEW TO THE NRC

PLANT	OPERATING FLEXIBILITY	POWER LEVEL	T _{HOT} REDUCTION	TUBE PLUGGING	FUEL CHANGE
NORTH ANNA		★		★	
SALEM 1		★			
CALLAWAY	★	★		★	★
BYRON/BRAIDWOOD			★	★	
DONALD C. COOK	★	★	★	★	
VIRGIL C. SUMMER	★		★	★	★
INDIAN POINT 2	★	★	★	★	★

CONSOLIDATED EDISON
INDIAN POINT UNIT 2
NSSS STRETCH RATING - 3083.4 MWT
LICENSING REPORT

SECTION	TITLE
1.0	INTRODUCTION
2.0	COMPARISON OF PARAMETERS
3.0	ACCIDENT ANALYSES
3.1	NON-LOCA EVENTS
3.2	LOCA EVENTS
3.3	HOT LEG SWITCHOVER
3.4	HYDRAULIC FORCES
3.5	LONG TERM LOCA CONTAINMENT INTEGRITY ANALYSIS
3.6	SHORT TERM SUBCOMPARTMENT ANALYSIS
3.7	MAIN STEAMLINE BREAK CONTAINMENT INTEGRITY ANALYSIS
3.8	STEAM GENERATOR TUBE RUPTURE
3.9	RADIATION SOURCE TERMS
3.10	FUEL HANDLING ACCIDENT
4.0	NSSS COMPONENTS IMPACT
4.1	BASIS FOR EVALUATION
4.2	EQUIPMENT REVIEWS
4.2.1	Reactor Vessel
4.2.2	Reactor Internals
4.2.3	Reactor Coolant Pumps
4.2.4	Control Rod Drive Mechanisms
4.2.5	Reactor Coolant Piping & Supports
4.2.6	Pressurizer
4.2.7	Steam Generators
4.2.8	Nuclear Fuel
4.2.9	Auxiliary Systems Components
4.3	CONCLUSIONS

CONSOLIDATED EDISON
INDIAN POINT UNIT 2
NSSS STRETCH RATING - 3083.4 MWT
LICENSING REPORT

SECTION TITLE

5.0 NSSS SYSTEMS REVIEW

5.1 FLUID SYSTEMS

- 5.1.1 Reactor Coolant System
- 5.1.2 Chemical and Volume Control System
- 5.1.3 Safety Injection System
- 5.1.4 Containment Spray System
- 5.1.5 Auxiliary Coolant System
- 5.1.6 Waste Disposal System
- 5.1.7 Sampling System
- 5.1.8 Isolation Valve Seal Water System
- 5.1.9 Electrical Power Requirements

5.2 REACTOR CONTROL SYSTEMS

- 5.2.1 Turbine Trip Without Reactor Trip

5.3 REACTOR PROTECTION SYSTEMS

5.4 OTHER SYSTEMS

5.5 CONCLUSIONS

6.0 NSSS/BOP INTERFACES

6.1 INTRODUCTION

6.2 AUXILIARY FEEDWATER SYSTEM

6.3 COMPONENT COOLING WATER INTERFACE REQUIREMENTS

6.4 MAIN STEAM SYSTEM

6.5 MAIN FEEDWATER SYSTEM

CONSOLIDATED EDISON
INDIAN POINT UNIT 2
NSSS STRETCH RATING - 3083.4 MWT
LICENSING REPORT

SECTION TITLE

7.0 REFERENCES

APPENDIX 1 NON-LOCA ANALYSIS RESULTS

A1.1	LOSS OF NORMAL FEEDWATER
A1.2	LOSS OF ALL AC POWER TO THE STATION AUXILIARIES
A1.3	LOSS OF EXTERNAL ELECTRICAL LOAD
A1.4	EXCESSIVE LOAD INCREASE
A1.5	RUPTURE OF A STEAM PIPE
A1.6	EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS
A1.7	UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER
A1.8	LOSS OF REACTOR COOLANT FLOW
A1.9	LOCKED ROTOR
A1.10	REFERENCES

APPENDIX 2 MAIN STEAM LINE BREAK CONTAINMENT INTEGRITY
ANALYSIS RESULTS

APPENDIX 3 LONG TERM LOCA CONTAINMENT INTEGRITY ANALYSIS
RESULTS

NRC SUPPORT FOR REVIEW/APPROVAL

OF INDIAN POINT UNIT 2 STRETCH

RATING IS ESSENTIAL

LIMITING CONDITIONS SUBMITTAL	9/30/88
FINAL LICENSE AMENDMENT SUBMITTAL	3/30/89
CYCLE 10 START-UP	6/2/89
NRC APPROVAL OF STRETCH RATING	10/31/89
PLANT OPERATION AT STRETCH	12/25/89