

April 11, 1989

Docket Nos. 50-334  
and 50-412

Mr. J. D. Sieber, Vice President  
Nuclear Group  
Duquesne Light Company  
Post Office Box 4  
Shippingport, Pennsylvania 15077

Dear Mr. Sieber:

SUBJECT: BEAVER VALLEY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS  
(TAC NOS. 71829 AND 71830)

The Commission has issued the enclosed Amendment No. 139 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Units 1, and Amendment No. 14 to Facility Operating License No. NPF-73 for Unit 2, in response to your application dated January 5, 1989.

The amendments revise the Technical Specifications in the following way: (1) update the index, (2) correct Section 4.4.5.2, steam generator inspection requirements to be consistent with Section 4.0.5, (3) correct reporting requirement of Section 6.9.1 to be consistent with 10 CFR 50.4(b)(1), and (4) modify Basis Section 3/4.2.2 and 3/4.2.3 to reflect updated approved requirements.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

Peter S. Tam, Senior Project Manager  
Project Directorate I-4  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 139 to DPR-66
2. Amendment No. 14 to NPF-73
3. Safety Evaluation

cc w/enclosures:  
See next page

[5520 Document Name: TAC 71829/71830]

*OK w/ modifications*  
*↑*  
*Safety evaluation has been modified as requested by OGC.*  
*PSTam 4/11/89*

*C/P/*  
*et*

OFC : LA: PDI-4	: PM: PDI-4	: PD: PDI-4	: OGC : J. Moore:	:	:
NAME : S. Norris	: PTam: lm	: JStolz	: PSTam	:	:
DATE : 03/28/89	: 03/30/89	: 04/07/89	: 04/11/89	:	:

OFFICIAL RECORD COPY

8904250305 890411  
FDR ADDCK 05000334  
PDC

DFOL

DATED: April 11, 1989

AMENDMENT NOS. 139 & 14 TO FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

Docket File

NRC PDR

Local PDR

Plant File

S. Lainas (14E4)

B. Boger (14A2)

J. Stolz

S. Norris

P. Tam

OGC

D. Hagan (MNBB 3302)

E. Jordan (MNBB 3302)

B. Grimes (9A2)

T. Meek(8) (P1-137)

W. Jones (P-130A)

E. Butcher (11F23)

ACRS (10)

GPA/PA

ARM/LFMB

DFol  
1/1

Mr. J. Sieber  
Duquesne Light Company

Beaver Valley Power Station  
Units 1 & 2

cc:

Jay E. Silberg, Esquire  
Shaw, Pittman, Potts and Trowbridge  
2300 N Street, N.W.  
Washington, DC 20037

Bureau of Radiation Protection  
Pennsylvania Department of  
Environmental Resources-  
ATTN: R. Janati  
Post Office Box 2063  
Harrisburg, Pennsylvania 17120

Kenny Grada, Manager  
Nuclear Safety  
Duquesne Light Company  
P. O. Box 4  
Shippingport, Pennsylvania 15077

Mayor of the Borough of  
Shippingport  
Post Office Box 3  
Shippingport, Pennsylvania 15077

John A. Lee, Esquire  
Duquesne Light Company  
One Oxford Centre  
301 Grant Street  
Pittsburgh, Pennsylvania 15279

Ashley C. Schannauer  
Assistant City Solicitor  
City of Pittsburgh  
313 City-County Building  
Pittsburgh, Pennsylvania 15219

W.F. Carmichael, Commissioner  
Department of Labor  
1800 Washington Street East  
Charleston, West Virginia 25305

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

John D. Borrows  
Director, Utilities Department  
Public Utilities Commission  
180 East Broad Street  
Columbus, Ohio 43266-0573

Resident Inspector  
U.S. Nuclear Regulatory Commission  
Post Office Box 181  
Shippingport, Pennsylvania 15077

Director, Pennsylvania Emergency  
Management Agency  
Post Office Box 3321  
Harrisburg, Pennsylvania 17105-3321



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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139  
License No. DPR-66

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated January 5, 1989 complies 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 application, 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, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective on issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

  
John F. Stolz, Director  
Project Directorate I-4  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: April 11, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Appendix A (Technical Specifications) with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

<u>Remove</u>	<u>Insert</u>
VII	VII
VIII	VIII
IX	IX
X	X
XIII	XIII
XIV	XIV
XV	XV
XVI	XVI
XIX	XIX
XX	XX
XXI	XXI
XXII	XXII
XXIII	XXIII
XXIV	XXIV
XXV	XXV
XXVI	XXVI
3/4 4-8	3/4 4-8
3/4 11-22	3/4 11-22
6-13a	6-13a
6-15	6-15
6-22	6-22
B 3/4 2-5	B 3/4 2-5

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
3/4.4.9.2	Pressurizer . . . . .	3/4 4-27
3/4.4.9.3	Overpressure Protection Systems . . . .	3/4 4-27a
3/4.4.10	STRUCTURAL INTEGRITY - ASME Code Class 1, 2 and 3 Components . . . . .	3/4 4-28
3/4.4.11	RELIEF VALVES . . . . .	3/4 4-29
3/4.4.12	Reactor Coolant System Vents . . . . .	3/4 4-32
 <u>3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)</u>		
3/4.5.1	ACCUMULATORS . . . . .	3/4 5-1
3/4.5.2	ECCS SUBSYSTEMS - $T_{avg} \geq 350^{\circ}\text{F}$ . . .	3/4 5-3
3/4.5.3	ECCS SUBSYSTEMS - $T_{avg} < 350^{\circ}\text{F}$ . . .	3/4 5-6
3/4.5.4	BORON INJECTION SYSTEM	
3/4.5.4.1.1	Boron Injection Tank $\geq 350^{\circ}\text{F}$ . . . . .	3/4 5-7
3/4.5.4.1.2	Boron Injection Tank $< 350^{\circ}\text{F}$ . . . . .	3/4 5-7a
 <u>3/4.6 CONTAINMENT SYSTEMS</u>		
3/4.6.1	PRIMARY CONTAINMENT	
3/4.6.1.1	Containment Integrity . . . . .	3/4 6-1
3/4.6.1.2	Containment Leakage . . . . .	3/4 6-2
3/4.6.1.3	Containment Air Locks . . . . .	3/4 6-5
3/4.6.1.4	Internal Pressure . . . . .	3/4 6-6
3/4.6.1.5	Air Temperature . . . . .	3/4 6-8
3/4.6.1.6	Containment Structural Integrity . . .	3/4 6-10
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	
3/4.6.2.1	Containment Quench Spray System . . . .	3/4 6-11

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.6.2.2      Containment Recirculation Spray System	3/4 6-13
3/4.6.2.3      Chemical Addition System . . . . .	3/4 6-15
3/4.6.3        CONTAINMENT ISOLATION VALVES . . . . .	3/4 6-17
3/4.6.4        COMBUSTIBLE GAS CONTROL	
3/4.6.4.1      Hydrogen Analyzers . . . . .	3/4 6-20
3/4.6.4.2      Electric Hydrogen Recombiners . . . . .	3/4 6-21
3/4.6.5        SUBATMOSPHERIC PRESSURE CONTROL SYSTEM	
3/4.6.5.1      Steam Jet Air Ejector . . . . .	3/4 6-25
 <u>3/4.7 PLANT SYSTEMS</u>	
3/4.7.1        TURBINE CYCLE	
3/4.7.1.1      Safety Valves . . . . .	3/4 7-1
3/4.7.1.2      Auxiliary Feedwater System . . . . .	3/4 7-5
3/4.7.1.3      Primary Plant Demineralized Water (PPDW)	3/4 7-7
3/4.7.1.4      Activity . . . . .	3/4 7-8
3/4.7.1.5      Main Steam Line Isolation Valves . . .	3/4 7-10
3/4.7.2        STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION . . . . .	3/4 7-11
3/4.7.3        COMPONENT COOLING WATER SYSTEM . . . .	3/4 7-12
3/4.7.4        REACTOR PLANT RIVER WATER SYSTEM . . .	3/4 7-13
3/4.7.5        ULTIMATE HEAT SINK - OHIO RIVER . . . .	3/4 7-14
3/4.7.6        FLOOD PROTECTION . . . . .	3/4 7-15
3/4.7.7        CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS . . . . .	3/4 7-16
3/4.7.8        SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM . . . . .	3/4 7-19

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
3/4.7.9	SEALED SOURCE CONTAMINATION . . . . .	3/4 7-22
3/4.7.12	SNUBBERS . . . . .	3/4 7-26
3/4.7.13	AUXILIARY RIVER WATER SYSTEM . . . . .	3/4 7-34
3/4.7.14	FIRE SUPPRESSION SYSTEMS	
3/4.7.14.1	Fire Suppression Water System . . . . .	3/4 7-35
3/4.7.14.2	Spray and/or Sprinkler System . . . . .	3/4 7-39
3/4.7.14.3	Low Pressure CO <sub>2</sub> System . . . . .	3/4 7-41
3/4.7.14.4	Fire Hose Stations . . . . .	3/4 7-42
3/4.7.14.5	Halon Systems . . . . .	3/4 7-43
3/4.7.15	FIRE RATED ASSEMBLIES . . . . .	3/4 7-44

### 3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1	A.C. SOURCES	
3/4.8.1.1	Operating . . . . .	3/4 8-1
3/4.8.1.2	Shutdown . . . . .	3/4 8-5
3/4.8.2	ONSITE POWER DISTRIBUTION SYSTEMS	
3/4.8.2.1	A.C. Distribution - Operating . . . . .	3/4 8-6
3/4.8.2.2	A.C. Distribution - Shutdown . . . . .	3/4 8-7
3/4.8.2.3	D.C. Distribution - Operating . . . . .	3/4 8-8
3/4.8.2.4	D.C. Distribution - Shutdown . . . . .	3/4 8-10

### 3/4.9 REFUELING OPERATIONS

3/4.9.1	BORON CONCENTRATION . . . . .	3/4 9-1
3/4.9.2	INSTRUMENTATION . . . . .	3/4 9-2
3/4.9.3	DECAY TIME . . . . .	3/4 9-3

# INDEX

## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
3/4.9.4	CONTAINMENT BUILDING PENETRATIONS . . .	3/4 9-4
3/4.9.5	COMMUNICATIONS . . . . .	3/4 9-5
3/4.9.6	MANIPULATOR CRANE OPERABILITY . . . . .	3/4 9-6
3/4.9.7	CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING . . . . .	3/4 9-7
3/4.9.8.1	RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION . . . . .	3/4 9-8
3/4.9.8.2	LOW WATER LEVEL . . . . .	3/4 9-8a
3/4.9.9	CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM . . . . .	3/4 9-9
3/4.9.10	WATER LEVEL-REACTOR VESSEL . . . . .	3/4 9-10
3/4.9.11	STORAGE POOL WATER LEVEL . . . . .	3/4 9-11
3/4.9.12	FUEL BUILDING VENTILATION SYSTEM - FUEL MOVEMENT . . . . .	3/4 9-12
3/4.9.13	FUEL BUILDING VENTILATION SYSTEM - FUEL STORAGE . . . . .	3/4 9-13
3/4.9.14	FUEL STORAGE - SPENT FUEL STORAGE POOL	3/4 9-14
3/4.9.15	CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS . . . . .	3/4 9-16

### 3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1	SHUTDOWN MARGIN . . . . .	3/4 10-1
3/4.10.2	GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS . . . . .	3/4 10-2

# INDEX

## BASES

<u>SECTION</u>	<u>PAGE</u>
3/4.3.3.7 Chlorine Detection Systems . . . . .	B 3/4 3-3
3/4.3.3.8 Accident Monitoring Instrumentation . . . . .	B 3/4 3-3
3/4.3.3.9 Radioactive Liquid Effluent Monitoring Instrumentation . . . . .	B 3/4 3-4
3/4.3.3.10 Radioactive Gaseous Effluent Monitoring Instrumentation . . . . .	B 3/4 3-4
 <u>3/4.4 REACTOR COOLANT SYSTEM</u>	
3/4.4.1 REACTOR COOLANT LOOPS . . . . .	B 3/4 4-1
3/4.4.2 and 3/4.4.3 SAFETY VALVES . . . . .	B 3/4 4-1a
3/4.4.4 PRESSURIZER . . . . .	B 3/4 4-2
3/4.4.5 STEAM GENERATORS . . . . .	B 3/4 4-2
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE . . . . .	B 3/4 4-3
3/4.4.6.1 Leakage Detection System . . . . .	B 3/4 4-3
3/4.4.6.2 Operational Leakage . . . . .	B 3/4 4-3
3/4.4.7 CHEMISTRY . . . . .	B 3/4 4-4
3/4.4.8 SPECIFIC ACTIVITY . . . . .	B 3/4 4-4
3/4.4.9 PRESSURE/TEMPERATURE LIMITS . . . . .	B 3/4 4-5
3/4.4.10 STRUCTURAL INTEGRITY . . . . .	B 3/4 4-10
3/4.4.11 RELIEF VALVES . . . . .	B 3/4 4-10
3/4.4.12 REACTOR COOLANT SYSTEM VENTS . . . . .	B 3/4 4-11
 <u>3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)</u>	
3/4.5.1 ACCUMULATORS . . . . .	B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS . . . . .	B 3/4 5-1
3/4.5.4 BORON INJECTION SYSTEM . . . . .	B 3/4 5-1

# INDEX

## BASES

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.6 CONTAINMENT SYSTEMS</u>	
3/4.6.1 PRIMARY CONTAINMENT	
3/4.6.1.1 Containment Integrity . . . . .	B 3/4 6-1
3/4.6.1.2 Containment Leakage . . . . .	B 3/4 6-1
3/4.6.1.3 Containment Airlocks . . . . .	B 3/4 6-1
3/4.6.1.4 and 3/4.6.1.5 Internal Pressure and Air Temperature . . . . .	B 3/4 6-2
3/4 6.1.6 Containment Structural Integrity . . .	B 3/4 6-2
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS	
3/4.6.2.1 and 3/4.6.2.2 Containment Quench and Recirculation Spray Systems . . . . .	B 3/4 6-2
3/4.6.2.3 Chemical Addition System . . . . .	B 3/4 6-3
3/4.6.3 CONTAINMENT ISOLATION VALVES . . . . .	B 3/4 6-3
3/4.6.4 COMBUSTIBLE GAS CONTROL . . . . .	B 3/4 6-3
3/4.6.5 SUBATMOSPHERIC PRESSURE CONTROL SYSTEM	
3/4.6.5.1 Steam Jet Air Ejector . . . . .	B 3/4 6-3
<u>3/4.7 PLANT SYSTEMS</u>	
3/4.7.1 TURBINE CYCLE	
3/4.7.1.1 Safety Valves . . . . .	B 3/4 7-1
3/4.7.1.2 Auxiliary Feedwater Pumps . . . . .	B 3/4 7-2
3/4.7.1.3 Primary Plant Demineralized Water . . .	B 3/4 7-2
3/4.7.1.4 Activity . . . . .	B 3/4 7-3
3/4.7.1.5 Main Steam Line Isolation Valves . . .	B 3/4 7-3
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION . . . . .	B 3/4 7-4

# INDEX

## BASES

<u>SECTION</u>		<u>PAGE</u>
3/4.7.3	COMPONENT COOLING WATER SYSTEM . . . . .	B 3/4 7-4
3/4.7.4	RIVER WATER SYSTEM . . . . .	B 3/4 7-4
3/4.7.5	ULTIMATE HEAT SINK . . . . .	B 3/4 7-4
3/4.7.6	FLOOD PROTECTION . . . . .	B 3/4 7-4
3/4.7.7	CONTROL ROOM EMERGENCY HABITABILITY SYSTEM . . . . .	B 3/4 7-5
3/4.7.8	SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM . . . . .	B 3/4 7-5
3/4.7.9	SEALED SOURCE CONTAMINATION . . . . .	B 3/4 7-5
3/4.7.12	SNUBBERS . . . . .	B 3/4 7-6
3/4.7.13	AUXILIARY RIVER WATER SYSTEM . . . . .	B 3/4 7-7
3/4.7.14	FIRE SUPPRESSION SYSTEMS . . . . .	B 3/4 7-7
3/4.7.15	FIRE RATED ASSEMBLIES . . . . .	B 3/4 7-7
 <u>3/4.8 ELECTRICAL POWER SYSTEMS</u>		
3/4.8.1 AND 3/4.8.2	A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS . . .	B 3/4 8-1
 <u>3/4.9 REFUELING OPERATIONS</u>		
3/4.9.1	BORON CONCENTRATION . . . . .	B 3/4 9-1
3/4.9.2	INSTRUMENTATION . . . . .	B 3/4 9-1
3/4.9.3	DECAY TIME . . . . .	B 3/4 9-1
3/4.9.4	CONTAINMENT BUILDING PENETRATIONS . . .	B 3/4 9-1
3/4.9.5	COMMUNICATIONS . . . . .	B 3/4 9-2
3/4.9.6	MANIPULATOR CRANE OPERABILITY . . . . .	B 3/4 9-2
3/4.9.7	CRANE TRAVEL - SPENT FUEL STORAGE BUILDING . . . . .	B 3/4 9-2

## INDEX

### BASES

<u>SECTION</u>		<u>PAGE</u>
3/4.9.8	RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION . . . . .	B 3/4 9-2
3/4.9.9	CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM . . . . .	B 3/4 9-2
3/4.9.10 and 3/4.9.11	WATER LEVEL-REACTOR VESSEL AND STORAGE POOL . . . . .	B 3/4 9-3
3/4.9.12 and 3/4.9.13	FUEL BUILDING VENTILATION SYSTEM	B 3/4 9-3
3/4.9.14	FUEL STORAGE - SPENT FUEL STORAGE POOL	B 3/4 9-3
3/4.9.15	CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS . . . . .	B 3/4 9-4
 <u>3/4.10 SPECIAL TEST EXCEPTIONS</u>		
3/4.10.1	SHUTDOWN MARGIN . . . . .	B 3/4 10-1
3/4.10.2	GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS	B 3/4 10-1
3/4.10.3	PRESSURE/TEMPERATURE LIMITATIONS-REACTOR CRITICALITY	B 3/4 10-1
3/4.10.4	PHYSICS TESTS . . . . .	B 3/4 10-1
3/4.10.5	NO FLOW TESTS . . . . .	B 3/4 10-1

## INDEX

### ADMINISTRATIVE CONTROLS

---

<u>SECTION</u>	<u>PAGE</u>
<u>6.1 RESPONSIBILITY</u> . . . . .	6-1
<u>6.2 ORGANIZATION</u>	
6.2.1 Corporate . . . . .	6-1
6.2.2 Facility Staff . . . . .	6-1
<u>6.3 FACILITY STAFF QUALIFICATIONS</u> . . . . .	6-5
<u>6.4 TRAINING</u> . . . . .	6-5
<u>6.5 REVIEW AND AUDIT</u>	
6.5.1 ONSITE SAFETY COMMITTEE (OSC)	
6.5.1.1 Function . . . . .	6-5
6.5.1.2 Composition . . . . .	6-5
6.5.1.3 Alternates . . . . .	6-6
6.5.1.4 Meeting Frequency . . . . .	6-6
6.5.1.5 Quorum . . . . .	6-6
6.5.1.6 Responsibilities . . . . .	6-6
6.5.1.7 Authority . . . . .	6-7
6.5.1.8 Records . . . . .	6-7
6.5.2 OFFSITE REVIEW COMMITTEE (ORC)	
6.5.2.1 Function . . . . .	6-7
6.5.2.2 Composition . . . . .	6-8
6.5.2.3 Alternates . . . . .	6-8
6.5.2.4 Consultants . . . . .	6-8
6.5.2.5 Meeting Frequency . . . . .	6-9

# INDEX

## ADMINISTRATIVE CONTROLS

<u>SECTION</u>	<u>PAGE</u>
6.5.2.6 Quorum . . . . .	6-9
6.5.2.7 Review . . . . .	6-9
6.5.2.8 Audits . . . . .	6-10
6.5.2.9 Authority . . . . .	6-11
6.5.2.10 Records . . . . .	6-11
<u>6.6 REPORTABLE EVENT ACTION</u> . . . . .	6-11
<u>6.7 SAFETY LIMIT VIOLATION</u> . . . . .	6-12
<u>6.8 PROCEDURES</u> . . . . .	6-12
<u>6.9 REPORTING REQUIREMENTS</u> . . . . .	6-13
6.9.1 Routine Reports . . . . .	6-13a
6.9.1,2,3 Startup Reports . . . . .	6-13a
6.9.1.4,5 Annual Reports . . . . .	6-14
6.9.1.6 Monthly Operating Report . . . . .	6-15
6.9.1.10,11 Annual Radiological Environmental Report . . . . .	6-15
6.9.1.12,13 Semi-Annual Radioactive Effluent Release Report . . . . .	6-21
6.9.1.14 Radial Peaking Factor Limit Report . .	6-22
6.9.2 SPECIAL REPORTS . . . . .	6-22
<u>6.10 RECORD RETENTION</u> . . . . .	6-23
<u>6.11 RADIATION PROTECTION PROGRAM</u> . . . . .	6-25

## INDEX

### ADMINISTRATIVE CONTROLS

<u>6.12 HIGH RADIATION AREA . . . . .</u>	<u>6-25</u>
<u>6.14 PROCESS CONTROL PROGRAM (PCP) . . . . .</u>	<u>6-27</u>
<u>6.15 OFFSITE DOSE CALCULATION MANUAL . . . . .</u>	<u>6-27</u>
<u>6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT</u> <u>SYSTEMS . . . . .</u>	<u>6-28</u>
<u>6.17 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM . .</u>	<u>6-31</u>

# TABLE INDEX

<u>TABLE</u>	<u>TITLE</u>	<u>PAGE</u>
2.2-1	Reactor Trip System Instrumentation Trip Setpoints	2-6
3.1-1	Accident Analyses Requiring Re-evaluation in the event of an Inoperable Full or Part Length Rod	3/4 1-19A
3.2-1	DNB Parameters	3/4 2-13
3.3-1	Reactor Trip System Instrumentation	3/4 3-2
3.3-2	Reactor Trip System Instrumentation Response Times	3/4 3-9
4.3-1	Reactor Trip System Instrumentation Surveillance Requirements	3/4 3-11
3.3-3	Engineered Safety Features Actuation System Instrumentation	3/4 3-15
3.3-4	Engineered Safety Features Actuation System Instrumentation Trip Setpoints	3/4 3-22
3.3-5	Engineered Safety Feature Response Times	3/4 3-25
4.3-2	Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements	3/4 3-29
3.3-6	Radiation Monitoring Instrumentation	3/4 3-34
4.3-3	Radiation Monitoring Instrumentation Surveillance Requirements	3/4 3-36
3.3-7	Seismic Monitoring Instrumentation	3/4 3-39
4.3-4	Seismic Monitoring Instrumentation Surveillance Requirements	3/4 3-40
3.3-8	Meteorological Monitoring Instrumentation	3/4 3-42
4.3-5	Meteorological Monitoring Instrumentation Surveillance Requirements	3/4 3-43
3.3-9	Remote Shutdown Panel Monitoring Instrumentation	3/4 3-45
4.3-6	Remote Shutdown Monitoring Instrumentation Surveillance Requirements	3/4 3-46
3.3-10	Fire Detection Instruments	3/4 3-48

Table Index (cont.)

<u>TABLE</u>	<u>TITLE</u>	<u>PAGE</u>
3.3-11	Accident Monitoring Instrumentation	3/4 3-51
4.3-7	Accident Monitoring Instrumentation Surveillance Requirements	3/4 3-52
3.3-12	Radioactive Liquid Effluent Monitoring Instrumentation	3/4 3-54
4.3-12	Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements	3/4 3-57
3.3-13	Radioactive Gaseous Effluent Monitoring Instrumentation	3/4 3-60
4.3-13	Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements	3/4 3-65
4.4-1	Minimum Number of Steam Generators to be Inspected During Inservice Inspection	3/4 4-10c
4.4-2	Steam Generator Tube Inspection	3/4 4-10d
4.4-3	Reactor Coolant System Pressure Isolation Valves	3/4 4-14b
3.4-1	Reactor Coolant System Chemistry Limits	3/4 4-16
4.4-10	Reactor Coolant System Chemistry Limits Surveillance Requirements	3/4 4-17
4.4-12	Primary Coolant Specific Activity Sample and Analysis Program	3/4 4-20
4.4-5	Reactor Vessel Material Irradiation Surveillance Schedule	3/4 4-26
3.6-1	Containment Penetrations	3/4 6-19a
3.7-1	Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable Steamline Safety Valves During 3 Loop Operation	3/4 7-2
3.7-2	Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steamline Safety Valves During 2 Loop Operation	3/4 7-3
3.7-3	Steamline Safety Valves Per Loop	3/4 7-4
4.7-2	Secondary Coolant System Specific Activity Sample and Analysis Program	3/4 7-9

# Table Index (cont.)

<u>TABLE</u>	<u>TITLE</u>	<u>PAGE</u>
3.8-1	Battery Surveillance Requirements	3/4 8-9a
3.9-1	Beaver Valley Fuel Assembly Minimum Burnup vs. Initial U235 Enrichment For Storage in Region 2 Spent Fuel Racks	3/4 9-15
4.11-1	Radioactive Liquid Waste Sampling and Analysis Program	3/4 11-3
4.11-2	Radioactive Gaseous Waste Sampling and Analysis Program	3/4 11-13
3.12-1	Radiological Environmental Monitoring Program	3/4 12-3
3.12-2	Reporting Levels for Radioactivity Concentrations in Environmental Samples	3/4 12-6
4.12-1	Maximum Values for the Lower Limits of Detection	3/4 12-7
8 3/4.4-1	Reactor Vessel Toughness Data (unirradiated)	8 3/4 4-7
6.2-1	Minimum Shift Crew Composition	6-4
6.9-1	Environmental Radiological Monitoring Program Summary	6-20

## Figure Index

<u>FIGURE</u>	<u>TITLE</u>	<u>PAGE</u>
2.1-1	Reactor Core Safety Limit - Three Loops in Operation	2-2
2.1-2	Reactor Core Safety Limit - Two Loops in Operation (One Loop Isolated)	2-3
2.1-3	Reactor Core Safety Limit - Two Loops in Operation (No Isolated Loop)	2-4
3.1-1	Rod Group Insertion Limits Versus Thermal Power - Three Loop Operation	3/4 1-24
3.1-2	Rod Group Insertion Limits Versus Thermal Power - Two Loop Operation	3/4 1-25
3.2-1	Axial Flux Difference Limits as a Function of Rated Thermal Power	3/4 2-4
3.2-2	K(z) - Normalized $F_Q(z)$ as a function of Core Height	3/4 2-7
3.4-1	Dose Equivalent I-131 Primary Coolant Specific Activity Limit Versus Percent of Rated Thermal Power with the Primary Coolant Specific Activity > 1.0 Ci/gram Dose Equivalent I-131	3/4 4-21
3.4-2	Beaver Valley Unit No. 1 Reactor Coolant System Heatup Limitations Applicable for the First 9.5 EFPY	3/4 4-24
3.4-3	Beaver Valley Unit No. 1 Reactor Coolant System Cooldown Limitations Applicable for the First 9.5 EFPY	3/4 4-25
3.6-1	Maximum Allowable Primary Containment Air Pressure Versus River Water Temperature and RWST Water Temperature	3/4 6-7
B 3/4.2-1	Typical Indicated Axial Flux Difference Versus Thermal Power at BOL.	B 3/4 2-3
B 3/4.4-1	Fast Neutron Fluence ( $E > 1$ Mev) as a Function of Full Power Service Life	B 3/4 4-6a
B 3/4.4-2	Effect of Fluence, Copper Content, and Phosphorus Content on $\Delta RT_{NDT}$ for Reactor Vessel Steels Per Reg. Guide 1.99	B 3/4 4-6b

Figure Index (cont.)

<u>FIGURE</u>	<u>TITLE</u>	<u>PAGE</u>
5.1-1	Site Boundary for Gaseous and Liquid Effluents for the Beaver Valley Power Station	5-1b
5.1-3	Exclusion Area - Beaver Valley Power Station	5-1d
5.1-4	Low Population Zone - Beaver Valley Power Station	5-1e
5.1-5	Gaseous Release Points - Beaver Valley Power Station	5-2
5.1-6	Liquid Release Points - Beaver Valley Power Station	5-3
6.2-1	Corporate Organization (Partial)	6-2
6.2-2	Facility Organization	6-3

## REACTOR COGLANT SYSTEM

### STEAM GENERATORS

#### LIMITING CONDITION FOR OPERATION

---

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing  $T_{avg}$  above 200°F.

#### SURVEILLANCE REQUIREMENTS

---

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy Current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.
- b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
  1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and
  2. Tubes in those areas where experience has indicated potential problems.

## RADIOACTIVE EFFLUENTS

### 3/4.11.3 SOLID RADIOACTIVE WASTE

#### LIMITING CONDITION FOR OPERATION

---

3.11.3.1 The solid radwaste system shall be used, as applicable, to solidify and package radioactive wastes, and to ensure meeting the requirements of 10 CFR Part 20, 10 CFR Part 61 and of 10 CFR Part 71. Methods utilized to meet these requirements shall be described in facility procedures and in the Process Control Program (PCP).

APPLICABILITY At all times.

#### ACTION:

- a. With the applicable requirements of 10 CFR Part 20, 10 CFR Part 61 and 10 CFR Part 71 not satisfied, suspend affected shipments of solid radioactive wastes from the site.
- b. The provisions of Specification 3.0.3 are not applicable.

#### SURVEILLANCE REQUIREMENTS

---

4.11.3.1.1 Prior to shipment, solidification shall be verified in accordance with Station Operating Procedures.

4.11.3.1.2 Reports. The semi-annual Radioactive Effluent Release Report in Specification 6.9.1.12 shall include the following information for each type of solid waste shipped offsite during the report period:

- a. container volume;
- b. total curie quantity (determined by measurement or estimate);
- c. principal radionuclides (determined by measurement or estimate);
- d. type of waste (e.g., spent resin, compacted dry waste evaporator bottoms);
- e. type of container (e.g., LSA, Type A, Type B, Large Quantity);
- f. solidification agent (e.g., cement, urea formaldehyde); and
- g. classification and other requirements specified by 10 CFR Part 61.

## ADMINISTRATIVE CONTROLS

### 6.9 REPORTING REQUIREMENTS

#### ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk.

#### STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing will be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

## ADMINISTRATIVE CONTROLS

### MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.9.1.7 DELETED by Amendment No. 84

6.9.1.8 DELETED by Amendment No. 84

6.9.1.9 DELETED by Amendment No. 84

### ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT<sup>3</sup>

6.9.1.10 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year and will include reporting any deviations not reported under 6.9.2 with respect to the Radiological Effluent Technical Specifications.

6.9.1.11 The annual radiological environmental reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

<sup>3</sup> A single submittal may be made for a multiple unit site. The submittal should combine these sections that are common to both units.

## ADMINISTRATIVE CONTROLS

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real individual from reactor releases for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1. The SKYSHINE code (available from Radiation Shielding Information Center, ORNL) is acceptable for calculating the dose contribution from direct radiation due to N-16.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM.

The radioactive effluent release reports shall also include any licensee initiated changes to the ODCM made during the 6 month period.

### RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.14 The  $F_{xy}$  limit for Rated Thermal Power ( $F^{RTP}$ ) shall be provided for all Core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support  $F^{RTP}$  will be by request from the NRC and need not be included in this report.

### SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, 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:

## POWER DISTRIBUTION LIMITS

### BASES

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, and completely offsets any rod bow penalties (< 1.3% for the worst case which occurs at a burnup of 24,000 MWD/MTU).

This margin includes the following:

1. Design Limit DNBR of 1.30 vs. 1.28
2. Grid Spacing ( $K_g$ ) of 0.046 vs. 0.059
3. Thermal Diffusion Coefficient of 0.038 vs. 0.059
4. DNBR Multiplier of 0.865 vs. 0.88
5. Pitch reduction

The radial peaking factor  $F_{xy}$  (Z) is measured periodically to provide assurance that the hot channel factor,  $F_0$  (Z), remains within its limit. The  $F_{xy}$  limit for Rated Thermal Power ( $F_{xy}^{RTP}$ ) as provided in the Radial Peaking Factor Limit Report per specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

### 3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts.

The two-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on  $F_0$  is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.



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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14  
License No. NPF-73

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by Duquesne Light Company, et al. (the licensee) dated January 5, 1989 complies 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 application, 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.

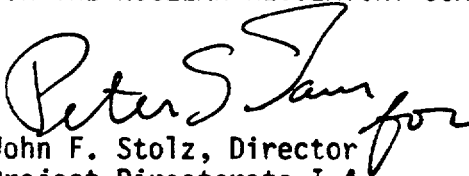
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 14, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated in the license. DLCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective on issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in dark ink, appearing to read "Peter S. Stolz", with a stylized flourish at the end.

John F. Stolz, Director  
Project Directorate I-4  
Division of Reactor Projects I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: April 11, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of the Appendix A (Technical Specifications) with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove

3/4 4-11

6-13

6-15

6-18

B 3/4 2-4

Insert

3/4 4-11

6-13

6-15

6-18

B 3/4 2-4

## REACTOR COOLANT SYSTEM

### 3/4.4.5 STEAM GENERATORS

#### LIMITING CONDITION FOR OPERATION

---

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing  $T_{avg}$  above 200°F.

#### SURVEILLANCE REQUIREMENTS

---

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy Current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10 CFR 50, Section 50.55a(g). The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;
- b. The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:
  - 1) All nonplugged tubes that previously had detectable wall penetrations (> than 20%),
  - 2) Tubes in those areas where experience has indicated potential problems.

## ADMINISTRATIVE CONTROLS

### PROCEDURE (Continued)

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the OSC and approved by the Plant Manager, predesignated alternate or a predesignated Manager to whom the Plant Manager has assigned in writing the responsibility for review and approval of specific subjects, within 14 days of implementation.

6.8.4 A Post-Accident monitoring program shall be established, implemented, and maintained. The program will provide the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples following an accident. The program shall include the following:

- (i) Training of personnel,
- (ii) Procedures for sampling and analysis, and
- (iii) Provisions for maintenance of sampling and analysis equipment.

6.8.5 A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

- a. Identification of a sampling schedule for the critical parameters and control points for these parameters;
- b. Identification of the procedures used to measure the values of the critical parameters;
- c. Identification for process sampling points;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for off control point chemistry conditions; and
- f. A procedure identifying:
  - 1) the authority responsible for the interpretation of the data, and
  - 2) the sequence and timing of administrative events required to initiate corrective action.

## 6.9 REPORTING REQUIREMENTS

### ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk.

## ADMINISTRATIVE CONTROLS

---

### ANNUAL REPORTS (Continued)

- b. Documentation of all challenges to the pressurizer power operated relief valves (PORVS) or pressurizer safety valves.
- c. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

### MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.9.1.7 This item intentionally blank

6.9.1.8 This item intentionally blank

6.9.1.9 This item intentionally blank

### ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT<sup>3</sup>

6.9.1.10 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year and will include reporting any deviations not reported under 6.9.2 with respect to the Radiological Effluent Technical Specifications.

6.9.1.11 The annual radiological environmental reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the

---

<sup>3</sup>A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to both units.

## ADMINISTRATIVE CONTROLS

### SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real individual from reactor releases for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1. The SKYSHINE Code (available from Radiation Shielding Information Center, (ORNL) is acceptable for calculating the dose contribution from direct radiation due to N-16.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with ODCM.

The radioactive effluent release reports shall also include any licensee initiated changes to the ODCM made during the 6 month period.

### RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.14 The  $F_{xy}$  limit for Rated Thermal Power ( $F_{xy}^{RTP}$ ) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes at least 60 days prior to cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission.

Any information needed to support  $F_{xy}^{RTP}$  will be by request from the NRC and need not be included in this report.

### SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk 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. ECCS Actuation, Specifications 3.5.2 and 3.5.3.
- b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.
- c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

## POWER DISTRIBUTION LIMITS

### BASES

#### 3/4.2.2 and 3/4 2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS $F_Q(Z)$

AND  $F_{\Delta H}^N$  (Continued)

- c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained.
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in  $F_{\Delta H}^N$  as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.  $F_{\Delta H}^N$  will be maintained within its limits provided conditions a thru d above, are maintained.

When an  $F_Q$  measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate experimental error allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

The specified limit of  $F_{\Delta H}^N$  contains an 8% allowance for uncertainties which means that normal, full power, three loop operation will result in  $F_{\Delta H}^N \leq 1.55/1.08$ .

Fuel rod bowing reduces the value of the DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, and completely offsets any rod bow penalties (< 1.3% for the worst case which occurs at a burnup of 24,000 MWD/MTU).

This margin includes the following:

- 1. Design Limit DNBR of 1.30 vs. 1.28
- 2. Grid Spacing ( $K_g$ ) of 0.046 vs. 0.059
- 3. Thermal Diffusion Coefficient of 0.038 vs. 0.059
- 4. DNBR Multiplier of 0.865 vs. 0.88
- 5. Pitch reduction

The radial peaking factor  $F_{xy}(Z)$  is measured periodically to provide assurance that the hot channel factor,  $F_Q(Z)$ , remains within its limit. The  $F_{xy}$  limit for Rated Thermal Power ( $F_{xy}^{RTP}$ ) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.14 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

#### 3/4.2.4 QUADRANT POWER TILT RATIO

The Quadrant Power Tilt Ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 139 AND 14 TO

FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

DUQUESNE LIGHT COMPANY  
OHIO EDISON COMPANY  
PENNSYLVANIA POWER COMPANY  
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY  
THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNITS NO. 1 AND NO. 2

DOCKET NOS. 50-334 AND 50-412

INTRODUCTION

By letter dated January 5, 1989, Duquesne Light Company (the licensee, acting as agent for the above utilities), submitted a request to amend the Technical Specifications for Beaver Valley Power Station, Units 1 and 2. These changes are all administrative in nature.

DISCUSSION AND EVALUATION

(1) Index to the Unit Technical Specifications

The index is updated to reflect sections deleted or added by previous amendments. The index contains no requirements and its modification has no effects on requirements in the Technical Specifications. These change are acceptable.

(2) Section 4.4.5.2, Steam Generator Tube Sample Selection and Inspection

This section has been revised to delete specific references to the dates of ASME documents. The versions of the ASME documents to be used are clearly stated in 10 CFR 50.55a(g). Thus this change would eliminate future need of Technical Specification changes whenever the applicable ASME documents are revised. The revised section is consistent with Section 4.0.5, where inservice inspection is addressed. This change is acceptable.

(3) Section 4.11.3, Surveillance Requirement for Solid Radioactive Waste (Unit 1)

The contents of the semi-annual Radioactive Effluent Release Report has been expanded by an additional item 4.11.3.1.2.g. Now the "classification and other requirements specified by 10 CFR Part 61" of solid wastes shipped offsite will also be reported. This new requirement is in accordance with 10 CFR 61, is identical to a corresponding requirement in the Unit 2 Technical Specification, and is acceptable.

(4) Sections 6.9.1 and 6.9.2, Reporting Requirements

This section is revised to state that the NRC recipient for all routine and special reports is the "Document Control Desk." This change complies with 10 CFR 50.4(b)(1) and is acceptable.

(5) Basis Section 3/4.2.2, Heat Flux and Nuclear Enthalpy Hot Channel Factors

This basis section has been revised by reducing the value of the burnup at which the maximum rod bow penalty on core operating limits must be evaluated. The previous limit of 33,000 MWD/MTU has been reduced to 24,000 MWD/MTU based on NRC approval (letter from C. Berlinger to E.P. Rahe, Jr., dated June 18, 1986) of the previous conservative evaluation made at the time the rod bow report (December, 1983) was approved. This results in a reduction in the maximum rod bow penalty from less than 3% to less than 1.3%.

The licensee will revise appropriate sections of the Updated Final Safety Analysis Report to reflect this changed basis.

This change reflects an NRC approval issued to Westinghouse, and is acceptable.

ENVIRONMENTAL CONSIDERATION

The amendments relate to changes in recordkeeping, reporting or administrative procedures or requirements, or make changes in technical specification bases to reflect previous staff approvals. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

CONCLUSION

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 (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 7, 1989

Principal Contributor: Peter S. Tam