55-60755
2/21/88
A-5

DOCKETED
USNRC

BEFORE THE ADMINISTRATIVE JUDGE

'88 MAR -8 P4:00

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of	)	Docket No. 55-60755
	)	
Alfred J. Morabito	)	ASLBP No. 87-551-02-SP
	)	
Senior Operator License for	)	
Beaver Valley Power Station, Unit 1	)	February 4, 1988

OBJECTION TO LACK OF THOROUGHNESS OF JOINT AFFIDAVIT OF BARRY S. NORRIS AND DAVID M. SILK

By Memorandum And Order dated November 24, 1987, the Administrative Judge posed a series of questions which were originally to be answered by both parties by December 21, 1987. The answers were to be sworn into the record at an oral presentation scheduled for January, 1988.

The Staff filed a motion on December 2, 1987 requesting an extension of time until January 29, 1988 to answer the questions. The Staff argued that the time required to answer the sixty three questions was significant and would, for some answers, require travel to the Beaver Valley plant to locate procedures. (Motion, December 2, 1987 at 2).

By Order dated december 7, 1987, the Administrative Judge granted the Staff's motion for good cause. I did not object. The oral presentation was rescheduled to February 22, 1988.

On December 11, 1987, the Staff moved for reconsideration of the Administrative Judge's Order of November 24, 1987

Morabito
Exh 5
EMEA
NRC
2/2,/88

00:19 8- FAM 88

NUCLEAR REGULATORY COMMISSION

Docket No. 55-6075 Official Exh. No. 5

In the matter of Alfred Morabito

Staff ____ IDENTIFIED x

Applicant x RECEIVED ____

Intervenor ____ REJECTED ____

Cont'g Off'r ____

Contractor HR DATE 2/21/88

Other ____ Witness ____

Reporter Andrew Emerson

Case No. 55-6075 Morabito Exh 5

Morabito

2/21/88 EMEA

arguing that certain of the questions posed therein were beyond the Administrative Judge's jurisdiction.

By Memorandum And Order dated December 17, 1987, the Administrative Judge denied the Staff's motion but did modify and clarify rhetorical questions 2a-d (rebuttal submittal dated November 7, 1987, pp 8-9). In particular, a statistical base for ascertaining NRC grading practices was required to be developed with regard to simulator examinations, but only for those examinations administered at Beaver Valley Unit 1. (Memorandum And Oreder dated December 17, 1987, pp3-4).

In response to all of the above, the Staff filed their affidavit on January 29, 1988. That affidavit failed to answer all of the questions; though the stated need for an extension of time to accomplish that task formed the basis for the Administrative Judge's Order granting a time extension. In particular the Staff failed to provide the statistical basis for NRC grading practices on simulator examinations. In support of that failure, the Staff stated that it was not possible to find answers to the rhetorical questions, as modified and clarified by the Memorandum And Or der dated December 17, 1987, in other candidates' records because examiners are only required to document actions that support a rating of unsatisfactory. (Affidavit dated January 29, 1988, ¶ 62).

I believe the Staff has produced another, in a series of many, misrepresentation regarding the facts. For proof of my belief, I submit the attached letter from Harry B. Kister to A. Christopher Bakken, III dated March 21, 1986. The attachments to that letter, which must certainly be copies of originals which are on file at the Region I offices, show, at least in this one case, that comments related to good and to marginal performance, as well as several related to unsatisfactory performance are made and must be available to the Staff.

I request that, without any delay to the date of the oral presentation, the Staff develop the statistical base requested by Judge Bechhoefer using the attached letter and any similar letters available to the examiners. In addition, they should compare the number+seriousness of the comments in the attached letter to the number and seriousness of the comments generated by my performance.

As an aside, at least one of the more significant comments actually stated in the attachments is incorrect. Regardless though, when the examiner generated the comment he was convinced that it was accurate. Subsequently, Mr. Bakken was awarded his SRO license. He has performed his duties in an exemplary manner since then.

There are other errors in the Staff's affidavit of January 29, 1988. I request that the Presiding Officer allow me to explain those errors at the oral presentation.

Respectfully Submitted This 4th Day Of February, 1988,

Alfred J. Morabito

Alfred J. Morabito

Service of this document is through U.S. mail, 1st class, certified, return receipt requested as follows:

Secretary of the Commission:

Original plus 2 copies

Administrative Judge Charles Bechhoefer:

one copy

Administrative Judge David L. Hetrick:

one copy

Attorney Jay Gutierrez:

one copy

Attorney Colleen P. Woodhead:

one copy

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

MAR 2 1 1986

Docket No. 55-60744

A. Christopher Bakken, III
2995 Tuscarawas Road
Beaver, PA 15009

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974 (Public Law 93-438), and subject to the conditions and limitations incorporated herein, the Nuclear Regulatory Commission hereby licenses you to direct the licensed activities of licensed operators at, and to manipulate all controls of the Beaver Valley Power Station, Unit No. 1, Facility License No. DPR-66.

Your License No. is SOP-10479. The Docket No. is 55-60744. The effective date is March 12, 1986.

This license is subject to the provisions of Section 55.31 of the U.S. Nuclear Regulatory Commission's regulations, Title 10, Code of Federal Regulations, Chapter 1, Part 55, with the same force and effect as if fully set forth herein.

In directing the licensed activities of licensed operators and in manipulating the controls of the above facility you shall observe the operating procedures and other conditions specified in the facility license which authorizes operation of the facility.

The issuance of this license is based upon examination of your qualifications, including the representations and information contained in your application for license filed under the docket number indicated above.

Unless sooner terminated, this license shall expire two years from the effective date.

A copy of this license has been made available to the facility licensee.

For the Nuclear Regulatory Commission

Harry B. Kister

Harry B. Kister, Chief
Projects Branch No. 1
Division of Reactor Projects

cc: Duquesne Light Co.
ATTN: Station Superintendent
Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077

ES-301

NRC Form 157C, Page 1
(9-84)

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR OPERATOR EXAMINATION REPORT

INSTRUCTIONS: Provide all the information requested for Power and Non-Power Reactors. Screened items do not apply to Non-Power Reactors.

DOCKET NUMBER 55- 60744

TYPE OF REACTOR	TYPE OF EXAM
[X] POWER	[X] INITIAL
[ ] NON-POWER	[ ] RETAKE

CANDIDATE'S NAME	REACTOR	LOCATION
A. CHRISTOPHER BAKKEN III	Beaver Valley 1	Shippingport, PA

WRITTEN EXAMINATION

SENIOR OPERATOR (Section 55.22)	
ADMINISTERED BY	~~S. Barber~~ ss D.H. Coe
DATE	2/4/86
GRADED BY	D.H. Coe
GRADE	85.3
[ ] WAIVED	
EVALUATION	[X] PASSED 85.3 [ ] FAILED

CATEGORY GRADE	5	6	7	8
	93.6 %	91.4 %	70.3 %	85.8 %
	H %	I %	J %	K % / L %

OPERATING TEST

(Section 55.23)	
ADMINISTERED BY	S. Barber / R.M. Keller
DATE	2/5/86
RESULT	[ ] WAIVED [X] PASSED [ ] FAILED
	[X] HOT [ ] COLD

SIMULATOR TEST (Not applicable to Non-Power Reactors)

(Section 55.23)	
ADMINISTERED BY	S. Barber / R.M. Keller
DATE	2/5/86
RESULT	[ ] WAIVED [X] PASSED [ ] FAILED

COMMENTS

RMK 3/12/86

RECOMMENDATION	SIGNATURE - EXAMINER	DATE
[V] APPROVE FOR SENIOR LICENSE	S. Barber / R.M. Keller	2/9/86
[ ] DO NOT APPROVE FOR SENIOR LICENSE		
ISSUE LICENSE (Signature) NB Kostin	DATE 3/21/86	DENY APPLICATION (Signature) / DATE

SENIOR OPERATOR
OPERATING AND ORAL EXAMINATION SUMMARY REPORT

*For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.	EVALUATION SRO	*PAGE NUMBER FOR "U"
1. OPERATING ☐ DISCUSSION ☒ DEMONSTRATION		
1.1 Pre-startup and Instrument Checks	S	
1.2 Console Operation		
a. Manipulations	M	Form 304 pg 1
b. Understanding	S	
1.3 Plant Direction and Control	S ①	
2. FACILITY EQUIPMENT		
a. Major	S	
b. Auxiliary	S	
c. Engineered Safeguards Systems	M	pg 4,5
d. Electrical	S	
3. INSTRUMENTATION		
a. Nuclear	S	
b. Process	S	
4. PLANT PROTECTION	M	pg 4,5
5. PROCEDURES		
a. Normal	S	
b. Offnormal/Abnormal	S	
c. Emergency	M	Form 304 pg 2,3
6. a. REACTIVITY EFFECTS (Except Console Operation)	S	
b. THERMODYNAMICS AND HYDRAULICS	S	
7. ADMINISTRATIVE REQUIREMENTS	S	
8. RESPONSIBILITIES AND AUTHORITIES		
a. Radiation Protection and Control	S	
b. Emergency Plan	S	
c. Other Duties and Responsibilities	S	

COMMENTS
① Very agressive pursuit of total loss of FW

OPERATING AND ORAL EXAMINATION NOTES

A. OPERATING DEMONSTRATION — See Form 309	EVALUATION
CHECK ONE: [ ] REACTOR STARTUP [ ] STARTUP CERTIFICATION [X] SIMULATOR DEMONSTRATION	
1.1 Pre-Startup or Instrument Checks. Type of Checkout *(specify)*	
1.1.1 Familiarity with checksheet	
1.1.2 Accuracy when reading instruments	
1.1.3 Understanding of what is being checked	
1.1.4 Understanding of reasons for checkout	
1.2 Console Operation	
a. Initial conditions	
b. Program	
c. Understanding	
1.2.1 Ability to predict response for specific program	
1.2.2 Understanding of instrument response	
1.2.3 Knowledge of reactivity effects	
d. Manipulations	
1.2.4 Follows procedures	
1.2.5 Observes and checks instrumentation	
1.2.6 Ability to follow specified program accurately	
1.2.7 Dexterity and "feel" for console controls	
1.3 Plant Facility Direction and Control	
1.3.1 Ability to direct plant operation	S
1.3.2 Plant parameter verification *(ECP, heat balance, etc.)*	S ②
1.3.3 Technical specification requirements	M ①
1.3.4 Equipment OOS requirements	S
1.3.5 Experimental Facilities *(Non Power Reactors only)*	N/A

COMMENTS

① With both Ximp channels INOP said T.S. 3.0.3. applied vice action 12 ② Said emergency boration would lower only temp - corrected self later

B. CONTROL ROOM (Major, Auxiliary and Engineered Safeguards Systems)	SYSTEMS CVCS A	SGWLCS B	 C	AFW D	FIRE PROTECTION E	 F	CORE SIS ACC G	SSPS-ESF H
2.0 EQUIPMENT								
2.1 Purpose	S	S						
2.2 Flow Path	S			S	S		S	
2.3 Normal Parameters	S						S	S
2.4 Components					S		S	S
2.5 System Behavior and Response	S	S		S			S	U②
3.0 INSTRUMENTATION								
3.1 Detector		S						
3.2 Malfunction		S		S			S	
3.3 Control Room Indication	S				S		S	S
4.0 PLANT PROTECTION								
4.1 Alarms/Setpoints	S	S					S	S
4.2 Safety System Input				S				
4.3 Interlocks	M①				S			
5.0 PROCEDURES								
5.1 Normal Procedures	S						S	
5.2 Offnormal/Abnormal Procedures	S	S			S		S	
5.3 Emergency Procedures	S			S				U③
6.0 A. Reactivity Effects		S		S				
B. Thermodynamic Analysis/Thermal Effects		S						
7.0 ADMINISTRATIVE REQUIREMENTS								
7.1 Technical Specifications	S	U④			S		U④	S
7.2 Facility Requirements	S							

COMMENTS (Required for "U")
① Listed all chg. pump trips except racking in C lnk.
② Failed to remove FW isolation signal by SI reset.

☐ CONTINUED ON REVERSE

COMMENTS *(Continued)*

③ Skipped procedure step to verify contain. FW isolation valves open. Confused SI reset and SI termination criteria. as PO requested
④ ~~Misinterpreted require~~ permission to perform immediate actions after a Rx trip.

B. CONTROL ROOM (Nuclear and Radiation Instruments)	SYSTEMS: SR	IR		Containment RMS	
	A	B	C	D	E
3.0 INSTRUMENTS					
3.1 Detectors	S	S			
3.2 Malfunctions	S	S		S	
3.3 Control Room Indications	S	S		~~U~~ (1)	
3.4 Channel Components	S	S		S	
3.5 Compensation/Discriminator	S	S		~~S~~	
3.6 Input to Control System					
4.0 PLANT PROTECTION					
4.1 Alarms/Setpoints				S	
4.2 Safety System Input	S	U (2)			
4.3 Interlocks	S	S			
5.0 PROCEDURES					
5.1 Normal Procedures					
5.2 Offnormal/Abnormal Procedures				S	
5.3 Emergency Procedure					
7.0 ADMINISTRATIVE REQUIREMENTS					
7.1 Technical Specifications					
7.2 Facility Requirements	S			S	

COMMENTS (Required for "U")

(1) Effects of ~~of~~ a small RCS leak into containment with failed fuel - RMS ↑

(2) Unaware that pulling either control or instrument power fuse would trip channel.

B. CONTROL ROOM (Electrical)	SYSTEMS: A Offsite Elec Dist.	B	C Vital AC Pwr Inverters	D
2.0 EQUIPMENT				
2.1 Purpose			S	
2.2 Flow Path	S		S	
2.3 Normal Parameters				
2.4 Components	S			
2.5 System Behavior or Response	S		U ①	
3.0 INSTRUMENTS				
3.2 Interlocks				
3.4 Control Room Indication	S		S	
5.0 PROCEDURES				
5.1 Normal Procedures	S			
5.2 Offnormal/Abnormal			S	
5.3 Emergency Procedures				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS (Required for "U")

① Unable to determine how inverter converted DC to AC

C. REACTOR AND AUXILIARY BUILDINGS *(Power Reactors)*
(Major, Auxiliary, Electrical Safeguards, Fuel Handling)
FACILITY WALK THROUGH *(Non-Power Reactors)*

	SYSTEMS					
	AFW		Inverters			
	A	B	C	D	E	F
2.0 EQUIPMENT						
2.2 Flow Paths	S		M ①			
2.3 Normal Parameters	S		S			
2.4 Equipment Location	S		S			
2.5 System Behavior and Response	S		S			
3.0 INSTRUMENTS						
3.8 Local Instrumentation	S		S			
5.0 PROCEDURES						
5.1 Normal Procedures (Local)						
5.2 Offnormal/Abnormal Procedures (Local)						
5.3 Emergency Procedures (Local)	S					
6.0A. REACTIVITY EFFECTS						
B. THERMODYNAMICS ANALYSIS/Thermal Effects	S					
7.0 ADMINISTRATIVE REQUIREMENTS						
7.1 Technical Specifications						
7.2 Facility Requirements						

COMMENTS *(Required for "U")*

① Unsure of DC flowpath upstream of inverter

D. DISCUSSIONS *(Integrated Plant Response)* See Form 309	SYSTEMS A	B	C	D
2.0 EQUIPMENT				
2.6 Components Response				
3.0 INSTRUMENTS				
3.4 Control Room Indications				
3.8 Automatic Control				
3.9 Ability to Manipulate Manual Control				
4.0 PLANT PROTECTION				
4.1 Automatic Actions				
4.2 Alarm/Setpoints				
5.0 PROCEDURES				
5.1 Normal Procedures				
5.2 Offnormal/Abnormal Procedures				
5.3 Emergency Procedures				
6.0 REACTIVITY EFFECTS AND THERMODYNAMIC ANALYSIS				
6.3 Coefficient Effects/Reactivity Effects				
6.6 Transient Analysis/Thermal Analysis				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS *(Required for "U")*

D. DISCUSSION *(Power Reactors)*	EVAL-UATION
6.0 THEORY OF NUCLEAR POWER PLANT OPERATION A. REACTIVITY EFFECT *(Nuclear Theory)*	
6.A.1 Subcritical Multiplication	S
6.A.2 Delayed Neutrons Effect	S
6.A.3 Coefficients	S
6.A.4 Poison Effects	S
6.A.5 Long Term Exposure Effects	S
6.A.6 Axial and Radial Limits	S
6.A.7 Shutdown Margin	S
6.A.7 Safety Limits	S
B. THERMODYNAMICS AND HYDRAULICS 6.B.1 Steam Tables	S
6.B.2 Instrumentation	S
6.B.3 Pump Characteristics	S
6.B.4 Inadequate Core Cooling	S
6.B.5 DNBR, MCPR, etc.	S
6.B.6 Operational Analysis	S
8.0 RESPONSIBILITY AND AUTHORITY A. RADIATION PROTECTION CONTROL	
8.A.1 Source and Hazards of Radiation	S
8.A.2 Exposure Limits (10 CFR 20, Facility)	S
8.A.3 Portable Instrumentation (Knowledge and Use)	S
8.A.4 Procedures (RWP, containment entry, etc.)	S
8.A.5 Release Permits (gaseous, liquid, purge)	S
B. EMERGENCY PLAN IMPLEMENTING PROCEDURES 8.B.1 Duties	S
8.B.2 Classification	S
8.B.3 Evaluation Criteria	S
8.B.4 Personnel Assignments	S
C. ADDITIONAL DUTIES AND RESPONSIBILITIES 8.C.1 Surveillance Testing	
a. Instrumentation and Control	S
b. Other *(Specify)*	S
8.C.2 Security	S
8.C.3 Shift Turnover	S

COMMENTS *(Required for "U", use reverse)*

D. DISCUSSION *(Non-Power Reactors)* N/A	EVAL-UATION
THEORY OF NON-POWER REACTOR FACILITY OPERATION	
6.1 Reactivity Effects	
a. Subcritical Multiplication	
b. Dropped Rod/Stock Rod	
c. Coefficients	
d. Experiment Effects	
e. Long-Term Exposure Effects	
f. Rod Worth	
g. Shutdown Margin/Excess Reactivity	
h. Power Increases and Decreases	
6.2 Thermal Effects	
a. Core Cooling	
b. Heat Balance	
c. Instrumentation	
1 Radiation Protection Control	
a. Source and Hazards of Radiation	
b. Exposure Limits *(10 CFR 20, Facility)*	
c. Portable Instrumentation *(Knowledge and Use)*	
d. Procedures *(RWP, Contamination Control, etc.)*	
e. Release Permits *(gaseous, liquid, purge)*	
8.2 Emergency Plan Implementing Procedures	
a. Duties	
b. Classification	
c. Evaluation Criteria	
d. Personnel Assignments	
8.3 Additional Duties and Responsibilities	
a. Surveillance Testing	
b. Core Alterations	
1. Refueling procedures	
2. Fuel/Component Handling	
3. Fuel Component Storage	
c. Security	
d. Shift Turnover	

COMMENTS *(Required for "U", use reverse)*

NRC FORM 30? Pg 1

U.S. NUCLEAR REGULATORY COMMISSION

SIMULATOR EXAM REPORT

OPERATING EXAMINATION REPORT – PLANT OPERATIONS AND RESPONSE TO MALFUNCTIONS

DATE: 2/5/86

CANDIDATE: CHRISTOPHER BAKKEN III	FACILITY/SIMULATOR: Beaver Valley 1
Bakken / Clarke	Haser
EXAMINERS: Barber / Keller / Coe	Norris / Dudley

	A	B	C	D	E	F	G	H	I	J
SPECIFY IN EACH COLUMN IF CANDIDATE IS AN RO OR SRO DURING THIS EVENT →	PO	PO	PO		SF	SF	SF	SF		
SPECIFY THE INITIAL CONDITIONS →	A S/G Lvl high	Spurious CIA	ATWS		SIS Acc leak	Pump failed	VSST IC failed	Total loss of FW		
1. CONTROL BOARD AWARENESS	S	S	S		S	N/A	S	S		
2. EVENT DIAGNOSIS	S	S (3)	N/A		S	M (10)	S	S		
3. UNDERSTANDING OF INSTRUMENT RESPONSE	S	S	M (5)		S	S (11)	S	S		
4. EFFECTS OF MALFUNCTION	S	S	S		S	S	U (14)	M (13)		
5. COMMUNICATIONS	S	S	M (4)		M (6)	S	M (13)	S		
6. IMMEDIATE ACTIONS	S	S	S		S	S	S	S		
7. AUTOMATIC ACTIONS	N/A	S	S		N/A	N/A	S	S		
8. KNOWLEDGE OF REFERENCE DATA AND USE	N/A	N/A	N/A		S (9)	N/A	S	S (17)		
9. SUBSEQUENT ACTIONS	M (1)	N/A	S		S	S	S	S		
10. CONSOLE MANIPULATIONS	M (2)	N/A	S		N/A	N/A	S	N/A		
11. SUPERVISORY ABILITY IN A FOR RO	N/A	N/A	N/A		U (7)	S	S	M (16)		
12. USE OF PROCEDURES/TECHNICAL SPECIFICATION	S	N/A	S		S (8)	U (12)	S	M (15)		

COMMENTS:

1) Did not verify steam dump demand against actual stm flow

2) Very slow to attempt feeding A S/G caused manual reactor trip
Concerned only ~~w so~~ with excessive shrink while feeding

3) Quick verification of valid CIA based on valve stroking

4) Unaware that page party system was malfunctioning
Could have suggested alternative communications (walkie-talkie

5) Failed to check WR S/G lvl recorder when asked to,
check affect of AFW said "Couldn't tell if feeding."

6) Failed to notify SNSS of low level

7) Allowed PO attempt to fill Accumulator w/o authorization from RO

RECOMMENDATION: PASS

SIGNATURE EXAMINER: Barber

NRC FORM 305 Pg 2

US NUCLEAR REGULATORY COMMISSION

SIMULATOR EXAM REPORT

OPERATING EXAMINATION REPORT – PLANT OPERATIONS AND RESPONSE TO MALFUNCTIONS

CANDIDATE: A CHRISTOPHER BAKKEN III

FACILITY/SIMULATOR:

EXAMINERS:

SPECIFY IN EACH COLUMN IF CANDIDATE IS AN RO OR SRO DURING THIS EVENT →										
SPECIFY THE INITIAL CONDITIONS →										
	A	B	C	D	E	F	G	H	I	J
CONTROL BOARD AWARENESS										
EVENT DIAGNOSIS										
UNDERSTANDING OF INSTRUMENT RESPONSE										
EFFECTS OF MALFUNCTION										
COMMUNICATIONS										
IMMEDIATE ACTIONS										
AUTOMATIC ACTIONS										
KNOWLEDGE OF REFERENCE DATA AND USE										
SUBSEQUENT ACTIONS										
CONSOLE MANIPULATIONS										
SUPERVISORY ABILITY IN A FOR RO										
USE OF PROCEDURES/TECHNICAL SPECIFICATION										

COMMENTS

(8) Good review of T.S. to determine appropriate action

(9) PID used to determine alternate fill path for SIS Acc.

(10) Initially said "Load rejection in progress" with Pump failure

(11) Good determination to lower turbine load with high NI power from stm dumps actuation

(12) On both Pump inop. Said "T.S. 3.0.3 applied" vice action 12

(13) PO reported complete loss of AFW failed to acknowledge for 10 minutes.

(14) Never verified natural circulation

(15) Entered FRH 1 via E-0 step 19 instead of via Red Path conditions

RECOMMENDATION: PASS

SIGNATURE EXAMINER: [signature] / DL

NRC FORM 305 Pg 3

U.S. NUCLEAR REGULATORY COMMISSION

SIMULATOR EXAM REPORT

OPERATING EXAMINATION REPORT – PLANT OPERATIONS AND RESPONSE TO MALFUNCTIONS

CANDIDATE: CHRISTOPHER BAKKEN III

FACILITY/SIMULATOR:

EXAMINERS:

SPECIFY IN EACH COLUMN IF CANDIDATE IS AN RO OR SRO DURING THIS EVENT ⟶

SPECIFY THE INITIAL CONDITIONS ⟶

	A	B	C	D	E	F	G	H	I	J
CONTROL BOARD AWARENESS										
EVENT DIAGNOSIS										
UNDERSTANDING OF INSTRUMENT RESPONSE										
EFFECTS OF MALFUNCTION										
COMMUNICATIONS										
IMMEDIATE ACTIONS										
AUTOMATIC ACTIONS										
KNOWLEDGE OF REFERENCE DATA AND USE										
SUBSEQUENT ACTIONS										
CONSOLE MANIPULATIONS										
SUPERVISORY ABILITY IN A FOR RO										
USE OF PROCEDURES/TECHNICAL SPECIFICATION										

COMMENTS:

16 RO requested feed path direction and was not given the appropriate direction. Told RO to choose path

17 Very aggressive pursuit of inadequate feed flow. Checked PIDs and boards.

18 Unaware that SI reset would have removed FW isolation.

RECOMMENDATION: PASS

SIGNATURE EXAMINER: E Barber

EXIBITS

A. Chronology

B. Resume

C. Letter, DLS:AJM:210, dated March 28, 1983

D. Letter, DLS:AJM:231, dated April 22, 1983

E. Letter, DLS:AJM:253, dated 5/16/83

F. Letter, DLS:AJM:471, dated May 3, 1984

G. Letter, DLS:AJM:157, dated Nov. 29, 1982

H. Duquesne Light News, Oct. 1980 (Only the cover & pages 10 & 11 in the copies of the EXHIBIT book.)

I. EXCELLENCE OF OPERATIONS displays

J. License examination grade report prepared by Region 1

K. Lesson plan, LP-LRT-II-54, dated Jan 2, 1984 and Attachment 1

L. Letter from W.F.Kane dated Nov. 12,1986

M. BVPS-1-Updated FSAR, pages 10.3-2 & 10.3-5.

N. Design Change Package 189 report excerpts

O. BVPS-O.M. 1/2.48.2. page 8

P. Letter to W.T.Russell dated Dec. 16, 1986

Q. Letter from W.T.Russell, dated Feb 2, 1987

R. Station Administrative Procedure, Ch. 4, page 13 of 52

S. Letter to Harry B. Kister, dated Sept. 11, 1986

Case No.
55-6075

Official Exhibit No.
Morabito Exh Index

Disposition:
Rejected
IN TR

Morabito

Reporter:
FMEA

Date:
12/21/88

No. Pages:

T. NRC letter re: FOIA-87-151, dated March 24, 1987

U. BVPS - EOP, Executive Volume, page 65 of 140

V. Letter DLS:AJM:151, dated Aug. 9, 1982

W. Letter DLS:AJM:154, dated Nov. 1, 1982

X. Letter DLS:AJM:159, dated Dec. 9, 1982

Y. Notarized statement from Ms. Neuder

Z. Letter from Edward C. Wenzinger, dated Sept. 23, 1986

AA. Letter from Victor Stello, Jr. re: FOIA-87-A-26

BB. FOIA-87-352 Acknowledgement plus additional information

Chronology of Events

May 20, 1986	Duquesne Light Company submits application dated 5/20/86 for my license as a Senior Reactor Operator at the Beaver Valley Power Station by letter ND1DOT: 1991 dated 5/19/86.
July 22, 1986	NRC administers written exam
July 23, 1986	NRC administers operating exam consisting of: Simulator part Oral Plant Walkthrough
August 27, 1986	By letter dated 8/27/86, Region I informs me that I passed the operating test but did not pass the written examination or the simulator examination. Therefore, my application for a Senior Reactor Operator license was denied. The letter is signed by Harry B. Kister, Chief; Projects Branch No.1; Division of Reactor Projects. That same letter informed me that the provisions of Part 2, Title 10, Code of Federal Regulations, Section 2.103(b)(2) provide that an applicant has the right to request a hearing within 20 days from the date of the denial. The referenced section from 10CRF was attached to the letter.

Sept. 15, 1986 Laurie Heilman of NRC Region I signs for receipt of my letter dated Sept. 11, 1987 and mailed by certified mail on Sept. 12, 1987, in which I requested a hearing be held concerning the denial of my application for a Senior Operator license. The letter contained support for my opinions concerning why certain questions were graded incorrectly or too severely. The letter cited 10CRF2.103(b)(2) as the basis for my request.

Sept. 17, 1986 Harry B. Kister acknowledges receipt of my letter dated 9/11/86. He states that my request will be reviewed in accordance with NUREG-1021 dated April 15, 1986.

Nov. 12, 1986 By letter dated 11/12/86, William F. Kane informs me that a complete regrade of the written examination and a complete reevaluation of the simulator examination was conducted. No adequate basis was found for reversing the original decision. The denial of application remained in effect. I was further informed that I could pursue a hearing on the denial by informing Mr. W. T. Russell, Director, Division of Human Factors Technology in Washington, D.C. within 20 days of the date of that letter.

NOTE: I had originally passed the overall written examination with an 82.2%. 80% is required, but I had failed one of the four sections with a 59.7%. 70% is required on each section and therefore the exam was recorded as failed. In this regrade, the section I had failed was increased to 67.6% though other sections were reduced due to the regrade. Nevertheless, I still had an 80.6% overall passing grade.

NRC Comments on my comments on Simulator exam:

1. Agreed that I had insufficient time to refer to procedure when pressurizer level channel 459 failed.
2. Acknowledged that examiner made an incorrect statement in his report regarding position indication lights for RHR valve.

Dec. 1, 1986 By certified letter, I informed W. T. Russell of my wish to pursue a hearing of the denial of my application. This letter was received at NRC by N. W. Matovich on 12/3/86. I informed NRC that due to an unforseen circumstance, I cound not provide additional information at that time but I would in the near future.

Dec. 8, 1986 Theodore L. Szymanski, Acting Chief, Operator Licensing Branch, Division of Human Factors Technology, acknowledges the receipt of my letter of 12/1/86, though he doesn't refer to it by date. He stated that the processing of the appeal would be scheduled when the additional supporting information was provided. He says to call Len Wiens at 301-492-7735 if I had any questions regarding the processing of the appeal.

Dec. 22, 1986 My letter dated 12/16/86 is received by certified mail at NRC headquarters. The letter contained information to support my request for a hearing made in my letter of 12/1/86. In that letter I closed with a request for all information, including log sheets and examiners comment sheets, concerning my examination so that I could prepare for the next phase of the appeal process.

Jan. 5, 1987 Theodore L. Szymanski acknowledges receipt of my letter. He states that the review of my examinations would be completed by Jan. 22, 1987.

Feb. 2, 1987 By letter dated 2/2/87, B.A. Bogar, signing for W. T. Russell, upheld the denial of my license. I was informed that the appeal would be forwarded to the Office of General Counsel and that they would contact me concerning the details of the hearing process. I was told to contact Ted Szymanski if I had additional concerns on this matter.

Feb. 9, 1987 — I called Ted Szymanski to ask why my request for information concerning my exam had not been addressed in their letter of 2/2/87.

Len Weins called back. He suggested I contact Region I. I asked if Headquarters would contact Region I instead of me contacting them. He said he would have to talk to his boss. He said he would get back to me. I asked when I could expect to hear from the legal folks. He responded, "Soon, perhaps within 1 week. It won't be a call. It will be a letter." He told me to contact Dick Hoeffling at 301-492-7013 if no correspondence after about 1 week.

Feb. 10, 1987 — Len Wiens called back to give me two ways to get information - use FOIA - he gave me a contact name - or use legal "discovery". He gave me names of Jack Goldberg and Dick Hoeffling.

Feb. 10, 1987 — Talked to Dick Hoeffling. He outlined the hearing process. He informed me that I could get the documents which I requested by invoking the Freedom of Information Act.

Feb. 24, 1987 — I called Dick Hoeffling to find out the status of my appeal. He said that there was a letter coming that would acknowledge my desire to continue the hearing process. Based on this phone call, they would dispense with any further need of response from me and will move the papers on to the hearing commission.

Feb. 26, 1987 By certified letter dated 2/26/87, Mr. Jack R. Goldberg, Deputy Assistant General Counsel for Enforcement, Office of General Counsel, recounted the history of my requests for a hearing. He stated that steps would be taken to initiate a hearing. He stated that a Notice of Hearing would be sent to me as soon as it was issued. He further stated that I would have the Burden of Proof to demomstrate that I had met the requirements of 10CFR55.11(b). He identified himself as a contact.

March 6, 1987 I called Goldberg. I requested to know the time frame for publication of the Notice of Hearing and set up of the pre-hearing conference. He replied that my request had been forwarded to a hearing commission. They would refer it to the Atomic Safety & Licensing Board who would appoint an Administrative Law Judge. Time frame would probably be approximately one month. After publication of the Notice of Hearing, the process would start in weeks rather than months. He gave me the name of a contact in the Hearing Division:

Ben Vogler, phone number 301-492-7618

March 10, 1987	By certified mail, I requested copies of the notes and logs which I generated during my examination and a copy of the final response of the NRC to Ms. Sue Neuder, who had been the reactor operator during one of the scenarios for which I was the supervisor. This information was requested under the Freedom of Information Act (FOIA).
March 20, 1987	NRC holds teleconference with Duquesne Light Nuclear Group Management. Subsequent to the conference, it is suggested to me that I drop the request for hearing. When I responded that I would continue, I was told that I would receive a personal phone call from the Senior Vice president of the Nuclear Group.
March 22, 1987	By certified letter to Jack Goldberg, I informed him that I considered the phone call of March 20 irregular, unethical and an attempt to deny me rights that were mine under the law. I also informed him and Duquesne Light Co., by copy of the letter, that I did not intend to ask Duquesne to become involved in the proceedings; in fact, I asked them to remain neutral. The letter cautioned the NRC against harassment of Duquesne Light Co., and subsequently myself, through a perceived threat that I might lose my job or be demoted.

March 24, 1987	I received partial response to my FOIA request of 3/10/87. I was denied the letter to Ms. Sue Neuder because it would cause an unwarranted invasion of privacy. I was informed that I would need a signed notarized release from Ms. Neuder to get the letter. The request was identified as FOIA-87-151.
April 4, 1987	By certified mail, I requested records concerning the 3/20/87 teleconference between the NRC and Duquesne Light Company and further requested the letter to Ms. Neuder under the provisions of 10CFR9.6 which allowed deletion of private information from records to be released under FOIA requests.
April 6, 1987	Sue Neuder signed a release letter and had it notarized. I did not use this release as of 6/20/87.
April 8, 1987	I received a letter form Benjamin H. Vogler, Senior Supervisory Trial Attorney attempting to justify the March 20 teleconference and again stating that the Notice of Hearing would be sent to me as soon as it was issued. He again noted that I had the Burden of Proof in this matter. He identified himself as future contact point.

April 16, 1987	By certified letter received on 4/20/87, I filed charges with the Director of the Office of Inspector and Auditor. I identified the appropriate areas of 10CRF which I contend were violated by the NRC when they made the March 20 phone call to Duquesne which then precipitated a discussion with me which I perceived as a significant threat and which diminished my faith in the integrity of my government.
April 22, 1987	I was contacted by Ms. Carol Kagan of the Office of the General Counsel. She wanted to know whether I wanted a formal or informal hearing. When I questioned what differences there were, she was very unsure. She did state that in an informal hearing I would not have subpoena or discovery rights. I then informed her that I wanted a formal hearing since I need the discovery process to gain access to various documents.
May 5, 1987	My FOIA request of 4/4/87 was identified as FOIA-87-202. In response to that request, the NRC stated that there were no requested records concerning the March 20 teleconference. They again denied my request for the NRC's letter to Ms. Neuder under provisions of the Privacy Act.

May 15, 1987 By certified letter received on 5/20/87, I filed an appeal with the Executive Director for Operations requesting that under the provisions of 10CFR 9.5(b), 10CFR9.6 and 5USC552 the letter to Ms. Neuder dated 1/21/87 be released to me or at least a memorandum which acknowledged the existence of the NRC's acknowledgement to Ms. Neuder that they incorrectly judged her actions regarding reactor coolant boration.

May 29, 1987 I called Ms. Carol Kagan, NRC General Counsel, to request status of my request for hearing. She informed me that it still hadn't been resolved as to whether it would be an evidentiary hearing or some other type. She felt it would be resolved within 1 or 2 weeks. I informed her that I request the Notice of Hearing be published by June 30 or I would send copies of everything that has transpired regarding these proceedings to my Congressional Representatives.

June 2, 1987 By certified letter to Ms. Carol Kagan, I documented my request that the Notice of Hearing be published by June 30, 1987. I stated my belief that I was entitled to the hearing process defined by 10CFR 2.104(a)(1), (2)(3) and (4). I also noted that 10CFR 2.104 (c) requires the presiding officer to consider any matters in controversy among the parties.

July 1, 1987 The Commissioners issue an Order granting me a hearing. The Order specifies that there will be a single Presiding Officer who will be chosen from among the Atomic Safety and Licensing Board Panel. It further specifies that the hearing will be an informal adjudication and that the requirements of 10 CFR Part 2, Subpart G do not apply.

AL MORABITO

Station Chemist, Shippingport, 6/66 - 11/69.
Performed chemical analyses on reactor coolant to ensure proper chemical constituents for corrosion resistance. Analyzed for levels of fission and activation products. Analyzed for specific radioisotope to verify power levels and cladding integrity. Analyzed secondary waters to ensure optimum corrosion and purity standards. Advised station personnel on the safe handling of chemicals.

Station Operating Foreman, Shippingport, 1/71-8/71
Supervised a crew of reactor plant and turbine plant operators during normal operating conditions and during maintenance shutdowns and startups. Responsible for emergency squad and first aid. Supervised trainees through many reactor plant/turbine plant startups and shutdowns.

Shift Reactor Engineer, Shippingport, 9/71 - 5/72
Responsible for reactor testing program, operating procedure revision, radioactive waste disposal operations, reactor instrumen calibrations and periodic test.

Training Engineer, Shippingport, 5/72 - 12/73
Responsible for training operations personnel on Reactor theory, heat transfer, procedure performance administrative responsibilities and emergency procedures.

Training Supervisor, Shippingport, 12/73 - 2/80
Supervised instructors and engineers to accomplish the Training of operations personnel, radiation control technicians, and maintenance personnel. Acted as a consultant to the Onsite Safety Committee.

Results Coordinator, Shippingport, 2/80 - 5/82
Responsible for technical advice to operations. Supervised the testing and engineering and chemistry groups. Planned testing and maintenance outages. Responsible for Emergency preparedness plans, procedures, and facilities. Supervisor of the Emergency Control Center. Liaison with off-site technical support. Member of Onsite Safety Committee.

Chief Engineer, Shippingport, 6/82 - 12/82
Responsible for all aspects of station operation.

Superintendent, Shippingport, 1/83 - 8/84
Overall responsibility for Duquesne Light Company operation of the Shippingport Atomic Power Station. Represented the Company in its relations with the U.S. Government Oversaw the orderly shutdown and defueling of the Light Water Breeder Reactor. Chaired the Onsite Safety Committee. Guided the turnover of the station to the decommissioning contractor.

March 28, 1983
DLS:AJM:210

Defueling Procedure Compliance

TO: All Defueling Supervisors

I have become aware that some of you do not define procedure compliance the same as I do. Although your definitions differ from mine, your actions in complying with procedures have generally been consistent with my intentions. Nevertheless, unless all Duquesne personnel at Shippingport interpret procedure compliance with the same word definition and mental interpretation, someone will eventually apply the compliance criteria too loosely and create a violation. The following criteria is my direction to you regarding a uniform understanding of and application of procedure compliance:

> Defueling procedures are to be followed verbatim - that means word for word, sentence by sentence, step by step, in the sequence written.
>
> If a procedure cannot be followed as written, stop; ensure that the plant or equipment is in a safe, stable condition; change the procedure in accordance with the applicable instructions.
>
> Verbatim procedure compliance does not preclude the use of good work practice and common sense. Some procedures are written such that they give only general direction to perform certain actions. In those cases it is expected that you will rely on good work practices, experience, and common sense to perform the work. The method you use may be different than the method used by another supervisor. Other procedures are written such that specific direction on exactly how to perform certain actions and what tools to use is given. In those cases you cannot substitute a different method. Any questions on procedure interpretation must be directed to the Joint Defueing Group. They have the final authority for resolution of interpretation.
>
> There are three (3) variances to verbatim procedure compliance criteria that have been authorized by site Management. These are:
>
> 1. When the wrong size standard hand tool is incorrectly called for in a procedure, you may substitute the proper tool without processing a procedure change.
> 2. Procedure work may continue even though material is not staged in accordance with the numbers required by the staging list.

3. When a procedure contains an obvious typographical error but the intent of the procedure is understood by the craft personnel, the Defueling Shift Supervisor and foremen, and the Shift Defuleing Engineer, the procedure may be performed without requiring a procedure change.

You may rely on these criteria to resolve procedure compliance problems at the work site. In any other instance of compliance problems a formal change must be processed.

A. J. Morabito

A. J. Morabito
Station Superintendent

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

April 22, 1983
DLS:AJM:231

Portable Radiation Monitors

Mr. F. R. Huey, Manager
Shippingport Branch Office
U.S. Department of Energy
Shippingport, PA 15077

Reference: (a) NR:RR:DEFry G#7442 dated April 8, 1983

Dear Mr. Huey:

In Reference (a) Naval Reactors approved the use of portable radiation monitors during head area disassembly operations and during fuel handling operations. They also approved a course of action in response to an alarm from these instruments which is as follows:

> Radiological controls personnel would first verify that the alarm is valid, then require personnel not involved in the defueling operations to evacuate the canal area, frisking before departure. Radiological controls personnel would then determine what recovery actions are required and whether other personnel should evacuate the canal area. Radiological controls personnel should actuate the manual RBMS alarm (as discussed in Naval Reactors memorandum G#7306 dated March 2, 1983) when a rapid and complete evacuation of the Fuel Handling Building is necessary.

This action is not consistent with the training that personnel at Shippingport have received regarding the action to take in response to a Refueling Building Monitoring System alarm. It is also not conservative because it assumes that the most likely cause of the alarm is the lifting of a radioactive component too close to the canal water surface and; therefore, the obvious corrective action is simply to lower the component deeper into the water. It also assumes that the event is under control and that time is available to perform an evaluation.

It is Duquesne's intention to use at least two Radiation Technicians to monitor the movement of highly radioactive components and fuel. The occurrence of a portable radiation monitor alarm under these conditions implies that only one of two situations could exist. These are:

1. The alarm is the spurious result of instrument malfunction or stray radioactivity.

2. The alarm is a true response to an actual increase in radiation levels from the handling of reactor components or fuel.

In either case, the alarm signifies a breakdown of the radiological controls that have been procedurally and managerially established. Under these conditions it does not make sense to have the Radiation Technicians, who were responsible for the breakdown of the established controls, evaluate the cause of the alarm and determine who should evacuate the area and who should stay. Moreover, the occurrence of the alarm should require immediate Management reaction to determine the cause and regain the proper controls. In addition, there remains the overriding concern that in spite of alert personnel, properly functioning instrumentation, and conservative estimates of radiation levels an actual high radiation field could occur and be detected simultaneously by the Radiation Technicians and the portable instrumentation. Under those circumstances it may not be obvious what caused the condition or what immediate action should be taken to correct the condition.

For all of these reasons, the proper reaction to a Refueling Building Monitoring System alarm or a portable radiation monitor alarm is to evacuate through the nearest exits without frisking, assemble at an approved location, and evaluate the corrective action needed to control the situation.

Naval Reactors is requested to revise the comments in Reference (a) to be consistent with the above stated, preferred reaction to a portable radiation monitor alarm.

Yours truly,

A. J. Morabito

A. J. Morabito
Station Superintendent

cc: T. D. Jones
G. T. Howard
J. M. Crum
C. K. Schultz
D. M. DiNuzzo
V. F. Kraker
M. G. Haydin
W. E. Strayhorn
A. D. Konopka
Central File

5/16/83
DLS:AJM:253

Improved Control of Defueling Operations

Messrs: G. T. Howard
F. R. Huey
J. A. Redfield

I consider that our recent lack of success in removing the reactor vessel head on schedule and the extensive lack of productivity during the decontamination of the core support flange are indications that we need a better way of doing things. Accordingly, I met with Messrs. Crum, DiNuzzo, Haydin, Konopka, Kraker and Schultz to define our problems and develop solutions. I am implementing those corrective actions that are appropriate for Duquesne Light. I ask that you do the same in your organizations. The following is provided for your information:

1) Problem: Defueling Supervisors are mentally weary. They have had extended 6 day work weeks over several months. This mental lethargy has led to a number of oversights and a reduced ability to plan ahead.

Solution: Additional supervisors will be qualified as defueling supervisors. They will commence a rotating schedule with the current supervisors who will subsequently be assigned to non defueling jobs for short intervals. In addition, consideration will be given to suspending continuous 6 day work weeks and conducting weekend planning sessions.

2) Problem: There is a lack of understanding regarding why we do what we do.

Solution: Messrs. Brunner and Massimino will meet to discuss reasons behind the various defueling rules and controls. Consideration will be given to presenting live lectures on small items controls and cleanliness controls to defueling personnel. Workers will be given an opportunity to get clarification on reasons behind various operations.

3) Problem: The personnel at the station see an apparent inconsistent application of requirements that have previously been inviolate.

Solution: Existing rules are being re-interpreted in light of the fact that Shippingport is being decommissioned, In the future, an explanation of the reason's behind rule changes will be available to station personnel. When possible, supervisors will provide the explanations when the changed rule is initially applied.

4) Problem: Lack of aggressive persuit of responsibility.

Solution: A) Defueling personnel will be required to inform their supervisors immediately when the work effort slows down. In addition, D & S Management is requested to require their SDE's to immediately inform the Chief Defueling Engineer when Duquesne's work effort is in trouble, either because of procedure problems or because of lack of materials or because of lack of supervisory attention. The Chief Defueling Engineer must then involve the Manager, LWBR D & S and the Superintendent.

B) An additional Duquesne supervisor will be assigned to the defueling effort. His purpose will be to tie all loose ends together; be the contact to whom all deficiencies or problems are initially referred; and be the expediter of the solutions to those problems.

C) Greater emphasis will be placed on meeting scheduled work efforts and holding supervisors accountable when scheduled work is not completed on time.

5) Problem: There is a lack of meaningful communication amoung DLC, SBO, and Bettis.

Solution: Pre-shift meetings have been organized. These meetings will be attended by the NSS, DSS, SDE and SBO Representatives. The intent of the meeting is to establish better working relationships.

Yours truly,

A. J. Morabito

A. J. Morabito
Station Superintendent

cc: T. D Jones
F. X. Bayer
Staff
Central File

May 3, 1984
DLS:AJM:471

Priority and Security Equipment Repair - Revision 5

TO:	T. A. Porter	A. M. Moody	J. M. Crum
	G. VanSickle	D. M. DiNuzzo	M. J. Weiss
	A. H. Brunner	M. S. Helms	D. A. Wacker
	A. D. Konopka	R. N. Boyle	D. C. Reeves

This letter supersedes DLS:AJM:309 dated August 4, 1983 and establishes priority repair for environmental, radiological, defueling, and security equipment. Repair should commence in accordance with the following schedule unless waived by the Superintendent or his designated representative.

PRIORITY EQUIPMENT

Items requiring immediate repair:

ORMS 12, 12S and 12 Backup (2 AMS-2) when two of four are unavailable
ORMS 13 Fixed Filter
Fixed Filter Stations 14, 15, 16
Pressure Recorder (43-H1-3)
RBMS and Backups (less than two available)

Other priority items:

Neutralizing Tank pH Inst.	Next normal working day Chemistry samples required during interim
2 H.P. Counters	Next normal working day when both are unavailable.
Turbine Room Oil Separator	Next normal working day
ORMS 8, 9, 10 and 11	Next normal working day
BCMS 1*, 2*, 3, 4, 5, and 6	Next normal working day

*Not required after Reactor Pit gate installed.

SECURITY EQUIPMENT

Explosive Detector	Next normal working day
Duress Alarm	Next normal working day
Fence Alarm	Next normal working day

While the list represents items which are important for environmental, defueling, radiological, and security records, it is not intended to be all inclusive. Other equipment failures should be evaluated and repaired according to need. Station organizations responsible for the use or operation of priority equipment are also obligated to report deficiencies immediately to the Maintenance Organization and follow up with a Work Item Card. Reporting of priority items noted above should be made directly to the Station Maintenance Supervisor.

A. J. Morabito

A. J. Morabito
Station Superintendent

cc: NSS
Captain of Guards

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

November 29, 1982
DLS:AJM:157

1983 Station Objectives

TO: T. D. Jones

The 1983 objectives for the Shippingport Atomic Power Station are:

1. Maintain a zero lost time accident record.
2. Improve the radiological awareness of Station personnel thereby reducing radiological incidents.
3. Achieve the 1983 Defueling schedule objectives ahead of schedule.
4. Place the turbine plant in layup status and disposition salvageable equipment.
5. Develop and implement a plan for the disposition of Station operating records.
6. Continue the Excellence of Operations concept.
7. Improve the in-house surveillance program.

Attached for your information are the Station group objectives.

A. J. Morabito

A. J. Morabito

Attachment

cc: J. F. Zagorski
A. J. Morabito
W. E. Strayhorn
V. F. Kraker
J. M. Crum
M. G. Haydin
L. R. Freeland
R. G. Williams
D. M. DiNuzzo
A. L. Couper
Central File

Attachment A to DLS:AJM:157
dated November 29, 1982

1982 Shippingport Atomic Power Station Group Objectives

General Office

1. Review and eliminate reports and various forms not applicable during the SAPS Defueling and Decommissioning period.

    a. Department of Energy reports and forms

    b. Duquesne Light reports and forms (includes Nuclear Division required interface)

2. Review Station Manual distributions for minimum copies required during the Defueling and Decommissioning period. Obsolete copies will be retired.

    a. Department of Energy related manuals

    b. Duquesne Light related manuals (includes Nuclear Division activities)

Security

1. Update the Station Security Indoctrination presentation.

    a. Badging requirements

    b. Security rules and regulations

2. Review of Security record-keeping activities for applicability and need.

    a. Daily records (personnel)

    b. Training records (personnel)

    c. Equipment records

Maintenance

I. Operational Goals:

1. Complete head removal before the scheduled date of April 7, 1983.

2. Perform six M-130 fuel shipments by year end.

3. Replace the deteriorated treated water line in the FHB by the end of April, 1983.

4. Maintain zero lost time accidents and reduce the number of other reportable accidents in the Maintenance group by one half of those reported in 1982, by stressing safe working practices in weekly safety meetings.

II. Non-Operational Goals:

1. Remove all stored material from the site which is no longer required to support Defueling operations.

2. Maintain the FHB in a safe, efficient condition by stressing good housekeeping and using close follow-up of Defueling activities by Maintenance supervisors.

3. Complete the lay-up of out-of-service turbine plant systems and remove all available salvageable equipment.

Training

1. Prepare Shutdown Training lesson plans to support a new session every six weeks.

2. Prepare Continuous Radiation Worker Training lesson plans to support a new session each quarter.

3. Conduct on-shift casualty drills each month.

4. Transfer records that no longer need to be kept at the Station to a storage facility.

5. Dispose of training manuals and books that are no longer needed.

6. Conduct safety meetings each week.

Operations

1. Qualify all Nuclear Operators in all attendant positions.

2. Revise Operating Memorandum 010 to be consistent with "shutdown for defueling" conditions.

3. Prepare and institute a training qualification for the safe and efficient operation of the thin film evaporator.

4. Discharge 30,000 gallons of RWP liquid prior to start of the canal boration sequence or as necessary to ensure sufficient space is available at RWP to receive non-borated canal tank effluent.

5. Reduce to zero the number of 1983 High Radiation Door Incidents that are a result of Operations personnel deficiencies. This will support an overall Station goal of no such incidents.

Planning and Scheduling

1. Meet the key dates given in the Master Activity Schedule (DLS:MGH:084 dated October 30, 1982).

2. Ensure Plan of Actions for unexpected problems that arise during the Defueling period are developed and implemented. Take the lead in implementing the Master Activity Schedule via use of Plan-of-the-Week and Plan-of-the-Day scheduling.

3. Follow daily progress of Defueling operations and plan Defueling support activities. Conduct daily planning and scheduling meetings to keep work on schedule and ensure open items are identified and resolved in a timely manner.

4. Plan and schedule the Inactivation of Government Systems as requested by the Joint Support Group, consistent with manpower availability, during 1983.

5. Complete Planning & Scheduling support for the inspection of the 1D steam generator as identified in the Master Activity Schedule.

6. Complete Planning & Scheduling support for the salvage of the 1650 KW diesel generators and associated support equipment as identified in the Master Activity Schedule.

Quality Control

1. Zero audit findings to be accomplished by group meetings and internal checks and balances.

2. Limit personnel radiation exposure to stay below 1983 exposure estimate.

3. Transfer all nonessential records to the General Office in preparation for plant closing.

4. Train all QC personnel in appropriat Defueling instructions.

Radiological Control

1. Limit total DLCo Station man-rem to 93.0 rem. This Station man-rem objective is higher than recent annual man-rem objectives to account for the increased personnel exposures anticipated as a result of the LWBR Defueling Program scheduled for 1983. This objective was determined using exposure records generated during previous shutdowns and refuelings adjusted for differences in work planned during the LWBR Defueling Program.

2. The requalification of one Radiation Technician whose qualification expires in 1983.

3. Revise the theory portion of the Initial Radiation Worker Training tapes along the guidelines of Regulatory Guide 8.27 issued March, 1981.

4. Maintain zero skin contaminations in 1983.

5. Continue to minimize the number and size of contaminated areas.

6. Return to zero high radiation area incidents in 1983.

Results Coordinator

1. a. Since manpower is being reduced in the technical service groups, especially Testing, it will be necessary to train people from several groups to manage the Emergency Control Center when the Emergency Plan is implemented.

   b. An additional goal concerning the Emergency Plan will be to provide training exercises and drills that adequately reflect plant conditions during Defueling.

2. The ability to draft water from the Ohio River into the SAPS fire system has been demonstrated. An objective for 1983, will be to purchase all necessary equipment, install equipment and weather protection and maintain, as necessary, the ability to draft water.

3. All personnel still assigned to the Technical Service Groups after December 31, 1982, should view the Defueling Training Tapes (especially the "overview") as time permits. This will be informative to them and should help in the overall Station objective to defuel the LWBR core.

Chemistry

1. Take appropriate measures to have all extraneous mercury removed from the Station.

2. Increase personnel knowledge of radiochemistry theory, counting equipment and techniques.

Engineering

1. Increase personnel awareness of the requirements in the Quality Control manual. This will aid in resolution of problems denoted by Quality Control Non-Conformance and Corrective Action Reports (NCAR's).

2. Disposition of all NCAR's received within seven days. In the past year, most resolutions to NCAR's were prepared within seven days. However, they were often delayed in the approval circuit for several weeks. The goal is to complete approval of the resolution within seven days.

Test Engineering

1. Now that the formal testing program is completed, the test files will be removed from site. The goal is to aid in the effective disposition of all test data files.

2. With the reduced amount of instrumentation now required, some systems (example - DAS) will no longer be needed. Goal is to aid in the retirement of no longer used plant instrumentation.

Safety and Budget Administrator

1. Perform daily plant tours.

2. Establish plant conformance to hearing protection requirements.

3. Maintain overtime expenditures within the overtime budget.

4. Achieve zero lost time accidents.

5. Reduce the number of OSHA reportable accidents.

6. Perform monthly trend analysis of accident reports.

Surveillance Coordinator

1. Expedite action on surveillance trend report open items, so that these items are not carried on this report more than three months.

2. Perform a monthly review of the Emergency Control Center for its readiness in the event of an emergency.

3. Job Order Goals:

   a. Maintain or better a two-week turnaround for review and return of comments, or sign-off of Job Order drafts.

   b. Maintain or better a one-week turnaround for review and sign-off or submittal of comments on Job Orders submitted for initial issue.

   c. Maintain or better a four-day turnaround for review and sign-off, or submittal of comments on Job Orders submitted for addendum changes.

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

May 25, 1983
DLS:AJM:256

1983 Shippingport Atomic Power Station Objectives
Second Quarter Progress Report

Mr. T. D. Jones:

The Second Quarter, 1983 Shippingport Station Objectives (Attachment A) and the Group Objectives (Attachment B) is submitted for your consideration.

Also forwarded for your information are the following attachments: (1) a progress report on the Staff Inspection Program, (Attachment C); (2) an updated report on LWBR defueling operations and Turbine Plant salvage activities (Attachment D), and (3) a revised long term inactivation schedule for the Reactor Plant systems, (Attachment E).

A. J. Morabito

A. J. Morabito
Station Superintendent

JJP/kmb

Attachments

cc: J. J. Carey
A. D. Konopka
C. K. Schultz
J. M. Crum
W. E. Strayhorn
M. G. Haydin
D. M. DiNuzzo
V. F. Kraker
J. J. Peters (2)
Central File

1983 Shippingport Atomic Power Station Objectives
Second Quarter Report

Objective #1 - Maintain a zero lost time accident record

Status: No lost time accidents have occurred during the second quarter.

Objective #2 - Improve the radiological awareness of Station personnel thereby reducing radiological incidents

Status: One radiological Control incidect occurred during the second quarter. This incident involved personnel failing to follow a written procedure, which resulted in a localized above limits, airborne condition. Improved pre-shift briefings (concerning defueling work), Radcon memos, and discussions at the quarterly Radcon training sessions, are being used to increase the station radiological awareness to avoid future incidents.

Objective #3 - Achieve the 1983 Defueling schedule objectives ahead of schedule

Status: Bettis completed checkpoint 2A work and issued the prerequisite list for checkpoint 3 on May 1, 1983, 3½ weeks behind the MAS schedule. Duquesne removed the Reactor Vessel Closure Head two days after start of the checkpoint, as scheduled.

Corrective actions to recover the schedule are being taken including working weekends when controlling path time can be made up. The goal of having all fuel removed from the site by June, 1984, is still considered achievable.

Objective #4 - Place the turbine plant in layup status and disposition salvageable equipment

Status: Turbine plant layup is complete. Disposition of salvageable equipment is 75 percent complete.

Objective #5 - Develop and implement a plan for the disposition of Station operating records

Status: A plan for the disposition of Station operating records has been developed. Implementation will begin following PNR approval of the plan.

Objective #6 - Continue the Excellence of Operations concept

Status: A one hundred (100%) achievement was achieved during the second quarter. Station Personnel have also shown an increased interest in the quarterly EOO displays.

Objective #7 - Improve the in-house surveillance program

Status: A Staff Inspection Program is in progress. The cited deficiencies are followed daily in the Plan-of-the-Day (POD) meetings; and each month, a summary report of the deficiencies and resolution is issued to staff members. A progress report on the Staff Inspection Program is included (Attachment C) for information.

1983 Shippingport Atomic Power Station Group Objectives Second Quarter Progress Report

General Office

1. Review and elminate reports and various forms not applicable during the SAPS Defueling and Decommissioning period.

   a. Department of Energy reports and forms

   b. Duquesne Light reports and forms (includes Nuclear Division required interface)

   Status: 50% complete. Review is continuing.

2. Review Station Manual distributions for minimum copies required during the Defueling and Decommissioning period. Obsolete copies will be retired.

   a. Department of Energy related manuals

   b. Duquesne Light related manuals (includes Nuclear Division activities)

   Status: 50% complete. Review is continuing.

Security

1. Update the Station Security Indoctrination presentation.

   a. Badging requirements

   b. Security rules and regulations

   Status: 50% complete. Draft presentation has been prepared.

2. Review of Security record-keeping activities for applicability and need.

   a. Daily records (personnel)

   b. Training records (personnel)

   c. Equipment records

   Status: 60% complete. To date forms have been reduced by 60%.

Maintenance

I. Operational Goals:

1. Reactor Vessel Closure Head removal before the scheduled date of April 7, 1983.

   Status: Reactor Vessel Closure Head was removed on May 3, 1983.

2. Perform six M-130 fuel shipments by year end.

   Status: Defueling operations are 3½ weeks behind the MAS Schedule and as a result six (6) M-130 fuel shipments may not be achieved.

3. Replace the deteriorated treated water line in the FHB by the end of April, 1983.

   Status: This item is 100% complete.

4. Maintain zero lost time accidents and reduce the number of other reportable accidents in the Maintenance group by one half of those reported in 1982, by stressing safe working practices in weekly safety meetings.

   Status: The Maintenance department has achieved zero lost time accidents for the second quarter. There have been four (4) reportable accidents in the first 4 months of 1983 vs 2 for the same period in 1982. Maintenance supervision will continue to emphasize safe work practices to reduce this number.

II. Non-Operational Goals:

1. Remove all stored material from the site which is no longer required to support Defueling operations.

   Status: During the second quarter, stored material was removed and disposed of from the Wampum Mine. The Maintenance Departments goal is to have all material removed from the mine by June 30, 1983. This goal is approximately 85% complete. The goal of removing all stored material from the site which is no longer required to support defueling operations is approximatly 30% complete.

2. Maintain the FHB in a safe, efficient condition by stressing good housekeeping and using close follow-up of Defueling activities by Maintenance supervisors.

   Status: This is a continuous effort by maintenance supervision.

3. Complete the lay-up of out-of-service turbine plant systems and remove all available salvageable equipment.

   Status: All turbine plant systems currently out-of-service have been layed up. All available salvable equipment has been removed. During the second quarter the charging pumps, boiler feed water regulating valves and various turbine plant instumentation were removed for salvage.

Training

1. Prepare Shutdown Training lesson plans to support a new session every six weeks.

   Status: 50% complete.

2. Prepare Continuous Radiation Worker Training lesson plans to support a new session each quarter.

   Status: 50% complete.

3. Conduct on-shift casualty drills each month.

   Status: 25% complete.

4. Transfer records that no longer need to be kept at the Station to a storage facility.

   Status: 25% complete.

5. Dispose of training manuals and books that are no longer needed.

   Status: Ongoing.

6. Conduct safety meetings each week.

   Status: 35% complete.

Operations

1. Qualify all Nuclear Operators in all attendant positions.

   Status: 90% complete.

2. Revise Operating Memorandum #010 to be consistent with "shutdown for defueling" conditions.

   Status: 100% complete.

3. Prepare and institute a training qualification for the safe and efficient operation of the Thin Film Evaporator.

   Status: A training program has been developed and implemented. The Thin Film Evaporator was operated safely and effectively during the three week period from 4/11/83 to 4/29/83 averaging approximately 136 gal/hr. of dischargeable water for actual running time.

4. Discharge RWP liquid prior to start of the "Canal Boration Sequence" to ensure sufficient space for receiving non-borated Canal Tank effluent.

   Status: 100% Complete.

5. Reduce to zero the number of 1983 High Radiation Door Incidents that are a result of Operations personnel deficiencies. This will support an overall Station goal of no such incidents.

   Status: Operations has supported the overall Station goal of having no such incidents.

Planning and Scheduling

1. Meet the key dates given in the Master Activity Schedule (DLS:MGH:084 dated October 30, 1982).

   Status: Satisfactory

   Checkpoint No. 2, Breach the Primary Boundary was completed on January 6, 1983, within two days of the Master Activity Schedule date of January 4, 1983. Checkpoint No. 3, Completion of DLC Head Area Disassembly was delayed two weeks due to unexpected repairs of the Fuel Handling Building Main Crane (which required a complete second load test). In DLS:MGH:101 dated March 16, 1983, DLC Planning & Scheduling recommended methods to recover at least two weeks of the schedule by Checkpoint No. 5, the start of Fuel Handling Operations.

2. Ensure Plan of Actions for unexpected problems that arise during the Defueling period are developed and implemented. Take the lead in implementing the Master Activity Schedule via use of Plan-of-the-Week and Plan-of-the-Day scheduling.

   Status: Ongoing - Satisfactory

   Preparations of plant systems for Defueling were scheduled early so that problems could be identified and resolved promptly on a not-to-impact-Defueling basis. The Plan-of-the-Week and Plan-of-the-Day meetings are instrumental in identifying what actions are required for timely resolution of problems.

3. Follow daily progress of Defueling operations and plan Defueling support activities. Conduct daily planning and scheduling meetings to keep work on schedule and ensure open items are identified and resolved in a timely manner.

   Status: Ongoing

   Currently holding daily Plan-of-the-Day meetings to monitor the progress of Plan-of-the-Week schedules.

4. Plan and schedule the Inactivation of Government Systems as requested by the Joint Support Group, consistent with manpower availability, during 1983.

   Status: Ongoing - Safisfactory

   Manpower will be scheduled in accordance with availability to provide support of the Systems Inactivation Schedule issued by the Joint Supervisory Group (JSG). Open items for issuance of software for support of system inactivations are followed in the Plan-of-the-Week meetings.

5. Complete Planning & Scheduling support for the inspection of the 1D Steam Generator as identified in the Master Activity Schedule.

   Status: Upcoming

   A pre-inspection meeting was successfully conducted on March 22, 23 and 24, 1983. Drainage of the '1D' loop is scheduled to begin on June 27, 1983 to support the July inspection of the '1D' Steam Generator.

6. Complete Planning & Scheduling support for the salvage of the 1650 Kw Diesel Generators and associated support equipment as identified in the Master Activity Schedule.

   Status: Ongoing - Satisfactory

   Final modifications to the security fence are scheduled to be completed by June 1, 1983, to support the diesel generator salvage scheduled to begin in June. Meetings will be conducted and support provided as necessary to ensure that any problems that arise can be resolved promptly.

Quality Control

1. Zero audit findings to be accomplished by group meetings and internal checks and balances.

   Status: Meetings are held weekly. Objective 50% complete.

2. Limit personnel radiation exposure to stay below 1983 exposure estimate.

Status: At the present rate, the QC department will be well below the 1983 exposure estimate. Objective 50% complete.

3. Transfer all nonessential records to the General Office in preparation for plant closing.

   Status: Objective 50% complete.

4. Train all QC personnel in appropriate Defueling instructions.

   Status: All QC inspectors have received training and have passed written tests on the appropriate Administrative Refueling Instructions. Objectives 100% complete.

Radiological Control

1. Limit total DLCo man-rem to less than the revised DLCo Man-Rem Budget of 49.371 Rem which was adopted on April 13, 1983 (DLS:MDM:298). This station man-rem objective is higher than recent annual man-rem objectives to account for the increased personnel exposures anticipated as a result of the LWBR Defueling Program scheduled for 1983. This objective was determined using exposure records generated during previous shutdowns and refuelings adjusted for differences in work planned during the LWBR Defueling Program.

   Status: Total DLC man-rem as of May 9, 1983 was 15.9 rem. The first and second quarter combined total DLCo man-rem is projected to be approximately 21.5 rem. This total is for station personnel who are included in the DLCo man-rem budget program and does not include DOE personnel, visitors, or Westinghouse personnel who are on a separate man-rem budget. The estimated 21.5 rem represents approximately 88% of the allotted first half-year man-rem budget.

2. The requalification of one Radiation Technician whose qualification expires in 1983.

   Status: The requalification of one Radiation Technician was completed on February 10, 1983.

3. Revise the theory portion of the Initial Radiation Worker Training tapes along the guidelines of Regulatory Guide 8.27 issued March, 1981.

   Status: The final draft of proposed revisions to the theoretical portion of the Radiation Worker Training tapes has been completed. As time permits, the theoretical tapes will be revised to incorporate these revisions.

4. Maintain zero skin contaminations in 1983.

   Status: There were no skin contaminations to date of this report.

5. Continue to minimize the number and size of contaminated areas.

   Status: There were no significant changes in the number or size contaminated areas to date of this report.

6. Return to zero high radiation area incidents in 1983.

   Status: There was zero High Radiation Area incident during the Second quarter.

Results Coordinator

1. a. Since manpower is being reduced in the technical service groups, especially Testing, it will be necessary to train people from several groups to manage the Emergency Control Center when the Emergency Plan is implemented.

      Status: Training is ongoing.

   b. An additional goal concerning the Emergency Plan will be to provide training exercises and drills that adequately reflect plant conditions during Defueling.

      Status: The first exercise was conducted in April.

2. The ability to draft water from the Ohio River into the SAPS fire system has been demonstrated. An objective for 1983, will be to purchase all necessary equipment, install equipment and weather protection and maintain, as necessary, the ability to draft water.

   Status: The necessary equipment has been purchased and is staged at the screenhouse. 100% complete.

3. All personnel still assigned to the Technical Service Groups after December 31, 1982, should view the Defueling Training Tapes (especially the "overview") as time permits. This will be informative to them and should help in the overall Station objective to defuel the LWBR core.

   Status: This review in ongoing.

Chemistry

1. Take appropriate measures to have all extraneous mercury removed from the Station.

   Status: All mercury from the Chemistry Lab and unused Turbine Plant components has been sent to Manchester. Other plant mercury will be inventoried and removed from the site as conditions permit.

2. Increase personnel knowledge of radiochemistry theory, counting equipment and techniques.

   Status: Program is 35% complete.

Engineering

1. Increase personnel awareness of the requirements in the Quality Control manual. This will aid in resolution of problems denoted by Quality Control Non-Conformance and Corrective Action Reports (NCAR's).

   Status: Ongoing

2. Disposition of all NCAR's received within seven days. In the past year, most resolutions to NCAR's were prepared within seven days. However, they were often delayed in the approval circuit for several weeks. The goal is to complete approval of the resolution within seven days.

   Status: Four of eleven NCAR's were completed within seven days.

Test Engineering

1. Now that the formal testing program is completed, the test files will be removed from site. The goal is to aid in the effective disposition of all test data files.

   Status: This item is approximately 10% complete. Several discussions regarding the remaining records have been held with the Station Office Manager. The Test Engineering department has been dissolved. Disposition of Testing records is now being handled by the Surveillance Coordinator.

2. With the reduced amount of instrumentation now required, some systems (example - DAS) will no longer be needed. Goal is to aid in the retirement of no longer used plant instrumentation.

   Status: The Data Acquisition System (DAS), Inverse Kinetics Simulator (IKS), and other testing related equipment have been removed. This item is 100% complete.

Safety and Budget Administrator

1. Perform daily plant tours.

   Status: 50% complete.

2. Establish plant conformance to hearing protection requirements.

   Status: 50% complete.

3. Maintain overtime expenditures within the overtime budget.

   Status: 50% complete.

4. Achieve zero lost time accidents.

   Status: 50% complete.

5. Reduce the number of OSHA reportable accidents.

   Status: 40% complete.

6. Perform monthly trend analysis of accident reports.

   Status: 50% complete.

Surveillance Coordinator

1. Expedite action on surveillance trend report open items, so that these items are not carried on this report more than three months.

   Status: No item was carried more than three months.

2. Perform a monthly review of the Emergency Control Center for its readiness in the event of an emergency.

   Status: Review has been performed each month.

3. Job Order Goals:

   a. Maintain or better a two-week turnaround for review and return of comments, or sign-off of Job Order drafts.

      Status: All Job Order drafts were commented on or signed off within two weeks or less.

   b. Maintain or better a one-week turnaround for review and sign-off or submittal of comments on Job Orders submitted for initial issue.

      Status: Three of five Job Orders submitted for initial issue were commented on or signed off within one week.

   c. Maintain or better a four-day turnaround for review and sign-off, or submittal of comments on Job Orders submitted for addendum changes.

      Status: All Job Orders submitted for addendum changes were commented on or signed off within four days.

Shippingport Atomic Power Station
Staff Inspection Program

Areas Inspected

1. Area A: Control Room, Turbine Building, ACR, Flywheel Generator Building, Diesel Enclosures

   Status: 69 deficiencies reported. 87% complete.

2. Area B: Chambers and Enclosures, 1B Aux. Equipment Room, Sample Prep Room, Chemistry Labs, Gas Bottle Storage Area

   Status: 106 deficiencies reported. 84% complete.

3. Area C: Fuel Handling Building including Canal Area, Core Vault, Cont. Instrument Shop, Clean Room, Decon Room, Machine Shop, 1A Aux. Equipment Room, 1A & 1B Mechanical Equipment Room

   Status: 104 deficiencies reported. 76% complete.

4. Area D: RWP and Yard areas including Screenhouse, Gas Bottle Storage Building, Turbine Deck, Heat Dissipation Area SIS Tank & Pumphouse, Outfall, Laydown Building, Dimineralizer Building

   Status: 96 deficiencies reported. 82% complete.

LWBR SIMPLIFIED DEFUELING SEQUENCE

-------- In Progress ________ Complete

SHUTDOWN, COOLDOWN, BORATE THE REACTOR COOLANT SYSTEM

REMOVE THE REACTOR CHAMBER DOME

HEAD AREA DISASSEMBLY
remove cables and LPI coils
remove Service Lead Support Structure

VENT REACTOR VESSEL AND LOWER WATER LEVEL

REMOVE CDM MOTOR TUBES AND D/P CELL PIPING. REMOVE LWBR CLOSURE HEAD

MOVE M-130 INTO FUEL STORAGE PIT

INSTALL REFUELING SEAL AND FLOOD REACTOR PIT

PLACE M-130 IN DEEP PIT

REMOVE HOLDDOWN BARREL AND DISASSEMBLE LWBR
Remove, disassemble, and load core components for ten M-130 shipments:
1. Reflectors V-3, V-4, IV-3, IV-4
2. Blankets II-1, III-6, I-3
3. Reflectors V-2, V-5, IV-1, IV-7
4. Blankets III-4, II-3, I-1
5. Reflectors IV-6, V-6, V-1
6. Blankets I-2, IV-1, III-2
7. Seeds III-2, I-2, II-2, III-1, II-1, III-6
8. Reflectors IV-2, IV-5, IV-8, IV-9
9. Blankets III-2, III-5, III-3
10. Seeds III-3, I-3, I-1, III-5, II-3, III-4

LOAD SHIPPING CONTAINER FOR COMPONENTS TO BE EXAMINED AT THE EXPENDED CORE FACILITY

LOAD ADDITIONAL MATERIAL IN PWR 2 LOWER CORE BARREL

TURNOVER TO DECOMMISSIONING CONTRACTOR

_____ Maint. Work Initiated

TURBINE PLANT SALVAGE ITEMS

_____ Maint. Work Completed

AUXILIARY STEAM

Aux. Steam Traps ST-1 & 3 through 6	Phillips
Gland Steam Reg. Valve MI-HHI	Phillips

CIRC. WATER

Circ. Water Pumps	Phillips

CONDENSATE

Air Leakage Rec.	Elrama
Chem. Add. Pumps	Phillips
Phosphate Pump	Phillips
Hydrazine and Morpholine Pumps	Phillips
Hotwell Level Tran.	Phillips
Cond. Reg. Valves	Phillips
Heater Drain Pump Disch. Check Valve	Elrama

CONTROL AIR

Comp. Motor	Phillips
Comp. w/ Motor	Phillips
Air Dryer	Phillips

EXTR. STEAM

Moisture Sep. Level Control	Phillips

FEEDWATER

Drum Level Rec.	Phillips
BFP Press. Trans.	Elrama
Feed Reg. Valves	Phillips
BFP Recirc. Reg. Valves	Phillips
Feed Press. Trans.	Elrama

MAIN GEN.

Gen. Panel Hydrogen Inst.	Elrama
Thermal Conv.	Phillips
Freq. Recorder	Phillips
Recording VM	Phillips
Megastat V. Reg	Elrama
Main Gen. Field Breaker	Elrama
H_2 Seal Oil Sys. Pumps, Motors, etc.	Phillips
H_2 Gas Dryer	Phillips

MAIN TURBINE

H.P. Control Oil Dump Valves	Phillips
M.U. RPM Rec.	Phillips
Turning Gear Motor	Phillips
Oil Centrifuge Motor	Elrama
Turb. Supervisory	Inst. School

MAIN STEAM

Drip Pocket & Mixing Hdr. Traps	Phillips
Steam Press. Trans.	Elrama

SERVICE AIR

Comp. w/ Motor	Phillips

STATION SERVICE

2300V Breakers	Elrama
Emerg. Batt. MG Set	Elrama
Emerg. Ltg. Batt.	Elrama
Lighting & Power Panels	Phillips
440V MCC's	Ph. & El.

WATER TREATING

Softener & Demin. Flow Integ.	Elrama
pH Recorder	Phillips
pH Indicators	Phillips
Agitators (Mixers)	Phillips

	MARCH				APRIL				MAY				JUNE				JULY				AUGUST					SEPTEMBER		
	14/18	21/25	28/1	4/8	11/15	18/22	25/29	2/6	9/13	16/20	23/27	30/3	6/10	13/17	20/24	27/1	3/8	11/15	18/22	25/29	1/5	8/12	15/19	21/26	29/2	5/9	12/16	19/23	26/30
ARMS	P			I																									
DSLM (3)					P	I																							
DHR			P		I																								
MFC				PI																									
HI					PI																								
SEISMIC SCRAM						P					I																		
FLY-WHEEL GENERATOR								P	I																				
1550'*								P			I																		
HEAT DISSIPATION								P		I																			
CACS						P	I																						
CIS									P		I																		
PZR AND RELIEF					P			I																					
BLOWOFF TANK									P																I*				
CHARGE (1)										P															I*				
LAS										P		I																	
SAMPLING											P		I																
RFCCJ											P			I															
CRC												P			I														
CORE INST.												P		I															
VALVE OPERATING SYS																						P		I					
SIS (2)													P		I														
SIS CNTL (2)													P		I														
RWP & Decon.Sys. *														P		I													
FLASH TANK															P										I*				
CHEMICAL SHUTDOWN																P									I*				
DC CONTROL *															P		I												
CEEFS																P		I											
CHEMICAL ADDITION											P		I																
STATION SVC. ELEC. *																							P		I				
STANDBY PWR BUS *																								P		I			
CONDS *																									P		I		

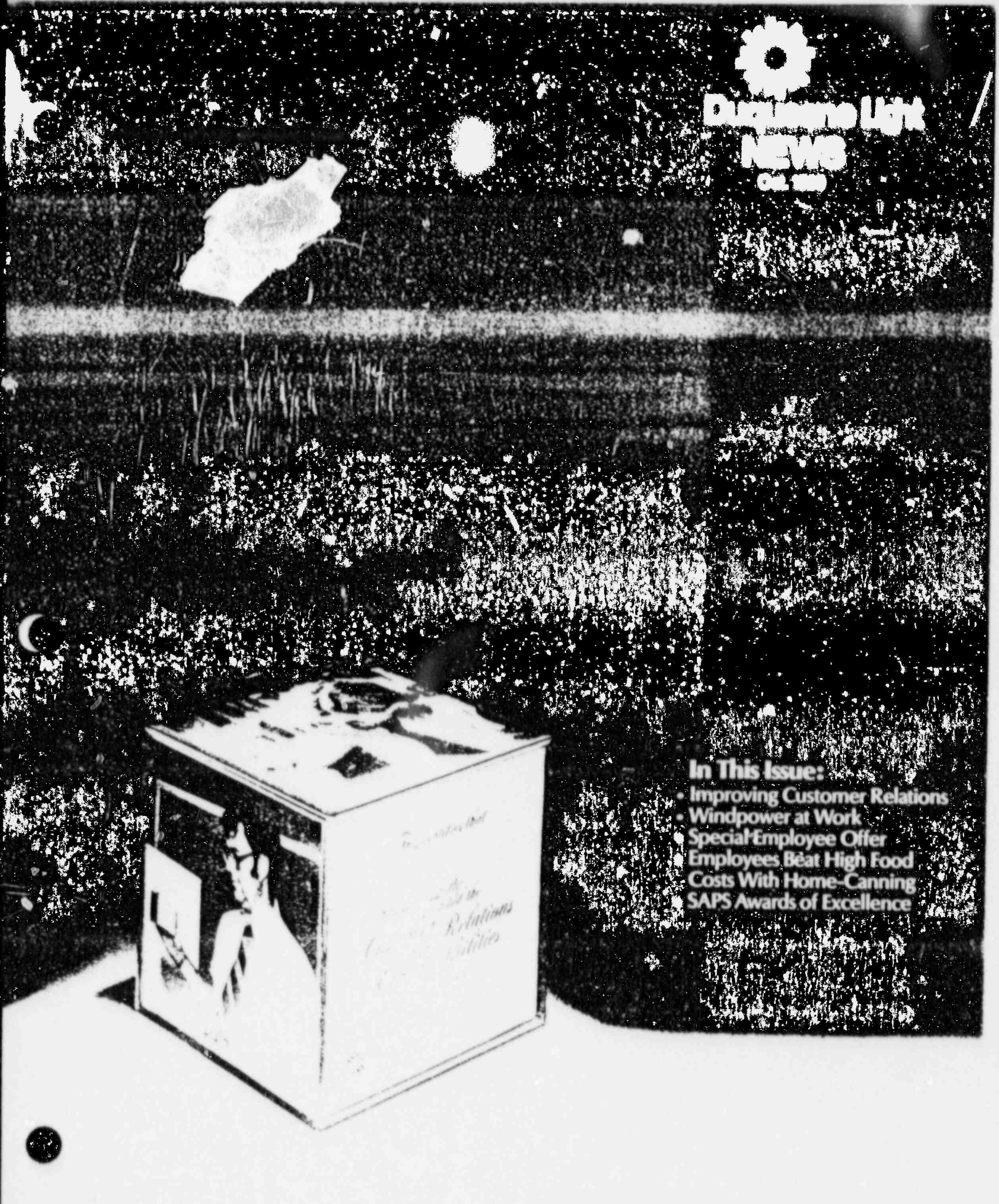
NOTES: (1) Fill pump after flushing last M-130 shipping container
(2) SIS Flow Path #1, flooding pump & fill add valve control after fuel removed
(3) Primary piping to be drained after final lowering of vessel level
* Partial inactivation
P Represents dates for completing paperwork to support inactivation
I Represents dates for the physical inactivation of the system.

* After S/G inspection

REV. 1
4/83

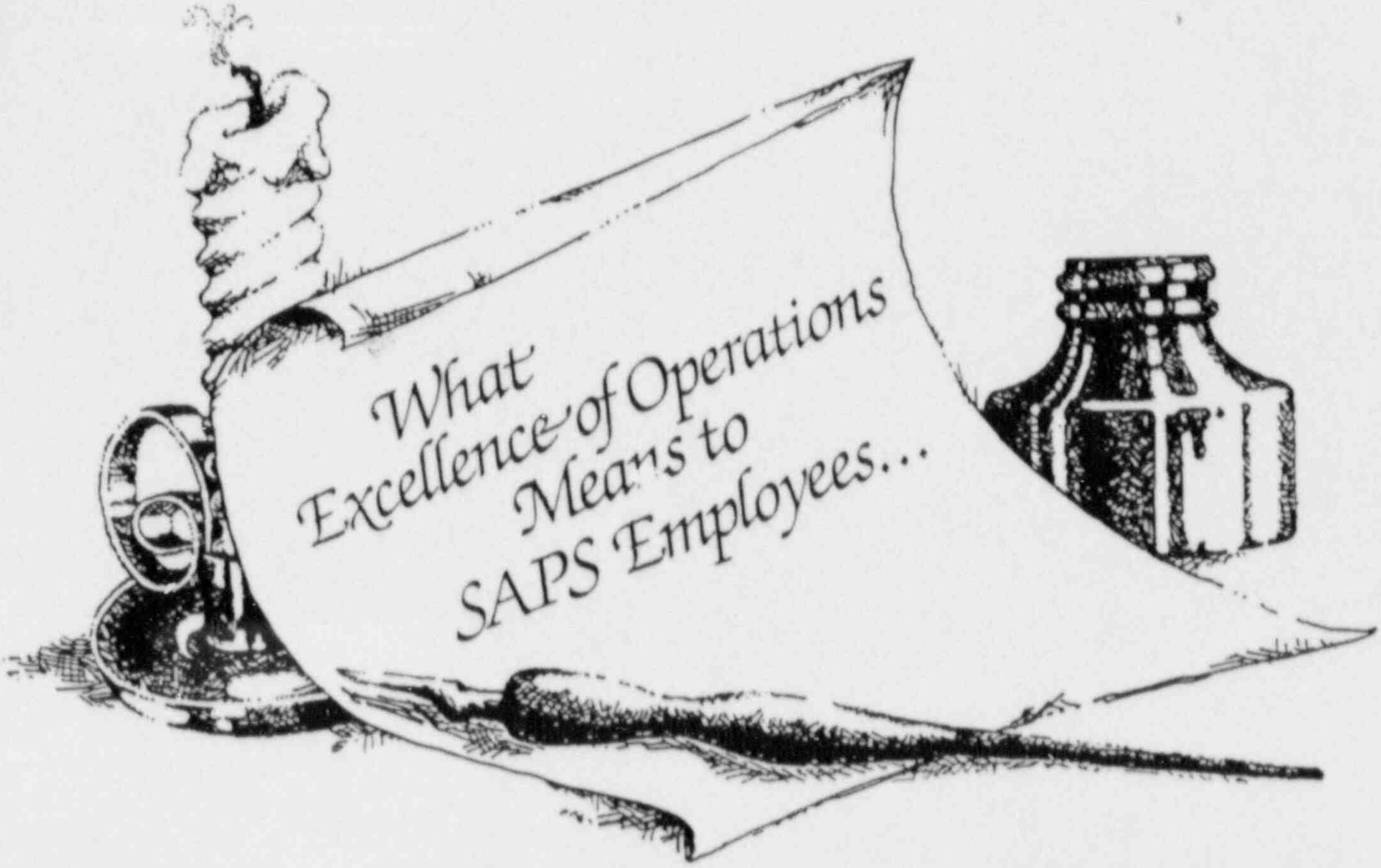
For details of system inactivation (boundaries, equipment status, etc.) see inactivation guidelines and the system status logs.

Duquesne Light NEWS



In This Issue:

- Improving Customer Relations
- Windpower at Work
- Special Employee Offer
- Employees Beat High Food Costs With Home-Canning
- SAPS Awards of Excellence



For the past 18 months the news media has carried numerous stories about the Three Mile Island accident and its overall effects on the future of nuclear power. At the same time, the Nuclear Regulatory Commission (NRC) has stressed to electric utility management that excellence of operations in nuclear power generation is a must.

At all Duquesne Light nuclear facilities, excellence of operations has always been a top priority. Recently, however, the staff at Shippingport Atomic Power Station (SAPS), the world's first full-scale nuclear power station operated by Duquesne Light, took the NRC's challenge a step further through a written essay contest.

The contest, formulated and coordinated by Al Morabito, former Training Supervisor and now Results Coordinator; former Station Superintendent T. D. Jones and Joe Zagorski, current Station Superintendent, encouraged SAPS personnel to describe how they could achieve excellence in their own personal work activities.

"By putting their thoughts into words," Zagorski explained, "it was hoped that each employee would be encouraged to put their words into actions which would result in the improved performance of duties necessary to achieve excellence of operations."

Although no guidelines were established for the contest, the 35 participants who entered had to complete the sentence: "To me, excellence of operations means...."

The grand prize winner, awarded $100, was Robert Gross, a First Class Mechanic.

Runnersup, who each received a $30 prize, included John Frye, Chemist, who has since left the Company; George Storolis, Nuclear Shift Supervisor; Wayne Graper, Radiation Technician; Joe Greco, Nuclear Operator; Gary Luff, Meter and Control Repairman; and Marge Hauschild, former Junior Steno-

DISCUSSING contest rules with entrant George Storolis (right) prior to the judging is Al Morabito, who formulated the contest.

ACCEPTING top prize for the essay contest from Station Superintendent Joe Zagorski (right) is Robert Gross.

grapher and now an Intermediate Clerk at Beaver Valley Power Station.

In describing what excellence of operations means to him, grand-prize winner Gross said it consists of pride, safety and good organization.

"This holds true in all industries," he pointed out. "You should plan your work to achieve the maximum quality, safety and efficiency possible, for without good organization and planning, excellence of operations will never be achieved."

For John Frye, excellence of operations means pride in performance.

"When someone is proud of their performance, they tend to expand their expertise beyond the confines of their own department," Frye said.

As one of the first students recruited for a nuclear operator class in 1974, contest runner-up George Storolis says excellence of operation is "constant and deliberate attention to the utmost detail in directing the operating force and monitoring the compliance of operating procedures."

To Wayne Graper, good job performance is the key to excellence of operations.

"Excellence is staying cognizant of all verbal and written changes in all aspects of your job," Graper said. "Another way of achieving excellence is in establishing good work practices and following them in order to set a good example for those less aware of the consequences."

For Joe Greco, excellence involves personal and public safety, safety of equipment, continuity of operations or services, and economy of operation.

He believes that these guidelines are best achieved when all plant personnel work together safely in continuing the most economical way of producing power at Shippingport.

Cooperation and a thorough review of work procedures are what excellence of operations means to Gary Luff.

"In my own responsibilities, if a job is to be carried over to another shift, time should be permitted to familiarize the oncoming shift with the status of the job, condition of equipment and the remaining work to be performed."

Performing assignments in an efficient and conscientious manner was Marge Hauschild's reason for writing her essay.

"Everyone should be excellent in their job because it provides us with our livelihood," she explained. "I also realize that I represent Duquesne Light to friends and acquaintances. The Company offers me an opportunity and way to support myself and I feel one way to repay the Company is to offer them my support by being excellent in my operations."

Although a number of entries were not accorded prize-winning status, the essays of all the writers were displayed on bulletin boards throughout the station.

Judging the essays were Gary Luff, Jim Friend, Gene Ramic and Joe Tyskewicz, all employees at the station.

EMPLOYEE PRIZE WINNERS in the competition included (front row, left to right) Robert Gross, Marge Hauschild and Joe Greco. Behind them are Wayne Graper, John Frye, who is no longer with the Company; Gary Luff, and George Storolis.

EXCELLENCE OF OPERATIONS

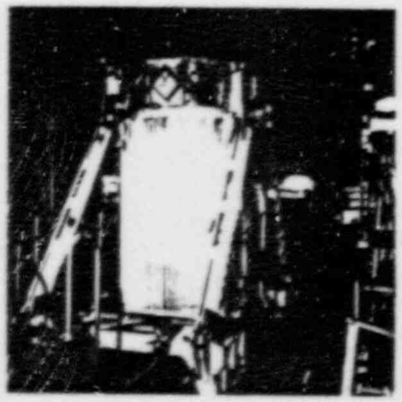
First Reflector M-130 Shipment to the Expended Core Facility on September 12, 1983.

Duquesne, Burns, Westinghouse, and Department of Energy personnel gave us a good example of Excellence of Operations. Our goal was to have the first M-130 Reflector Shipping Container ready to go out the gate by September 12, 1983. We met this important milestone on September 12, 1983, at 1240 hours.

On September 16, 1983, at 1125 hours, the Duquesne Light personnel pictured below moved the first blanket M-130 shipping container into the south end of the Fuel Handling Building. And so we continue to meet our challenges.

All of this contributes to the support of our M-130 shipment schedule and to the

Excellence of Operations



EXCELLENCE OF OPERATIONS

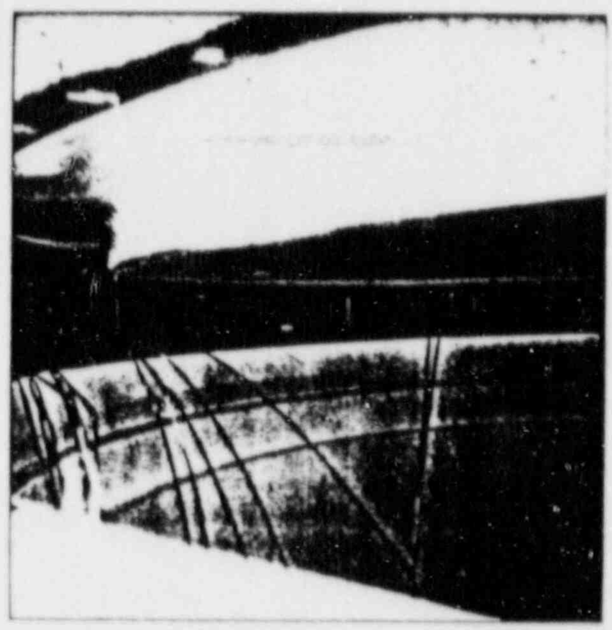
Wampum Mine Records Disposal

Clerical personnel recently began disposal of Shippingport station records at Wampum Mine. These records have accumulated over the Station's 25-year operating history. Phase I involved a total review of all documents containing some 1,300 transfer cases. Phase II involved the actual disposal of records per DLC recommendation and DOE/PNR approval. Clerical and Maintenance personnel scrapped approximately 900 transfer cases of miscellaneous documents, which had no history significance. Phase III involved recording the identity of each document and repackaging these records for retention.

Some 400 transfer cases of records were repackaged in 800 records storage boxes. Keep in mind, the dust and dirt did not help in making this project pleasant. However, Intermediate Clerks, Beth Walter and Becky Stern, as you can see, dug right in and dusted the job off. Station Office Manager, Bill Strayhorn directed the work activity to ensure DOE and DLC record-keeping requirements were maintained. Congratulations on a job well done in "record" time!!

A PROBLEM DISCOVERED AND CORRECTED
THROUGH
"EXCELLENCE OF OPERATIONS"

Though our defueling procedures did not require removal of the closure head from the PWR-2 core barrel inner shipping container until the container was in the water, our defueling personnel felt that it would be prudent to perform a trial removal with the container on dry land. This would facilitate resolution of problems. This action led to the discovery of the unspiraling of the closure gasket. We were able to expediently correct the problem. Note the tangled mess that could have ended up in the canal.



ES-301

NRC Form 157C, Page 1 (9-84)

U.S. NUCLEAR REGULATORY COMMISSION

SENIOR OPERATOR EXAMINATION REPORT

INSTRUCTIONS: Provide all the information requested for Power and Non-Power Reactors. Screened items do not apply to Non-Power Reactors.

DOCKET NUMBER	
55- 60755	
TYPE OF REACTOR	TYPE OF EXAM
X POWER	X INITIAL
NON-POWER	RETAKE

CANDIDATE'S NAME	REACTOR	LOCATION
Al Morabito	Beaver Valley I	Shippingport PA

WRITTEN EXAMINATION

SENIOR OPERATOR (Section 55.22)			
ADMINISTERED BY: David SILK	DATE 7/22/86	WAIVED	
GRADED BY: David SILK	GRADE 82.2	EVALUATION	
		PASSED	
CATEGORY GRADE: 5 87.6 % · 6 59.7 % · 7 85.3 % · 8 96.3 % · H % · I % · J % · K % · L %		X FAILED	

OPERATING TEST

(Section 55.23)	ADMINISTERED BY:	DATE	
	David SILK (observed by Barry Norris) BF 7 Aug 86	7/23/86	WAIVED
			X PASSED
			FAILED
			X HOT · COLD

SIMULATOR TEST (Not applicable to Non-Power Reactors)

(Section 55.23)	ADMINISTERED BY:	DATE	
	David SILK (Observed by Barry Norris) BF 7 Aug 86	7/23/86	WAIVED
			PASSED
			X FAILED

COMMENTS

RECOMMENDATION	SIGNATURE – EXAMINER	DATE
APPROVE FOR SENIOR LICENSE	David Silk 8/7/86	
X DO NOT APPROVE FOR SENIOR LICENSE	RM Keller	8/20/86
	BF	7 Aug 86

ISSUE LICENSE (Signature)	DATE	DENY APPLICATION (Signature)	DATE
		KB Koster	8/27/86

SENIOR OPERATOR
OPERATING AND ORAL EXAMINATION SUMMARY REPORT

*For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.	EVALUATION SRO	*PAGE NUMBER FOR "U"
1. OPERATING ☑ DISCUSSION ☐ DEMONSTRATION		
1.1 Pre-startup and Instrument Checks		
1.2 Console Operation		
a. Manipulations		
b. Understanding		
1.3 Plant Direction and Control		
2. FACILITY EQUIPMENT		
a. Major	S	
b. Auxiliary	S	
c. Engineered Safeguards Systems	S	
d. Electrical	S	
3. INSTRUMENTATION		
a. Nuclear	S	
b. Process	S	
4. PLANT PROTECTION	S	
5. PROCEDURES		
a. Normal	S	
b. Offnormal/Abnormal	S	
c. Emergency	S	
6. a. REACTIVITY EFFECTS (Except Console Operation)	S	
b. THERMODYNAMICS AND HYDRAULICS	S	
7. ADMINISTRATIVE REQUIREMENTS	S	
8. RESPONSIBILITIES AND AUTHORITIES		
a. Radiation Protection and Control	S	
b. Emergency Plan	M	pg 10
c. Other Duties and Responsibilities	S	

COMMENTS:

OPERATING AND ORAL EXAMINATION NOTES

A. OPERATING DEMONSTRATION	EVALUATION
CHECK ONE: ☐ REACTOR STARTUP ☐ STARTUP CERTIFICATION ☒ SIMULATOR DEMONSTRATION	
1.1 Pre-Startup or Instrument Checks, Type of Checkout *(specify)* Refer to Simulator	N/A
1.1.1 Familiarity with checksheet Forms attached	
1.1.2 Accuracy when reading instruments Attachments 5 and 11 of ES-302	
1.1.3 Understanding of what is being checked	
1.1.4 Understanding of reasons for checkout	
1.2 Console Operation	
a. Initial conditions	
b. Program	
c. Understanding	
1.2.1 Ability to predict response for specific program	
1.2.2 Understanding of instrument response	
1.2.3 Knowledge of reactivity effects	
d. Manipulations	
1.2.4 Follows procedures	
1.2.5 Observes and checks instrumentation	
1.2.6 Ability to follow specified program accurately	
1.2.7 Dexterity and "feel" for console controls	
1.3 Plant/Facility Direction and Control	
1.3.1 Ability to direct plant operation	
1.3.2 Plant parameter verification *(ECP, heat balance, etc.)*	
1.3.3 Technical specification requirements	
1.3.4 Equipment OOS requirements	
1.3.5 Experimental Facilities *(Non Power Reactors only)*	

COMMENTS

B. CONTROL ROOM
(Major, Auxiliary and Engineered Safeguards Systems)

	SYSTEMS							
	Rod Control	Feedwater		Makeup	AFW		HHSI	Containment Depress.
	A	B	C	D	E	F	G	H
2.0 EQUIPMENT								
2.1 Purpose								S
2.2 Flow Path		S		S	S		S	S
2.3 Normal Parameters					S			
2.4 Components		S		S	S		S	S
2.5 System Behavior and Response	S	S		S	S		S	U
3.0 INSTRUMENTATION								
3.1 Detector								
3.2 Malfunction	S	S						
3.3 Control Room Indication	S	S		S	S		S	S
4.0 PLANT PROTECTION								
4.1 Alarms/Setpoints	S	S		S				
4.2 Safety System Input		S						
4.3 Interlocks	S	S			S		S	S
5.0 PROCEDURES								
5.1 Normal Procedures								
5.2 Offnormal/Abnormal Procedures	S							S
5.3 Emergency Procedures		S					S	
6.0 A. Reactivity Effects				S				
B. [redacted]/Thermal Effects		S						
7.0 ADMINISTRATIVE REQUIREMENTS								
7.1 Technical Specifications	S				S			S
7.2 Facility Requirements								

COMMENTS *(Required for "U")* 2.5.H on next page.

Columns A, D, and G covered during simulator examination

☐ CONTINUED ON REVERSE

COMMENTS (Continued)

2.5. H. Candidate said that MOV-RS-155 A&B (outside Recirculation pump Suction Isolation Valves) are normally open. OM 1.13.1 page 23 indicates that the valves are normally close and will open if control switch in "open" position or a Phase C signal is present.

B. CONTROL ROOM
(Nuclear and Radiation Instruments)

	SYSTEMS: PR (A)	B	SR (C)	D	Secondary Rad Monitors (E)
3.0 INSTRUMENTS					
3.1 Detectors			S		S
3.2 Malfunctions	S				S
3.3 Control Room Indications	S				S
3.4 Channel Components			S		
3.5 Compensation/Discriminator					
3.6 Input to Control System	S				S
4.0 PLANT PROTECTION					
4.1 Alarms/Setpoints			S		S
4.2 Safety System Input	S		S		
4.3 Interlocks	S		S		
5.0 PROCEDURES					
5.1 Normal Procedures			M		
5.2 Offnormal/Abnormal Procedures	U				
5.3 Emergency Procedure					S
7.0 ADMINISTRATIVE REQUIREMENTS					
7.1 Technical Specifications	S				
7.2 Facility Requirements					

COMMENTS: *(Required for "U")*

5.1.C Candidate said that SR fuses were removed at power in accordance with a standing night order. Fuses are removed by personal preference to prevent spurious trip - no longer proceduralized.

5.2.A Candidate did not consult any procedure when decreasing load to check Power Range indicator response for two PR indicators that were reading lower than the other two. If candidate considered two PR indicators inoperable then plant should be in Mode 3 with 1 hour as per AOP-10.

B. CONTROL ROOM
(Electrical)

	SYSTEMS			
	138 KV		4160 KV	
	A	B	C	D
2.0 EQUIPMENT				
2.1 Purpose				
2.2 Flow Path	S		S	
2.3 Normal Parameters				
2.4 Components	S		S	
2.5 System Behavior or Response	S		S	
3.0 INSTRUMENTS				
3.2 Interlocks	S		S	
3.4 Control Room Indication	S		S	
5.0 PROCEDURES				
5.1 Normal Procedures			S	
5.2 Offnormal/Abnormal				
5.3 Emergency Procedures				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications	S		S	
7.2 Facility Requirements				

COMMENTS: *(Required for "U")*

C. REACTOR AND AUXILIARY BUILDINGS *(Power Reactors)*
(Major, Auxiliary, Electrical Safeguards, Fuel Handling)
FACILITY WALK THROUGH *(Non-Power Reactors)*

	SYSTEMS: A Fuel Handling	B	C DG	D	E AFW	F
2.0 EQUIPMENT						
2.2 Flow Paths	S		S		S	
2.3 Normal Parameters						
2.4 Equipment Location	S		S		S	
2.5 System Behavior and Response	S		S		M	
3.0 INSTRUMENTS						
3.8 Local Instrumentation			S		S	
5.0 PROCEDURES						
5.1 Normal Procedures (Local)	S					
5.2 Offnormal/Abnormal Procedures (Local)			S		S	
5.3 Emergency Procedures (Local)						
6.0A. REACTIVITY EFFECTS	S					
B. THERMODYNAMICS ANALYSIS/Thermal Effects						
7.0 ADMINISTRATIVE REQUIREMENTS						
7.1 Technical Specifications	S		M		S	
7.2 Facility Requirements						

COMMENTS *(Required for "U")*

2.5.E Candidate said he did not know if steam would be flowing to TDAFWP while the trip valve was being reset following an overspeed trip

7.1 Candidate said that diesel powered air compressor was addressed by T.S.. It is not. It is a backup to motor powered compressors which ensure starting air tanks are at sufficient pressure to start DG.

D. DISCUSSIONS *(Integrated Plant Response)*	SYSTEMS A	B	C	D
2.0 EQUIPMENT				
2.6 Components Response				
3.0 INSTRUMENTS				
3.4 Control Room Indications				
3.8 Automatic Control				
3.9 Ability to Manipulate Manual Control				
4.0 PLANT PROTECTION				
4.1 Automatic Actions				
4.2 Alarm/Setpoints				
5.0 PROCEDURES				
5.1 Normal Procedures				
5.2 Offnormal/Abnormal Procedures				
5.3 Emergency Procedures				
6.0 REACTIVITY EFFECTS AND THERMODYNAMIC ANALYSIS				
6.3 Coefficient Effects/Reactivity Effects				
6.6 Transient Analysis/Thermal Analysis				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS *(Required for "U")*

See Attachments 5 and 11 of ES-302

D. DISCUSSION *(Power Reactors)*	EVALUATION
6.0 THEORY OF NUCLEAR POWER PLANT OPERATION A. REACTIVITY EFFECT *(Nuclear Theory)*	
6.A.1 Subcritical Multiplication	S
6.A.2 Delayed Neutrons Effect	S
6.A.3 Coefficients	S
6.A.4 Poison Effects	S
6.A.5 Long Term Exposure Effects	S
6.A.6 Axial and Radial Limits	S
6.A.7 Shutdown Margin Could not convert from ppm boron to $\Delta k/k$	U
6.A.7 Safety Limits	S
B. THERMODYNAMICS AND HYDRAULICS 6.B.1 Steam Tables	S
6.B.2 Instrumentation	S
6.B.3 Pump Characteristics	S
6.B.4 Inadequate Core Cooling	S
6.B.5 DNBR, MCPR, etc.	S
6.B.6 Operational Analysis	S
8.0 RESPONSIBILITY AND AUTHORITY A. RADIATION PROTECTION CONTROL	
8.A.1 Source and Hazards of Radiation	S
8.A.2 Exposure Limits (10 CFR 20, Facility)	S
8.A.3 Portable Instrumentation (Knowledge and Use)	S
8.A.4 Procedures (RWP, containment entry, etc.)	S
8.A.5 Release Permits (gaseous, liquid, purge)	S
B. EMERGENCY PLAN IMPLEMENTING PROCEDURES 8.B.1 Duties	S
8.B.2 Classification	S
8.B.3 Evaluation Criteria	S
8.B.4 Personnel Assignments Took candidate excessive amount of time (~10 min) to find Evac Plan	U
C. ADDITIONAL DUTIES AND RESPONSIBILITIES 8.C.1 Surveillance Testing	
a. Instrumentation and Control	S
b. Other *(Specify)*	—
8.C.2 Security Did not know combination to NSS Key Cabinet	M
8.C.3 Shift Turnover	S

COMMENTS *(Required for "U", use reverse)*

EXAMINATION SUMMARY SHEET

Date: 7/23/86 Facility/Simulator Beaver Valley 1

Candidate A Morabito Examiner SILK / Norris

Type of Exam: SRO Instant ✓ SRO Upgrade ____ RO ____

RATING (Circle one for each competency)

S M U Understanding/Interpretation of Annunciators/Alarm Signals

S M U Diagnosis of Events/Conditions

S M U Understanding of Instrument/System Response

S M U Compliance/Use of Technical Specifications

S M U Compliance/Use of Procedures

See following pages for comments

S M U Control Board Operations (RO & SRO Instant)

See following pages for comments

S M U Supervisory Ability (SRO)

See following pages for comments

S M U Communications/Crew Interactions

See following pages for comments

Recommendations: Pass ____ Fail ✓ Signature David Silk

7 Aug 86

Examiner Standards

Compliance/Use of Procedures

During first scenario candidate did not consult any procedure when decreasing load to check power range indicator response for two power range indicators that were lower than the other two. AOP-10 calls for the plant to be in mode 3 if two power range channels are malfunctioning.

During second scenario while in ES-1.2 step 27 candidate asked "Are RCS hot leg temperatures greater than 395 F?". Candidate did not wait for an operator response and assumed the answer to the question was 'yes' by answering 'yes' aloud to himself. The operator then indicated the answer to the question was "No".

During third scenario candidate did not consult any ARP's in regards to decreasing pressurizer pressure and level caused by high failure of pressurizer level control channel.

During third scenario, after the reactor tripped and SI actuated, candidate did not check if LHSI pumps were running as required by immediate action step 11 b. of E-0. SRO had to remind candidate to check if LHSI pumps were running.

Control Board Operations (Third Scenario)

Following SI actuation as the RCS pressure was decreasing the candidate misread RCS wide range pressure indication. Candidate misread 1600 psig as 1040 psig and then checked with other operator to confirm RCP trip criteria.

In step 4 c of E-3, the Residual Heat Release valve was to be checked to ensure it was closed. Candidate was looking at the demand indicator for the manual control of Residual Heat Release Valve and not at the indication lights for the valve. Candidate was hesitant to respond to the check verification and appeared confused until other operator came over and explained the controls and indications to the candidate.

In step 9 a of E-3, the containment sump pumps were to be stopped. Candidate stopped one Containment sump pump and the Incore Instrument sump pump. The other operator came over to show the candidate where other containment sump pump switch was located.

In step 11 of E-3, CIA was to be reset. Candidate depressed the CIA Train B button and the CIB Train A button. CIA did not reset. Candidate did not verify CIA was reset following his attempt to reset C.A.

Supervisory Ability

In the second scenario, the candidate did not notice that the feed reg bypass valve indicator was indicating that the valve was open during diagnosis of unusual feed reg valve movement.

Unsatisfactory use of procedures and unsatisfactory crew interaction supports an unsatisfactory performance in supervisory ability.

Communications / Crew Interactions

During the first scenario, following the loss of offsite power, the candidate went to ECA-0.0 when he mistakenly observed that he had no emergency busses energized. Candidate should have relied upon verification of emergency buss s from his operator who did properly verify that one emergency bus was energized and informed the candidate as such.

During the second scenario, step 6 of E-1 calls for checking secondary radiation levels. An operator checked the monitors and said "One indicat is about this much (holding fingers about ½ to 1 inch apart) higher than norma Candidate proceeded in E-1. During followup questioning after the scenari the candidate admitted misunderstanding the operator's report of secondary radiation levels and assumed there was no reason to go to E-3.

During second scenario, while in ES-1.2 step 27, the candidate asked, "Are RCS hot leg temperatures greater than 395 F?" Candidate did not wait for an operator response and assumed the answer to the question was 'yes' and answered "yes" alouc to himself. The operator then indicated the answer to the question was "No".

ATTACHMENT 3

SIMULATOR SCENARIO FORM

Wed. 7/23/86

Facility/Simulator: BV 1 Scenario No. SB-I

Examiners: J. Barber
B. Norris (D Silk)
B. Gruel

Candidates: PO Ray Scneio (RO)
SRO Al Morabito (SRO)
RO Sue Neuder (RO)

TIME	MALF. NO.	DESCRIPTION
Pre-insert	CHS-7	Boron Dilution Accident
+5	CND-9	Vacuum Breaker Leak
+15	PRS-7	Pressurizer Reference Signal Failure
+25	TUR-7	Erratic Governor Valve Control
+30	EPS-1	Station Blackout

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SE-[illegible] Event No. ______ Page 1 of 6

Brief Description: Turnover [illegible] Conditions [illegible] Special Instructions

Plant Conditions	Position	Candidate Actions/Behavior
Turnover	All	A Diesel Generator is inoperable due to turbocharger replacement. OOS yesterday next surv due in 6 hrs.
		B Charging Pump is inoperable and torn down for impeller inspection - Should be returned to service in 16 hrs
		A spray valve closed and cleared for maintenance. B Spray Valve closed due to excessive leakage and tagged for emergency use only. O/C has to carry out 1# control
Spec Inst	Sim Inst	Rods in auto
		C Charging Pump lined up to DF bus and controlling Pzr lvl
IC-18	Sim Inst	Snap 18 100%, MOL, 480ppm B, 576°F 2235psig, equil Xenon

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-1 Event No. 1 Page 2 of 6

Brief Description: Boron Dilution

Plant Conditions	Position	Candidate Actions/Behavior
Boron Dilution	RO	Identify cause of alarms and ~~excessive~~ inward rod motion
		as leakage past CH-138
0-50 gpm		becomes more severe.
over 600 secs		
	RO/SRO	Monitor for excessive AFD
	SRO	Review CVCS valve alignment
		per the P&IDs. Coordinate recovery.
	RO/BOP	Have SO's attempt to determine
		improper valve alignment.
	RO	Order CH-138 closed

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-1 Event No. 2 Page 3 of 6

Brief Description: Vacuum Breaker Leak

Plant Conditions	Position	Candidate Actions/Behavior
Loss of Cond Vacuum	BOP	Identify lowering condenser vacuum.
Sized to not reach manual turbine trip	BOP	Monitor Vacuum and MWe
than 25"Hg Auto 20"Hg	SRO/RO/BOP	Coordinate to reduce load (Vacuum should stay above turbine trip point ≈25 in Hg)
	RO/BOP	Follow Low Vacuum ARP Sect 1.26.4 pg 74

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-1 Event No. 3 Page 4 of 6

Brief Description: Pressurizer Reference Signal Failure

Plant Conditions	Position	Candidate Actions/Behavior
Auctioneered Tave to pzr level fails to 530°F (459)	RO	Notice decrease in charging flow and pressurizer level. BU htrs on.
	RO	Respond to pzr low level and level deviation alarms
	RO	Control pzr level in manual
	SRO/RO	Contact I&C to troubleshoot the problem

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-1 Event No. 4 Page 5 of 6

Brief Description: Erratic Govenor Valve Control

Plant Conditions	Position	Candidate Actions/Behavior
Throttle valve oscillates 10% over 60 seconds	BOP	Notice S/G level swings and attendant level alarms
	RO	Notice Tave/Tref variations and alarms
Loss of Vacuum Should have them at ~10%	SRO/BOP	Identify cause of plant swings as being erratic throttle valve control. Contact I&C as necessary

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-1 Event No. 5 Page 6 of 6

Brief Description: Station Blackout

Plant Conditions	Position	Candidate Actions/Behavior
Station Blackout	BOP/RO	Take actions for turbine trip and reactor trip
abrupt	RO/BOP/SRO	E-0 Steps 1-7 Perform immediate action steps Kicked out to ES0.1
		ES0.1 Step 7 Verify DG have picked up emerg loads (Attach 1)
		Step 9 Verify NC (Attach 4)
		Step 14 Determine method of C/D or S/U. Cause will not allow startup.
		ES-0.2 NC Cooldown Establish requisite conditions to cooldown Initiate C/D

ATTACHMENT 3

SIMULATOR SCENARIO FORM

Facility/Simulator: BV 1   Scenario No. SB-2

Examiners: S. Barber   Candidates: Ray Schenk (RO)
K. Norris (D. Silk)   Al Moralito (SRO)
B. Gruel   Sue Mender (RO)

TIME	MALF. NO.	DESCRIPTION
Pre-insert	PRS-9B	B Spray valve fails closed
"	MSS-7	Tave input to steam dumps failed low
+5	FWM-8B	B Loop FRV bypass fails open
+15	RCS-14E	Loop 3 Th fails high
+28	TUR-4A,B,C,D	Turbine Govenor Valves Fail Closed
		Fail PORV block valve open for 455D (537) due to a breaker problem.
+29	PRS-4A	Fail Pzr PORV 455D open

Examiner Standards

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-2 Event No. 1 Page 1 of 4

Brief Description: B loop FRV bypass fails open

Plant Conditions	Position	Candidate Actions/Behavior
B Loop FRV bypass fails open	BOP	Notices level alarms on 'B' S/G
100 percent over 5 secs		Identifies high level on B and attempts to determine the cause.
		Notice valve position indication OPEN.

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-2 Event No. 2 Page 2 of 4

Brief Description: Loop 3 T_h fails high

Plant Conditions	Position	Candidate Actions/Behavior
Loop 3 T_h fails high	RO	Notices alarms from Tave/Tref deviation
		Control rods insert due to large Tave/Tref mismatch
		Large ΔT should cause C-3 stop
	RO	Block failed instrument input

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-2 Event No. 3 Page 3 of 4

Brief Description: Turbine Govenor Valves Fail Closed

Plant Conditions	Position	Candidate Actions/Behavior
Load rejection	BOP	Turbine load drops rapidly as the govenor valves close. Operator Identifies MWe ↓ S/G lvls ↓
	RO	Identifies rapid increase in pressurizer level, pressure and Tave
PRS-9B		'B' spray valve fails to open
MSS-7		Steam Dumps fail to open
		RCS pressure continues to increase until turbine and reactor trip on high pressure. E-0 actions
		PORV's open to reduce pressure

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-2 Event No. 4 Page 4 of 4

Brief Description: Fail Przr PORV 455D Open

Plant Conditions	Position	Candidate Actions/Behavior
SBLOCA	RO/SRO	Notice PORV (455D) open and attempt to close it. It fails to close. Attempt to close block valve
		Auto Low RCS pressure SI.
	SRO	E-0 Step 22 Requires checking PORVs closed. Go to E-1 Loss of Pri/Sec Cool. Step 1
	RO	E-1 Step 7 Restore power to PORV block valve
	SRO	E-1 Step 18 Go to Post-LOCA C/D and Depress. ES-1.2 Step 1
		ES-1.2 Step 24 Depressurize RCS to minimize break flow

ATTACHMENT 3

SIMULATOR SCENARIO FORM

Facility/Simulator: BV 1 Scenario No. SB-3

Examiners: S. Barker, B. Norris (D Silk), B. Gruel

Candidates: Roy Scheib (RO), Al Morahutd (SRO), Sue Dexter (RO)

TIME	MALF. NO.	DESCRIPTION
+5	NIS-3D	PRNI (44) Fails high
+10	RCS-3	S/G Tube Leak → Rupture
+15	PRS-6A	Fail pzr. level high
+20	FWP 1A	FWP-1A trip from 75% power Power reduction
(Pre-insert) +25	SIS-5A	"A" HHSI pump fails to auto start on low pressure SI

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-3 Event No. 1 Page 1 of 4

Brief Description: Power Range NI Fails High

Plant Conditions	Position	Candidate Actions/Behavior
PRNI fails high	RO	Respond to N-44 failing high
		Overpower rod stop will be in effect
		Respond to inward rod motion
	SRO	Order tripping of affected bistables per AOP #60

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-3 Event No. 2 Page 2 of 4

Brief Description: S/G Tube Leak → Rupture

Plant Conditions	Position	Candidate Actions/Behavior
S/G Tube Leak → Rupture	RO	Respond to Rad Monitor Alarm
0gpm to 400gpm over 20 minutes (1200 secs)	RO	Monitor charging flow & LD to determine leak rate
	SRO	Follow AOP-11. Take SG samples. Determine the affected SG
		Plant will trip on low RCS pressure as leakage increases
		Follow E-0 through Step 7 -auto SI-
'A' HHSI fails to start	RO	Identify cause of reduced SI flow
		Abnormal sec radiation requires E-3 Step 1
		Follow through E-3 to Step 43 then choose Post-SGTR C/D method

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-3 Event No. 3 Page 3 of 4

Brief Description: Fail pressurizer level high

Plant Conditions	Position	Candidate Actions/Behavior
Controlling Pzr lvl fails high	RO	Respond to level decrease and level alarms initially
		Letdown isolation will cause level increase from seal flow
	RO	Select operable level channel

ATTACHMENT 5

SIMULATOR ADMINISTRATION FORM

Scenario No. SB-3 Event No. 4 Page 4 of 4

Brief Description: FWP - 1A trip

Plant Conditions	Position	Candidate Actions/behavior
FWP-1A trip from 75%	BOP	Reduce load rapidly to restore S/G levels

Stop 1409

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: BEAVER VALLEY 1&2

REACTOR TYPE: PWR-WEC3

DATE ADMINISTERED: 86/07/22

EXAMINER: SILK, D.

APPLICANT: A. J. Moralito

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE		CATEGORY
25.00	25.00	21.9	87.6	5.	THEORY OF NUCLEAR POWER PLANT OPERATION, FLUIDS, AND THERMODYNAMICS
22.6 ~~25.00~~	25.00	13.5	59.7	6.	PLANT SYSTEMS DESIGN, CONTROL, AND INSTRUMENTATION
24.5 ~~25.00~~	25.00	20.90	85.3	7.	PROCEDURES - NORMAL, ABNORMAL, EMERGENCY AND RADIOLOGICAL CONTROL
25.00	25.00	24.07	96.3	8.	ADMINISTRATIVE PROCEDURES, CONDITIONS, AND LIMITATIONS
100.00	100.00				TOTALS

FINAL GRADE 82.2 %

All work done on this examination is my own. I have neither given nor received aid.

A. J. Moralito
APPLICANT'S SIGNATURE

QUESTION 5.01 (3.00)

a. Would the positioning of a neutron source TOO CLOSE to the neutron detector, being used for constructing a 1/M plot, result in OVERPREDICTING (not conservative) or UNDERPREDICTING (conservative) the reactivity addition needed to reach criticality? EXPLAIN. (1.0)

b. How does the initial source range level (cps) affect critical rod position? EXPLAIN. (1.0)

c. How does the positive reactivity insertion rate affect the source range count level at which criticality is achieved? EXPLAIN. (1.0)

QUESTION 5.02 (3.50)

a. Explain both HOW AND WHY the following factors affect differential boron worth (more negative, less negative or no change).

1. Boron concentration increase (0.75)
2. Moderator temperature decrease (0.75)
3. Fission product buildup (0.75)
4. Core burnup from MOL to EOL with constant rod position (0.75)

b. Why does the critical boron concentration drop rapidly from 0 to 150 MWD/MTU of burnup as seen in Figure 1? (0.5)

QUESTION 5.03 (3.00)

a. How does DNBR change (increase, decrease, no change) as the following are increased? (Consider each separately). (1.0)

1. Tavg
2. RCS pressure
3. RCS flow
4. Reactor power (Constant Tavg)

b. What adverse fuel assembly condition could result if actual heat flux exceeds the critical heat flux in a PWR core? Explain. (1.0)

c. From Figure 2, what parameter is being limited on Section A of the figure and what is the significance of it? (1.0)

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.04 (2.00)

After operation at 100% power for several weeks near the end of cycle, power is reduced to 75% using rods only.

a. Explain HOW and WHY Xenon concentration will change over the next 40 hours. (1.5)

b. What rod motion would be required to maintain the plant at 75% power over the same 40 hours assuming no change in boron concentration? Include applicable time frames. (0.5)

QUESTION 5.05 (1.50)

a. Does Beta bar effective increase, decrease, or remain the same from BOL to EOL? Explain your answer. (1.0)

b. For two equivalent positive reactivity additions to a critical reactor, will the SUR be the same, larger, or smaller at EOL as compared to BOL? No explanation is necessary. (0.5)

QUESTION 5.06 (2.50)

a. To increase the discharge head of a variable speed centrifugal hydro-pump from 1200 to 1800 psia by what factors should the speed and power inputs be increased? Show your calculations. (1.5)

b. What pressure is needed at the suction of a feed pump to provide 215 feet of NPSH if the water is at 384 F? (1.0)

QUESTION 5.07 (2.00)

When would a rod be worth more - if it were dropped while at power or if it were stuck out while all other rods were inserted? EXPLAIN.

(***** CATEGORY 05 CONTINUED ON NEXT PAGE *****)

QUESTION 5.08 (2.00)

With all systems in manual and no operator action, what effect (increase, decrease, no change) will decreasing the circulating water temperature have on the following?

a. Condenser vacuum
b. Condensate temperature
c. Steam generator pressure
d. Electrical output
e. Reactor power

QUESTION 5.09 (2.50)

a. What effect does increasing moderator temperature have on control rod worth? Explain. (1.0)

b. What is the effect of a dropped rod on long term reactor power and Tave? Explain. Assume all systems in manual and no reactor trip occurs. (1.5)

QUESTION 5.10 (3.00)

The reactor is at 70% power and Tave is 568 F. A governor valve failure raises load 15%.

a. From a reactivity standpoint, explain how and why reactor power responds. Assume rods in manual. (1.5)

b. Using Figure 3, calculate the new Tave if rods are in automatic. Show all work and state all assumptions. (1.5)

(***** END OF CATEGORY 05 *****)

QUESTION 6.01 (2.00)

a. Why is the loss of compensating voltage more noticeable during a startup than at 100% power? (0.5)

b. At 100% power the N44-B Power Range Detector fails high. With rods in manual, give five annunciators associated with the NIS that alarm. (1.5)

QUESTION 6.02 (2.50)

a. What is the major advantage of drawing an RCS activity sample from the letdown line instead of directly from the RCS? (0.4)

b. Besides the Condenser Air Ejector Radiation Monitor, list four radiation monitor alarms that may be indicative of a primary to secondary leak? (1.6)

c. What automatically happens when a high-high alarm from the Condenser Air Ejector Radiation Monitor occurs? (0.5)

QUESTION 6.03 (2.20)

a. How would an operator determine the location of a 10 GPM leak from the component cooling water system by using the indications available to him in the main control room? (0.7)

b. What three design features of the component cooling water system minimize the effects of a rupture of the RCP thermal barrier? (1.5)

QUESTION 6.04 (2.00)

a. What is the reason for maintaining a minimum pressure of 15 psig in the volume control tank? (0.5)

b. Normal operations has the '1C' charging pump breakers 1E15 and 1F15 disconnected from the bus. What prevents tying both emergency busses together? (0.5)

c. When is the Alternate Dilute mode used and what disadvantage accompanies its use? (1.0)

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.05 (2.00)

The plant is operating at 30% power when the controlling first stage impulse pressure transmitter PT 446 fails HIGH. Explain the effects of this failure and the sequence of events (control and protection) that lead to a reactor trip. Assume BOL, no operator action and initial plant conditions are in a normal system line-up for 30% power. (Setpoints are not required.) (2.0)

QUESTION 6.06 (2.70) Challenge this one

a. What is used to control RCS pressure during cold solid plant operations? (0.4)

b. What three plant conditions provide inputs to the interlocks associated with RHR suction valve MOV-RH-701? Setpoints are required. (1.5)

c. Prior to entering a water solid operating mode, describe how overpressure protection is enabled? (0.4)

d. If the air supply system for PORV's PCV-RC-455C & D fails, describe how the overpressure protection system functions? (0.4)

QUESTION 6.07 (2.50) Maybe challenge this one

a. Why is the operability of the steam generator code safety valves important during power operation? (0.5)

b. Give two reasons (NOT CONDITIONS) why the MSIV's are required to close during a steam line rupture. (1.0)

c. Which mode (HSB, HZP, HFP) and time in cycle (BOL, MOL, EOL) will have the most severe effect on a main steam line break accident. Explain each separately. (1.0)

Maybe [illegible] break to [illegible] steam generator [illegible] could [illegible] other side [illegible] [illegible] answers [illegible] [illegible]

(***** CATEGORY 06 CONTINUED ON NEXT PAGE *****)

QUESTION 6.08 (2.30)

a. Originating at the 4160V-1AE Bus, provide a sketch which shows how power is supplied to one 120 VAC Vital Bus and one 125 VDC Bus. The sketch should include all normal, alternate and emergency supplies. Label all electrical components, busses and transformers. (Breakers are not required.) (1.5)

b. Why do the two 480 V Emergency Motor Control Centers MCC1-E13 and E14 feeder breakers remain closed during a loss of offsite power? (0.4)

c. What signal is needed to allow sequencial loading following a loss of offsite power? (0.4)

QUESTION 6.09 (3.20)

a. What two simultaneous conditions will cause the quench spray flow cut-back valves (MOV-1QS-103A,B) to close? (0.8)

b. What is the purpose of the orifice that is parallel to quench spray flow cut-back valves? (0.8)

c. In the recirculation spray coolers, what is the reason for the recirculation water pressure being greater than the river water pressure? (0.8)

d. CIB has been reset and the spray pumps have been secured. If a CIB signal recurs, will the quench spray pumps restart automatically? EXPLAIN. (0.8)

QUESTION 6.10 (3.60)

a. The safety injection accumulators are required to be maintained within certain pressure limits. What problems exist if the pressure is significantly above and below its limits? (0.8)

b. What conditions are required before automatically transferring from the injection phase to the recirculation phase? Include logic and coincidences. (0.4)

c. What is the sequence of the automatic valve realignments that occur to transfer from the injection phase to the recirculation phase? (2.0)

d. If the low head safety injection pumps fail during recirculation, what can be done to provide suction to the high head safety injection pumps? (0.4)

(***** END OF CATEGORY 06 *****)

QUESTION 7.01 (1.70)

a. A Decay Tank Discharge is in progress when Gaseous Waste Gas Monitor, RM-1GW-108B, becomes inoperable. Briefly explain what has to be done to continue the release. (1.0)

b. During a liquid release, a liquid waste effluent high-high activity alarm comes in. The problem is identified and corrected. When pumping to the cooling tower is re-established, high activity is still present. Is this to be expected? Explain. (0.7)

QUESTION 7.02 (2.50)

a. What are the two entry conditions to FR-H.1, 'Response to Loss of Secondary Heat Sink'? (0.5)

b. What two conditions, caused by a loss of secondary heat sink, calls for tripping the RCP's and initiating feed and bleed? (0.8)

c. In the response to inadequate core cooling, what three system parameters are checked to verify adequate core cooling has been recovered? (1.2)

QUESTION 7.03 (2.50)

Answer the following concerning E-0, Reactor Trip or Safety Injection:

a. The Main Turbine has not tripped and you attempt a manual trip as required, with no response. What additional action are you required to take in order to shutdown the turbine? (0.6)

b. List three plant conditions that require SI initiation? Include setpoints. (0.6)

c. What four parameters are checked to determine if SI flow should be terminated? (0.9)

d. Following an SI reset, what condition must be met before an automatic reinitiation of SI will occur? (0.4)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.04 (2.50)

Answer the following concerning the EOPs'rules of usage.

a. How does the operator know if the sequencial performance of subtasks within a procedure is required? (0.5)

b. When does monitoring of the STATUS TREE's begin? List two circumstances. (1.0)

c. The STA reports the following:
   1. Heat Sink - Orange Path
   2. Containment - Red Path
   3. Core Cooling - Orange Path
   4. Subcriticality - Yellow Path

   List the order in which the above conditions should be addressed? (1.0)

QUESTION 7.05 (2.00)

In order to maintain the plant at 100% power, work must be performed inside the containment in a radiation field of 850 MREM/HR gamma and 300 MREM/HR thermal and fast neutron. The maintenance man selected is 28 years old and has a lifetime exposure through last quarter of 48 REM on his NRC Form 4; additionally, he has accumulated 1.0 REM so far this quarter.

a. How long may the man work in this area without exceeding his 10 CFR limit? Show all work. (1.2)

b. During a declared emergency, this individual volunteers to enter a high radiation area and perform work necessary to prevent further effluent release. In accordance with the Station Procedures, what is his maximum allowed whole body exposure and whose authorization is needed? (0.8)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.06 (2.50)

Answer the following questions in regard to a reactor startup.

a. An idle RCP in a non-isolated loop, with the RCS at 250 F, shall not be started unless what specification is met? (0.5)

b. What should be done if criticality is not achieved by 500 pcm past the ECP? (0.5)

c. The Reactor Coolant System lowest operating temperature (Tavg) is not allowed to go below 541 F during a reactor startup. What are the four bases for this limit? (1.5)

QUESTION 7.07 (2.00)

Answer the following questions in regards to Operating Manual 1.6.4 Q, 'Response to Voids in Reactor Vessel.'

a. What symptoms would be indicative of a void in the reactor vessel? (1.0)

b. When should venting the reactor vessel take priority over containment hydrogen limits? (0.5)

c. When should venting the pressurizer take priority over containment hydrogen limits? (0.5)

QUESTION 7.08 (3.30)

a. What are the two reasons for stopping all RCP's in the case of a small break LOCA? (0.6)

b. What are two criteria for determining if RCP's should be stopped if HHSI pumps are running? (0.8)

c. In accordance with E-3, Steam Generator Tube Rupture, list four ways that a ruptured steam generator can be identified. (1.0)

d. What is the definition of adverse containment conditions? (0.9)

(***** CATEGORY 07 CONTINUED ON NEXT PAGE *****)

QUESTION 7.09 (3.00)

a. While moving fuel, a high-high alarm from a containment purge exhaust monitor occurs. What automatic actions follow besides an evacuation alarm? (1.6)

b. What is the basis for the requirement that two RHR loops be operable when water level is less than 23 feet above the vessel flange? (0.6)

c. Under what conditions is it permissible to stop RHR flow during refueling? (0.8)

QUESTION 7.10 (3.00)

a. Per AOP-43, High Reactor Coolant Activity, what three plant conditions can cause high RCS activity due to the release of irradiated corrosion products? (1.0)

b. In the event of high RCS activity, what is the reason for securing the following: containment sump pumps, primary drain pumps and their containment isolation valves, containment vacuum pumps, and containment isolation valves for reactor plant sample systems? (0.5)

c. An increased concentration of what two gases sampled from the VCT gas sample space would be indicative of failed fuel? (0.5)

d. Under what conditions can power operations continue if the specific activity of the primary coolant is greater than its Technical Specification limit? (1.0)

(***** END OF CATEGORY 07 *****)

QUESTION 8.01 (1.50)

The concentration of the boric acid solution in the Refueling Water Storage Tank (RWST) shall be verified once per 7 days in accordance with Technical Specification 3.5.5. The chemist sampled the RWST on the following schedule. (All samples taken at 1200 hours.)

April 1 --- April 8 --- April 16 --- April 24 --- April 31

a. EXPLAIN why or why not surveillance time interval requirements were exceeded on April 16. (0.75)

b. EXPLAIN why or why not surveillance time interval requirements were exceeded on April 24. (0.75)

QUESTION 8.02 (1.00)

What restrictions are placed on the manning and composition of the Fire Brigade? (1.0)

QUESTION 8.03 (2.50)

The RCS is heating up at 50 F per hour with the RCS presently at 325 F. Maintenance reports that Charging Pump 1B repairs will not be completed for one hour but that Charging Pump 1A is operable. Technical Specifications Action Statement allows 72 hours to repair an inoperable pump in Mode 3. What action, if any, should be taken? (2.5)

QUESTION 8.04 (2.50)

What action(s) (BOTH operational AND administrative) must be taken if the RCS-PRESSURE-Safety Limit is exceeded in accordance with Technical Specifications? Consider ALL Modes AND include applicable time limits in your answer.

QUESTION 8.05 (2.50)

Discuss the relationship between Limiting Conditions for Operations, Limiting Safety System Settings, and Safety Limits in terms of preventing release of radioactivity to the environment.

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.06 (2.00)

The plant is operating at 50% load and you are the on-shift Shift Supervisor. Explain what actions, if any, would need to be taken in regards to shift staffing for the following conditions. CONSIDER EACH CASE SEPARATELY.

a. Your on-shift BOP operator is seriously injured in the plant and you send him to the hospital for treatment. (1.25)

b. The on-coming STA calls and says he won't be in. (0.75)

QUESTION 8.07 (2.00)

The plant is operating at 75% power and the latest leak rate data shows:

13.2 GPM	-	Corrected RCS leakage rate
1.5 GPM	-	Leakage into the Pressurizer Relief Tank
1.2 GPM	-	Leakage into the Primary Drains Transfer Tank
3.4 GPM	-	Leakage through SI-23, RCS Loop 1A, cold leg isolation (Previous leakage rate was 1.6 GPM)
0.8 GPM	-	Total primary to secondary leakage
4.2 GPM	-	Leakage past RCP seals

What RCS leakage limits, if any, have been exceeded? Refer to attached Technical Specifications.

QUESTION 8.08 (2.50)

Per Technical Specifications, when does containment integrity exist? (2.5)

QUESTION 8.09 (2.00)

a. If an emergency condition develops, who assumes the role of the Emergency Director? (0.5)

b. When is an immediate deviation from Technical Specifications justified? (1.5)

(***** CATEGORY 08 CONTINUED ON NEXT PAGE *****)

QUESTION 8.10 (3.50)

a. What is the MINIMUM number of operable excore channels indicating AFD outside the target band before AFD is considered outside its target band by Technical Specifications? (0.5)

b. Assume the plant is operating at full power and the Axial Flux Difference (AFD) has been outside the target band for the last 5 minutes. What are the TWO actions specified which you may choose between to meet the Technical Specification requirements? Include time limitations. (1.0)

c. Assume that it is 0310 on 05/13/85 and the plant is presently at 45% power. Considering the AFD penalty history below, at what date and time may power be increased above 50%? EXPLAIN. (Show all work.) Assume no deviation outside the band after 0310 on 05/13/85.

DATE	TIME WENT OUT OF BAND	TIME BACK IN BAND	POWER
05/12/85	0310	0318	85%
05/12/85	1557	1637	65%
05/13/85	0148	0310	45%

(2.0)

QUESTION 8.11 (3.00)

a. What are the responsibilities of the Nuclear Station Operating Foreman (NSOF) at shift change? (1.0)

b. If clearance is needed to do maintenance on a piece of non-ESF equipment, how is permission granted to do the work and who gives the permission? (1.0)

c. Can a SRO soley authorize the installation of a jumper for non-Technical Specification related equipment? Explain. (1.0)

(***** END OF CATEGORY 08 *****)

(************* END OF EXAMINATION ***************)

$f = ma$

$v = s/t$

Cycle efficiency = (Network out)/(Energy in)

$w = mg$

$s = V_o t + 1/2\ at^2$

$E = mc^2$

$KE = 1/2\ mv^2$

$a = (V_f - V_o)/t$

$A = \lambda N$

$A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$

$w = e/t$

$\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\ \Delta P$

$t_{1/2}eff = \dfrac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\ \Delta m$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

$\dot{Q} = UA \Delta t$

$I = I_o e^{-ux}$

$Pwr = W_f \Delta h$

$I = I_o\ 10^{-x/TVL}$

$TVL = 1.3/u$

$P = P_o 10^{sur(t)}$

$HVL = -0.693/u$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\bar{\beta} - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\bar{\beta} - \rho)/\bar{\lambda}\rho]$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$T = \ell/(\rho - \beta)$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\bar{\lambda}\rho)$

$SDM = (1 - K_{eff})/K_{eff}$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\ell^* = 10^{-5}$ seconds

$\bar{\lambda} = 0.1\ \text{seconds}^{-1}$

$\rho = [(\ell^*/(T\ K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda} T)]$

$I_1 d_1 = I_2 d_2$

$P = (\Sigma\phi V)/(3 \times 10^{10})$

$I_1 d_1^{\ 2} = I_2 d_2^{\ 2}$

$\Sigma = \sigma N$

$R/hr = (0.5\ CE)/d^2 (meters)$

$R/hr = 6\ CE/d^2$ (feet)

Water Parameters

1 gal. = 8.345 lbm.

1 gal. = 3.78 liters

1 ft^3 = 7.48 gal.

Density = 62.4 lbm/ft^3

Density = 1 gm/cm^3

Heat of vaporization = 970 Btu/lbm

Heat of fusion = 144 Btu/lbm

1 Atm = 14.7 psi = 29.9 in. Hg.

1 ft. H_2O = 0.4335 $lbf/in.^2$

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps

1 kg = 2.21 lbm

1 hp = 2.54×10^3 Btu/hr

1 mw = 3.41×10^6 Btu/hr

1in = 2.54 cm

°F = 9/5°C + 32

°C = 5/9 (°F-32)

1 BTU = 778 ft-lbf

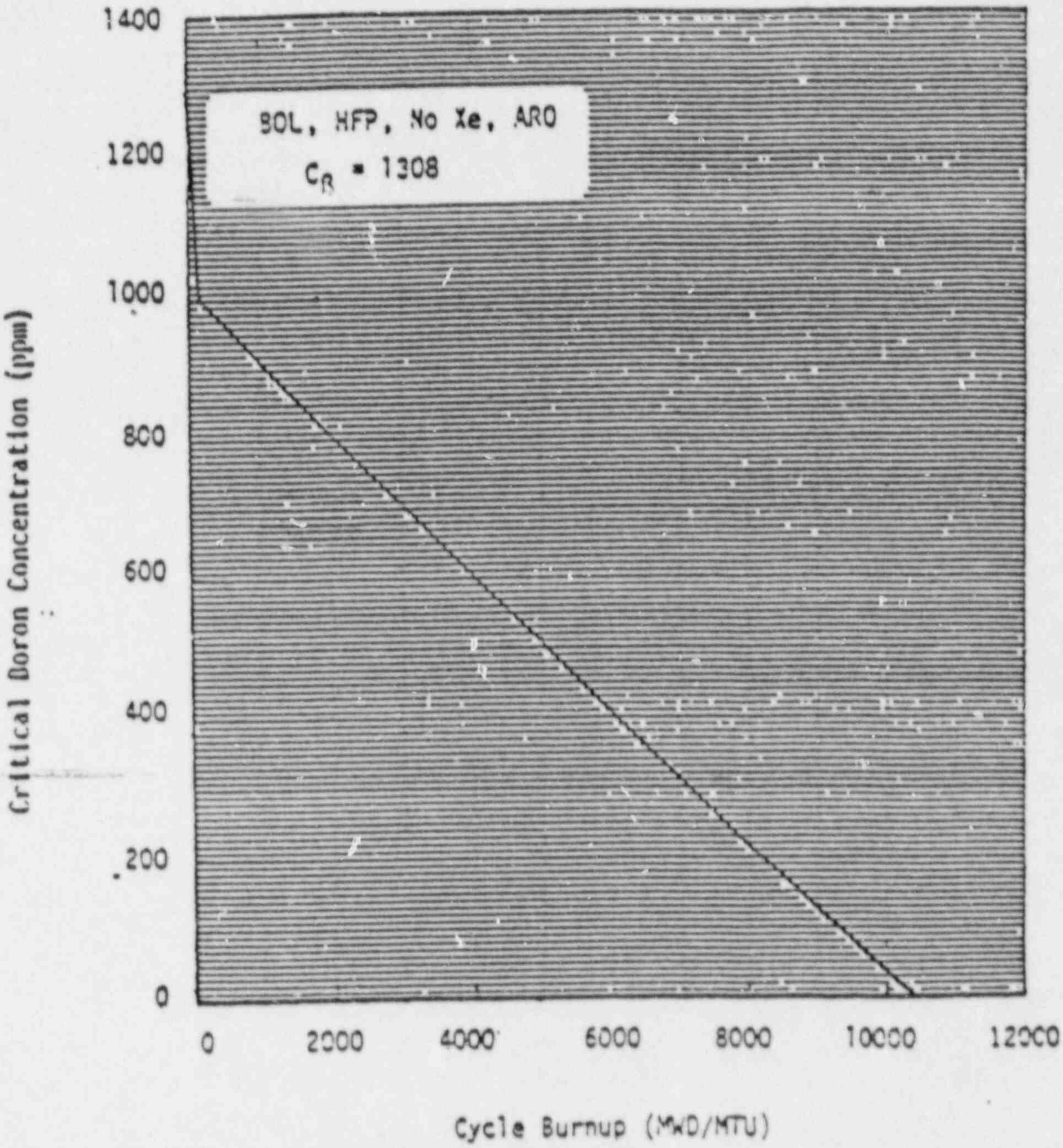
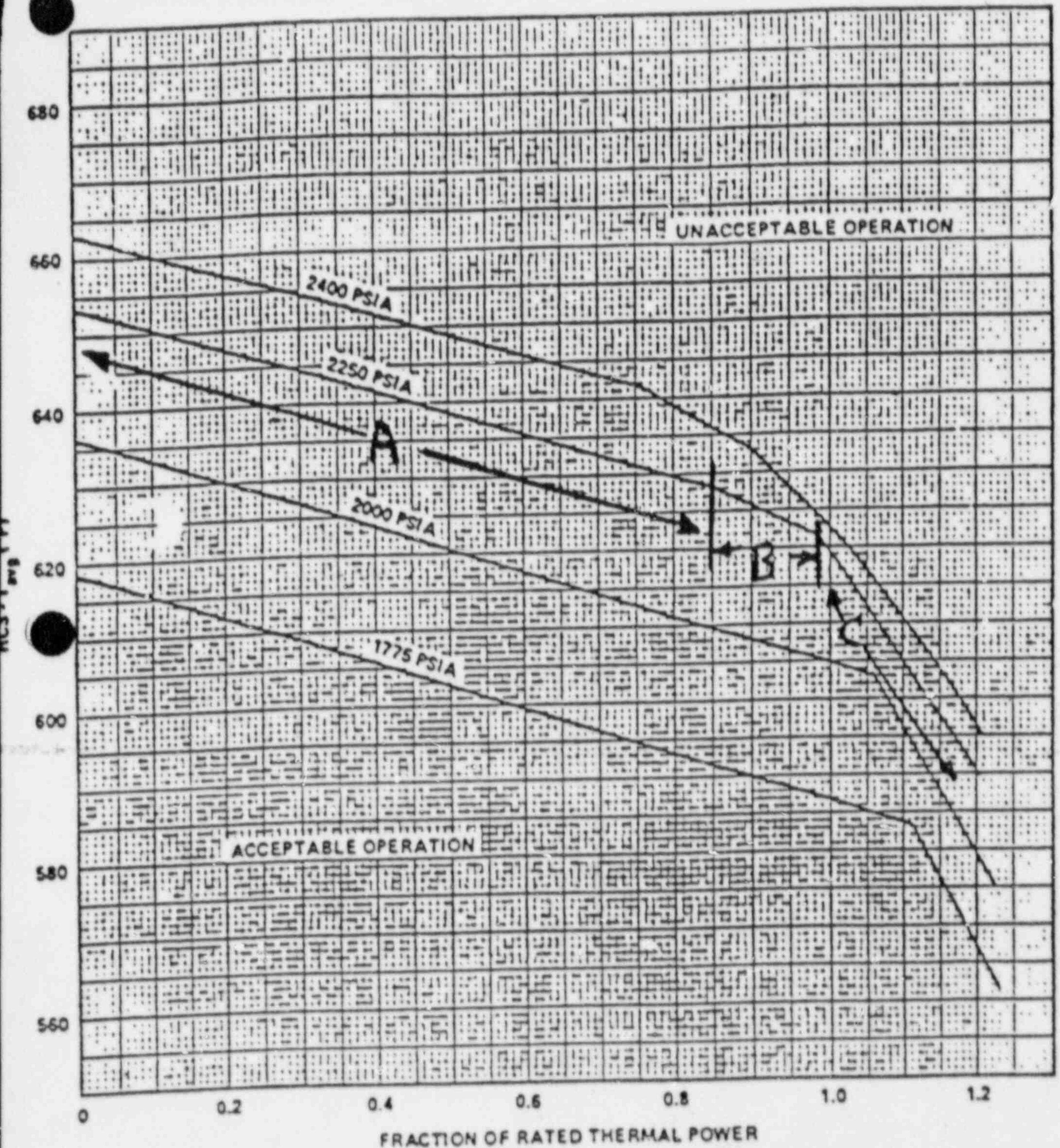


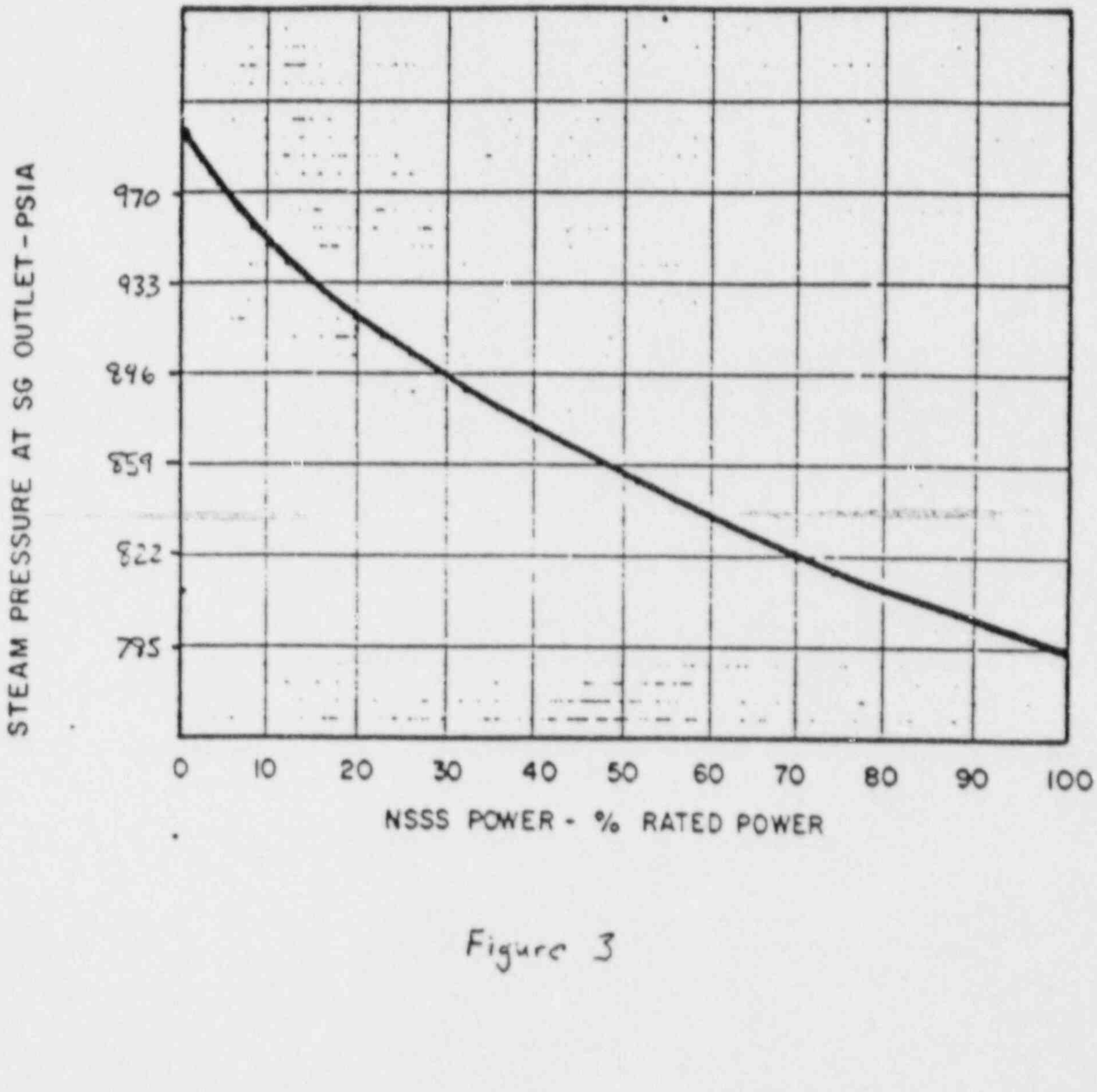
FIGURE 1

Critical Boron Concentration vs. Burnup
for HFP, ARO, Equilibrium Xenon Conditions



REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

Figure 2



STEAM PRESSURE AT SG OUTLET-PSIA
970
933
896
859
822
795
0
10
20
30
40
50
60
70
80
90
100
NSSS POWER - % RATED POWER

Figure 3

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 28 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2230 ±20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate and gaseous radioactivity monitor at least once per 12 hours.

REACTOR COOLANT SYSTEM

PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor coolant system pressure isolation valves shall be operational.

APPLICABILITY Modes 1,2, 3 and 4.

Action:

1. All pressure isolation valves listed in Table 4.4-3 shall be functional as a pressure isolation device, except as specified in 2. Valve leakage shall not exceed the amounts indicated.

2. In the event that integrity of any pressure isolation valve specified in Table 4.4-3 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in, and remain in, the mode corresponding to the isolated condition.[a]

3. If Specification 1 and 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4. The provision of specification 4.0.4 is not applicable for entry into Mode 3 or 4.

---

[a] Motor operated valves shall be placed in the closed position and power supplies deenergized.

4.4.6.3.1 Periodic leakage testing [a] on each valve listed in Table 4.4-3 shall be accomplished prior to entering Mode 1 after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months and prior to returning the valve to service after maintenance, repair or replacement work is performed.

4.4.6.3.2 Whenever integrity of a pressure isolation valve listed in Table 4.4-3 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

TABLE 4.4-3

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a)(b)] Allowable Leakage
Loop 1, cold leg	SI-23 SI-12	$\leq$ 5.0 GPM $\leq$ 5.0 GPM
Loop 2, cold leg	SI-24 SI-11	$\leq$ 5.0 GPM $\leq$ 5.0 GPM
Loop 3, cold leg	SI-25 SI-10	$\leq$ 5.0 GPM $\leq$ 5.0 GPM

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum test differential pressure shall not be less that 150 psid.

NRC RULES AND GUIDELINES FOR LICENSE EXAMINATIONS

During the administration of this examination the following rules apply:

1. Cheating on the examination means an automatic denial of your application and could result in more severe penalties.
2. Restroom trips are to be limited and only one candidate at a time may leave. You must avoid all contacts with anyone outside the examination room to avoid even the appearance or possibility of cheating.
3. Use black ink or dark pencil only to facilitate legible reproductions.
4. Print your name in the blank provided on the cover sheet of the examination.
5. Fill in the date on the cover sheet of the examination (if necessary).
6. Use only the paper provided for answers.
7. Print your name in the upper right-hand corner of the first page of each section of the answer sheet.
8. Consecutively number each answer sheet, write "End of Category _ " as appropriate, start each category on a new page, write only one side of the paper, and write "Last Page" on the last answer sheet.
9. Number each answer as to category and number, for example, 1.4, 6.3.
10. Skip at least three lines between each answer.
11. Separate answer sheets from pad and place finished answer sheets face down on your desk or table.
12. Use abbreviations only if they are commonly used in facility literature.
13. The point value for each question is indicated in parentheses after the question and can be used as a guide for the depth of answer required.
14. Show all calculations, methods, or assumptions used to obtain an answer to mathematical problems whether indicated in the question or not.
15. Partial credit may be given. Therefore, ANSWER ALL PARTS OF THE QUESTION AND DO NOT LEAVE ANY ANSWER BLANK.
16. If parts of the examination are not clear as to intent, ask questions of the examiner only.
17. You must sign the statement on the cover sheet that indicates that the work is your own and you have not received or been given assistance in completing the examination. This must be done after the examination has been completed.

18. When you complete your examination, you shall:

a. Assemble your examination as follows:

(1) Exam questions on top.

(2) Exam aids - figures, tables, etc.

(3) Answer pages including figures which are a part of the answer.

b. Turn in your copy of the examination and all pages used to answer the examination questions.

c. Turn in all scrap paper and the balance of the paper that you did not use for answering the questions.

d. Leave the examination area, as defined by the examiner. If after leaving, you are found in this area while the examination is still in progress, your license may be denied or revoked.

A. J. MORABITO
AJM

Section 5

5.01 a) It would overpredict (not conservative). The neutron source would tend to overpower the neutron detector such that it would be less likely to "see" the neutron multiplicati actually occurring in the core.

b) It doesn't. Critical rod position is based on core physics, time in core life, temperatu boron concentration, fission product inventor fuel load, etc but not on source level neutrons

c) The higher the insertion rate of positive reactivity, the lower the source range count level will be at criticality. This is due to less time for subcritical multiplication to increase the neutron flux level prior to the rods reaching the critical position.

-0.0

OK'M

5.02 a) 1. As boron concentration increases, differential boron worth becomes less negative. This is due to increased competition among boron atoms with regard to neutron absorption. Each atom has a lower probability of absorbing a neutron.

2. As moderator temperature decreases the differential boron worth increases. This is due to an actual increase in the boron population in the core due to moderator density increase

more boron
more competition?

more important thing more neutrons are thermalized ~~to~~ due to denser moderator and since boron is th absorber probability of abs ↑

-0.5

3. As fission products build up, differential boron worth decreases. This is due to increased competition among the various poisons in the core toward neutron absorption

-0.5

agm

5.02 a) 4. As core life progresses from MOL to EOL, differential boron worth increases. The actual concentration of boron is decreased during this period of time therefore each boron atom has an increased exposure to the neutron flux and therefore because of less competition with other poisons each boron atom has a greater chance of absorbing a neutron.?

b) Due to a build up of fission products which add their own poisoning affect to the core. Xe & Sm

5.03 a) 1. decrease

2. increase

-0.0

a/n

5.03 a) 3. increase

4. decrease

b) DNB could occur. Critical heat flux is the flux required to achieve DNB conditions in the existing local operating environment. If heat flux is greater than critical heat flux, fuel element surface temperatures will rise high enough above Tsat for the coolant to cause transition from nucleate boiling to film boiling.
Adverse conditions ? Clad failure

-0.5

c) T_{HOT} is being limited. Reactor power (primary) is calculated based on the equation:
$\dot{Q} = \dot{m} c_p (T_{HOT} - T_{COLD})$. By limiting T_{HOT} to less than T_{SAT} we ensure the validity of that calculation.

-0.5

AFM

5.04 a) Xenon concentration will increase for ~ 6 hrs to peak Xenon for 100%. It will then decrease down to equilibrium level for 75%. At the end of the 40 hr. period it will be at 75% power equilibrium concentration.

The increase will be caused by decay of iodine atoms which were produced at the 100% power level which is a greater concentration than at the 75% power level.

Burnout?

The decrease is due to Xenon decay occurring faster than Xenon production once the 100% iodine inventory decay decreases.

production from fission

The concentration levels off at 75% equilibrium level as Xenon production again equals Xenon decay. Burnout?

-0.5

page 5

-0.5

agm

5.04 b) Rods will be withdrawn during the Xenon build-up to peak concentrations over the first 6 hrs. Rods will be inserted during the remainder of the 40 hrs until 75% Xenon equilibrium is achieved.

5.05 a) decrease. As the core ages, $Pu^{239,241}$ isotopes build up. They have lower value of the weighted delayed neutron precoursor fraction. When they add their contributio to the weighted fraction for U^{235} there is a net numerical decrease

b) ~~larger~~ at EOL

-0.0

ajm

5.06 a) 1) $\Delta P \sim (\Delta \text{ speed})^2$

$$\frac{\text{Speed}_1}{\text{Speed}_2} \cong \sqrt{\frac{\Delta P_1}{\Delta P_2}}$$

$$\text{Speed}_2 = \text{Speed}_1 / \sqrt{\frac{\Delta P_1}{\Delta P_2}}$$

$$= \text{Speed}_1 / \sqrt{\frac{1800}{1200}}$$

$$= \text{Speed}_1 / 0.8$$

$= 1.22$ factor increase in speed

2) Power $\sim (\Delta \text{ speed})^3$

Therefore power increase by the factor

$(1.22)^3 = 1.82$

-0.0

agm

5.06 b) 215 ft NPSH = (215)(0.4335) psi

= 93 psi

Required NPSH = Psat + required P

$NPSH = \frac{P_{suct} - P_{sat}}{\rho}$ $P_{sat} = 205.29$ psia

$1/\rho = 0.018416\ ft^3/lbm$ = 205.3 psia + 93 psi

$P_{suct} = 286.3$ psia = 298.3 psia or 283.6 psig

-0.3

formula [illegible]

5.07 Worth more if stuck out while all other rods are inserted. Rod worth is proportiona to the square of relative flux at the rod tip. In the channel with the rod stuck out the flux would peak higher than flux in the other channels thus causing that rod to be worth more.

-0.3

afm

5.08 a. increase

b. decrease

c. decrease

d. no change increase -0.4

e. no change increase -0.4

-0.8

ajm

5.09 a) Moderator temperature increase causes rod worth to increase. The density of the moderator decreases as temperature increases. This allows more leakage of neutrons to the rods. The less dense moderator also allows the neutrons to remain at epithermal energies longer and the rods are good epithermal absorbers.

b) Since reactor power is determined by steam demand it will remain constant however moderator and fuel temperature will decrease to add enough positive reactivity to balance the negative reactivity from the dropped rod. Core burnout will tend to become uneven. A Xenon oscillation could be induced. If Xenon oscillations occur, core temperature and power production will becom erratic.

-0.0

AJM

5.10 a) The increased steam demand causes moderator and fuel temperatures to decrease which adds positive reactivity to the core. Core power production increases.

+ρ from MTC counter by doppler (-0.5)

Assume: 15% power increase requires 270 PCM

$\alpha_M = -10$ PCM/°F mod.

$\alpha_D = 1$ PCM/°F fuel

fuel temp change = 2°/% power

then $\Delta T_{fuel} = 2°/\% \times 15\% = 30°$

and $\alpha_D = 1$ PCM/°F $\times 30° = 30$ PCM

And $\Delta T_{mod} = \dfrac{270 \text{ PCM} - 30 \text{ PCM}}{10 \text{ PCM/°F}}$

$= -24°$

new $T_{avg} = 568 - 24 = 544°$

-0.5

5.10 b) S/G pressure for 85% = 804 psia

arm

$$Q = UA(T_{avg} - T_{atm})$$

$$\frac{Q_1}{Q_2} = \frac{(T_{avg_1} - T_{atm_1})}{(T_{avg_2} - T_{atm_2})}$$

$T_{avg_1} = 568°$ $T_{atm_1} = T_{sat}$ for 822 psia $= 522°$

$T_{avg_2} = x$, $T_{atm_2} = T_{sat}$ for 804 psia $= 518°$

therefore $\frac{70\%}{85\%} = \frac{(568-522)}{(x-518)}$

$$x = \frac{85(568-522)}{70} + 518$$

$$= 55.9 + 518$$

$$= 573.9°$$

end of Section 5

-0.0

A.J. MORABITO

Section 6 AJM

6.01 a) The gamma flux is greater in proportion to the neutron flux at low power levels like those during startup than it is at high power levels. Therefore a loss of compensating voltage would cause a noticeable increase in detector output at low neutron levels. It wouldn't be noticeable at power operation.

b) 1) Power range high flux
2) Power range high positive rate
3) Control group auto rod withdrawal stop

PR computer deviation
~~[illegible] PR high setpoint neutron flux high~~
PR low setpoint flux deviation or auto defeat
Computer alarm rod deviation / SEQ NIS PR tilts -0.6

6.02 a) The temperature has been reduced by the non-regenerative heat exchanger
deleted

-0.6

Appr

6.02 b) 1) RM-BD-100 SGBD tank draining
2) RM-BD-101
3) RM-GW-108 Gas waste
Process vent
4) RM-GW-109 — assuming air ejector discharge going to cooling tower discharge ok
Process vent
5) RM-GW-110
6) Chemistry sample room area monitor

c) Air ejector discharge switches from gaseous waste to containment.

6.03 a) 1) by monitoring CCR Surge tank level decrease and no increase in containment sump
2) by monitoring containment sump level increase
3) by monitoring area radiation monitors
4) by observing increased flows on the various component cooling water headers
5) by observing increased temperatures on various equipment.
6) by observing shutdown of liquid waste equipment

-0.0

agm

6.03 b) 1) Automatic trip/close of thermal, barrier CCR ~~inlet~~ outlet valves on high flow

check valves upstream of thermal barrier Hx close
Piping between valves is designed for 2485 psig

-1.0

2) Ability to manually isolate the thermal barriers

3) Auto back-up pump start on decreasing CCR pressure

4) Separation of thermal barrier CCR supply from pump motors and lube oil cooling supply.

6.04 a) to maintain back pressure on the #1 seal thus promoting sufficient leakage thru #1 seal to cool + lubricate the seal and supply seal water to #2 seal

b) There is a key interlock on the breakers. Only 1 unique key fits either breaker. The key must be inserted in the breaker to "rack" the breaker on the bus. When the breaker is on one of the busses, the key is captured in that cubicle

-1.0

ayn

04 c) Used when rapid dilution of boron needed. Disadvantage - dilutes VCT concentration also Disadvantage - adds large amounts of non hydrogenated water to the RCS

-0.5

6.05 a) Also assume rods in automatic. Rod control sees (T_{avg} - T_{ref}) mismatch. Rods step out to increase T_{avg}. Since temperature mismatch is greater than 5°, rods steps out at 72 steps/min. This causes high positive SUR which causes reactor trip

delete

6.06 a) The overpressure protection system. Two PORV's keyed to A loop + C loop WR pressure transmitters. Letdown pressure control valve PCV-CH-145

-0.4

b) Pressurizer temperature < ~~470°~~ 475
A loop wide range pressure < 430 psig
A loop wide range pressure > 630 psig

-0.9

apm

6.06 c) The appropriate OST is performed which verifies operability of the two PORV's that will be used. When the OST is complete, key locks have been switched which remove the PORV's from pressurizer pressure transmitter control and places them (one on each) on the A loop + C loop wide range pressure transmitters

d) Alarm sounds and nitrogen back-up is automatically fed to the PORV's

6.07 a) They provide the first means of protection for Tavg increases above program to prevent violating the safety limit curve

Ensures that secondary pressure will be limited to within its design pressure during the most severe transient

-0.5

-0.5

apm

.07 b) 1) To isolate the faulted steam generator

2) To prevent blowdown of the non-faulted steam generators thru the break

1) minimize +ρ effects of RCS cooldown

2) limit pressure rise in containment

-1.0

c) EOL most severe due to feedback from relatively large negative moderator temperature coefficient

HZP most severe because the cooldown will be equivalent to that which would occur at HFP but all parameters are farther from protective trip points

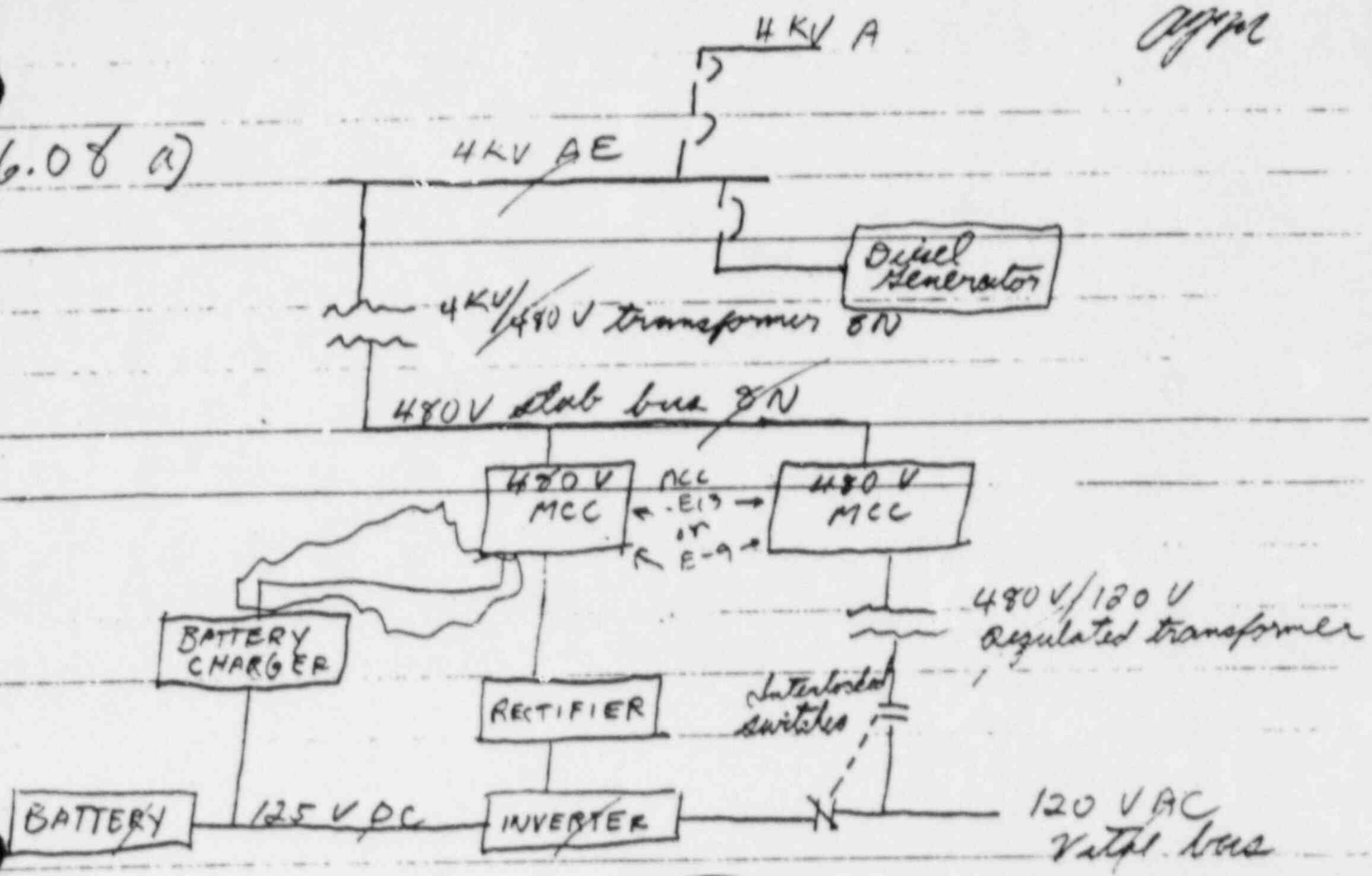
HZP because greatest mass in SG results in largest RCS cooldown

-0.25

-1.25

AJM

6.08 a)



-0.15

b) To supply DC control and field flash to Diesel Generators

Preserve power to the DG auto loading sequence circuits

-0.4

c) 4 KV Emergency bus voltage at rated voltage

Permissive signal from associated UV devices -0.4

-0.95

page 19

gpm

~~QS pump~~ Quench spray pump running

6.09 a) 1 hour running time elapsed, RWST level at setpoint.

-0.4

b) To permit ~1000 gpm flow to containment so that recirc spray and LHSI sumps will have adequate NPSH and to permit total addition of NaOH solution from chemical addition tank

Reduce QS flow to minimize negative pressure when containment returns

-0.8

c) To ensure that leakage thru a ruptured tube is into river water. This allows detection of the leak, isolation of the leaking heat exchanger, and keeps the water inventory in containment from exceeding design limits as it might if river water leaked into the heat exchange

want to avoid dilution to ensure necessary SDM

-0.8

d) No, reset of CIB signal locks out the auto start signal. CIB will have to be manually restarted.

-0.8

Yes CIB signal will auto start QS pumps

-2.8

appr

6.10 a) Above limit - N_2 gas could be injected into RCS during accumulator injection
increase amount of accumulator water carried out break (-0.4)

below limit - insufficient injection of accumulator volume to meet analyzed requirements of FSAR

b) 19' 2½" level on 2/4 RWST level instruments and SI signal (-0.2)

c) ① Low head SI pump suction valves from containment sumps open, low ⑤ head SI pump suction from RWST close, ② low head recirc to RWST close, ③ low head inlet to high head pumps open. After time delay, ④ high head pump suction from RWST closes
(-0.5)

d The outside recirc spray pumps can be valved in to back-up the low head pump

End Section 6

-1.1

A. J. MORABITO

ajm

Section 7

7.01 a) The discharge should be stopped. The permit should be re-analyzed by two separate qualified Radiological Controls supervisors. The discharge can then recommence. Grab samples will have to be obtained periodically during the discharge

Independent verification of release rate calculation and valve lineup

-0.5

b) Depends on what the initating problem was. If it was actual high activity and problem was corrected by reducing the discharge rate then high activity will be present on the rad monitor until reduced discharge mixes with dilution to reduce total activity.

-0.5

OPM

1.02 a) 1) Total feed flow < 350 GPM and < 5%
narrow range level on all steam
generators

2) Aux feed flow < 350 GPM in step 19
of E-0 immediate actions

b) 1) RCS pressure ≥ 2335
2) Wide range level < 10% on any 2
steam generators

c) 1) Core exit thermocouples < 1200° and
decreasing
2) Subcooling adequate per appropriate
attachment
3) Core exit thermocouples < 100° with
RVLIS full range < 40%
RCS hot leg temperatures

-0.4

-0.4

agm

?.03 a) Attempt to run back the turbine. If that doesn't work, close the main steam isolation valves.

close non-return valves

close the main steam bypass valves

-.03

b) 1) RCS pressure < 1845 psig

Containment pressure > 1.5 psig

Main steam line pressure < 510 psig

These are auto SI signals which require manual initiation if the auto initiation does not occur.

2) Pressurizer level cannot be maintained > 5%

3) Appropriate subcooling margin cannot be maintained

-0.3

AfM

7.03 c) 1) Pressurizer level > 5%
2) RCS pressure stable or increasing
3) Subcooling greater than appropriate attachment
4) Heat sink available

d) Reactor trip breakers must be closed.

7.04 a) If subtasks are lettered - sequential order is required. If bulleted - sequential order not required.

b) 1) When transitioning from E-0 to another emergency procedure

2) When told to by E-0 (step 23 or thereabouts)

0.0

appr

7.04 c) 1) Containment
2) Core Cooling
3) Heat Sink
4) Subcriticality

7.05 a) 1) Field = (850 MREM/hr + 300 MREM/hr)
= 1150 MREM/hr

2) Lifetime limit = 5 (28 - 18)
= 50 Rem/lifetime

Man's limit = (50 - 48) = 2 Rem during year

Since he already has 1 Rem he can only receive 1 REM additional ~~exposure~~.

Time in work area = $\frac{1 \text{ REM}}{1.15 \text{ REM/hr}} \times 60$ min/hr

= 52 min.

0.0

ajm

7.05 b) 1) 25 Rem max allowed

2) Must be authorized by the Emergency Director.

7.06 a) Loop T_H and T_C are within 25° of the highest operating loop temperatures or pressurizer level < 60%
R&L.1

b) Rods should be inserted to position that corresponds to -500 PCM then recalculate ECP

c) 1) To ensure moderator temperature coefficient is within its analyzed range.

2) To ensure pressurizer can be operable with a steam bubble

3) To ensure reactor vessel temperature > RTNDT

4) To ensure nuclear instrumentation is operating within normal operating temperature range

-0.0

ag/p2

7.07 a) 1) Unexplained increase in pressurizer level
abnormal pressure response to charging and spray

2) RVLIS < 100% with no RCP's running
Indication of departure from subcooled conditions

-0.25

b) When directed by procedure such as
FR C.1
Potential interruption of ~~core~~ core cooling

-0.5

c) Never
Not able to maintain pressure control

-0.5

7.08 a) 1) To conserve ~~mass~~ mass in RCS
2) To reduce heat input to RCS
3) To preserve pump operability
Preclude core uncovery from RCP's tripping at later time

-0.3

-1.55

apm

7.08 b) 1) ΔP between RCS + highest steam generator $\leq$ 145 psig (510 psig adverse)

2) Loss of RCP running support systems such as seal injection or CCR

c) 1) Radiation surveys of steam piping

2) Chemistry sampling

(3) Unusual level or pressure indication in the steam generator

4) Level remaining constant or increasing with reduced feed flow

High S/G blowdown line radiation

-0.25

d) Containment pressure $\geq$ 5 psig

Containment radiation dose rate $\geq 10^5$ R/hr

Containment total integrated dose $\leq 10^6$ R

-0.25

apm

7.09 a) Inlet damper to main filter bank opens
Bypass dampers close. BVPS normal
system arrangement has purge supply shutdown
and purge exhaust going thru main filters.
containment purge exhaust & supply dampers close
containment purge & exhaust fan damper closes
-0.6

b) To ensure adequate decay heat removal
capability

c) 1) Flow can be stopped for up to 4 hrs
if work requiring visual acuity is
being done in the vicinity of the
RCS hot legs

2) Flow can be stopped for periods of
1 hr every 8 hrs to aid refueling
operations.

-0.6

OPN

7.10 a) 1) Addition of chemicals to RCS such as hydrogen peroxide

2) Injection of O_2 gas

3) Drop in pH

4) RCP shutdown / startup

5) Plant heatup / cooldown

b) To prevent release of high activity water from containment. To ensure that those systems are operated with appropriate manual surveillance when they are returned to service

c) Xenon and Krypton

d) If it can be shown to be caused by waterlogging of fuel elements due to power level changes. The activity must be monitored and logged. It must be shown to be decreasing and it must be within limits in 24 hrs. 48

End of Section 7

-0.0

A.J. MORABITO

App 2

Section 8

8.01 a) They were not exceeded because the surveillance interval can be extended up to 25% if total of 3 consecutive intervals does not exceed 3.25 times total interval

b) The requirement was exceeded. Total time interval from April 1 to April 24 is 23 days. Total time allowed during that period is 7(3.25) = 22.75 days

8.02 Must include 5 qualified members. Cannot include the 3 members of shift complement required for safe shutdown.

-0.0

AJM

8.03 Stop the heatup before RCS temperature equals 350° to prevent changing modes while in an action statement requirement.

8.04 Modes 1-~~4~~ 2 restore pressure to within limits and be in hot standby in 1 hour.

Modes ~~5~~ 3,4,5 +6 Reduce pressure to within limits in 5 minutes.

-0.1

Notifications must be made to NRC within ~~1~~ 24 hr. Must also notify Plant Manager, General Manager Nuclear Operations, and Chairman of ORC. Must prepare written report and submit to OSC within 24 hrs. Must analyze effect on plant. Report must be submitted to NRC within 30 days. NRC permission required to start up.

-0.1

aTM

8.05 Limiting conditions for operations provide and ensure adequate equipment operability and minimum equipment operability to permit power operation while meeting the assumptions of the FSAR with regard to equipment availability to mitigate the consequences of an accident

Limiting safety system settings ensure that operating parameters remain within the boundaries from which the occurrence of accident conditions have been assumed in the FSAR

LSS Auto action prior to reaching LSSS then safety limits will not be exceeded!

Safety limits provide the absolute limits which if violated could lead to a release of radioactivity. They are based on plant design values and if they are exceeded, re-evaluation of plant conditions and re-analysis of plant paramete must be conducted.

-0.0

AFM

8.06 a) Must call out replacement if cannot meet staffing requirements with other people on shift. Can continue operating if replacement is likely within 2 hrs. If replacement not likely within 2 hrs must commence controlled shutdown. Shutdown can be stopped when replacement arrives.

b) The on-duty STA will be held over until replacement can be called in. STA could work up to 16 hrs without special permission but must be relieved after that unless Plant Manager waives that requirem

6.0

AJM

8.07 The 1 GPM unidentified leakage limit is exceeded. Total identified leakage is 6.9 GPM which is in limits but 13.2 total − (6.9 + 4.2 at seals) = 2.1 GPM unidentified. Also the leakage rate of SI-23 is out of limit since it replaced the previous margin by greater than 50%.

8.08 When all valves that are required to be closed in accident conditions are capable of being closed by an operable automatic closure system or are:

- Closed by manual valves, blind flanges, or de-energized automatic closure mechanisms secured in the closed position

and

The personnel airlock is operable and both doors closed (except for personnel passage)

and

equipment hatch is sealed and bolted

and

when containment leakage rates meet the appropriate Tech Spec requirements

0.0

apm

8.09 a) The NSS

b) If there is an imminent danger to the health and safety of the public and there is no appropriate instructions immediately apparent within approved procedures, Tech Specs, etc.

8.10 a) ~~1~~ 2 -0.5

b) Restore AFD to within limits or reduce power to ≤90%. Both actions together must be complete within 15 minutes total whichever you choose or any combination of the two.

-0.5

Appr

8.10 c) 5/12 8 min penalty at 0318
40 min penalty at 1637
5/13 82 min × 1/2 = 41 min penalty
at 0310
total for 24 hr period = 89 min

Power can be increased when penalty deviations are < 60 min in previous 24 hrs. That will occur when the 1637 penalty on 5/12 has been reduced to 18 min. That should occur 1637-18 or 1619 on 5/13/85

8.11 a) Complete the shift turnover checklist. Sign log as being properly relieved. Have personnel on his shift identified to him. Be aware of plant status, planned operations etc by virtue of shift turnover checklist. Perform board walk down, etc again by virtue of completing shift turnover checklist.

F0.C

ajm

8.11 b) Permission granted by NSS who does so by signing the approval signature space on the clearance permit. and MWR

-0.33

c) Yes, however a safety analysis must be conducted by the OSC before the jumper is installed.

End of Section 8

0.33

~~0.0~~

MASTER

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 5.01 (3.00)

a. Overpredicts (not conservative) (0.5). The source neutron flux dominates the detector reading until the flux level from core multiplication is higher than if the source was further from the detector (0.5).

b. It doesn't (0.5). The critical rod position reflects the positive reactivity necessary to bring the reactor critical and is independent of source magnitude (0.5).

c. The faster the rate, the lower the source range counts at criticality (0.5) due to the reduced time for subcritical multiplication (0.5).

REFERENCE
BVPS Reactor Theory Manual Chapter 5, pp 36,47,49

3.1 001 000 K 5.18 4.3
010 K 5.16 3.5

ANSWER 5.02 (3.50)

a. 1. Delta boron worth becomes less negative (0.25) due to increased competition for neutrons by more boron atoms (0.5).

2. Delta boron worth becomes more negative (0.25) because more neutrons are thermalized due to denser moderator and since boron is a 1/v absorber, the probability of absorption increases (0.5).

3. Delta boron worth becomes less negative (0.25) due to increased competition for neutrons by the poison atoms (0.5).

4. Delta boron worth becomes more negative (0.25) due to reduced boron concentration from MOL to EOL (0.5).

b. Negative reactivity caused by the buildup of Xe and Sm (0.5).

REFERENCE
BVPS Reactor Theory Manual Chapter 8, p 34, 45, 37

3.1 001 000 K 5.20 3.2
K 5.28 3.8
K 5.30 3.1

ANSWER 5.03 (3.00)

a. 1. Decrease (0.25)
   2. Increase (0.25)
   3. Increase (0.25)
   4. Decrease (0.25)

b. Clad failure (melting, burnout) probability is greatly increased because film boiling will reduce the heat being transferred from the fuel (1.0).

c. Coolant outlet temperature is limited (to below saturation temperature) (0.5). If coolant becomes saturated then there will be no change in RCS hotleg temperature and thus no indication of core power (0.5).

REFERENCE
BVPS Thermodynamics Manual, Chapter 7, pgs. 14-17, 19

3.4 003 000 K 5.01 [illegible].9
3.2 002 000 K 5.09 4.2
K 5.01 3.[illegible]

ANSWER 5.04 (2.00)

a. After the power decrease, the production of xenon from fission (0.25) and from the decay of iodine (0.25) is greater than the removal by decay of xenon (0.25) and burnout by flux (0.25). After five hours, the removal rate is greater than the production (0.25) and positive reactivity is being added until equilibrium at about 40 hours (0.25).

b. Rods will need to be withdrawn for about 5 hours (0.25) and then inserted for the next 35 hours (0.25).

REFERENCE
BVPS Rx Theory Manual chapter 7 pgs. 13-16

3.1 001 000 K 5.13 4.0
K 5.32 3.5

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 5.05 (1.50)

a. Decreases (0.5) Pu 239 concentration increases and Pu 239 has a smaller beta. (0.5)

b. Larger SUR (0.5)

REFERENCE
BVPS Reactor Theory Manual, Ch 5, pg 14-17

EO 3

001/000 K5.47 2.9/3.4 pg 3.1-3

ANSWER 5.06 (2.50)

a. The speed needs to increase to 1.225 times the original speed to raise the discharge head from 1200 to 1800 PSIA.

$H1/H2 = (N1/N2)^2 = 1200/1800$, $N2/N1 = (1800/1200)^{1/2} = 1.225$ (0.75)

The horsepower needs to increase to 1.837 times the original horsepower to raise the discharge head from 1200 to 1800 PSIA.

$P2/P1 = (N2/N1)^3 = (1.225)^3 = 1.837$ (0.75)

b. NPSH = (Psuct - Psat)/density (0.3)
Psat = 205.29 PSIA (0.2)
1/density = 0.018416 ft**3/lbm (0.2)

215 ft lbf/lbm = (P - 205.29 lbf/in**2) (144 in**2/ft**2) (0.018416 ft**3/lbm)

P = 286.3 PSIA or 271.6 PSIG (0.3)

REFERENCE
BVPS Thermo Manual chapter 4 pgs. 14, 21d, 33

Appendix pg. A-9 2.6
3.6

ANSWER 5.07 (2.00)

The stuck rod would be worth more (0.5). Reactivity worth is proportional to the relative flux squared (0.5). For a dropped rod, the flux is depressed adjacent to it (0.5) whereas if the same rod was stuck out, while the others were inserted, it would be exposed to a much higher flux than the flux in the rest of the core (0.5).

REFERENCE
BVPS Rx Theory Manual chapter 8 pgs. 14-16

3.1 000 003 EK 1.03 3.8
005 EK 1.05 4.1

ANSWER 5.08 (2.00)

a. Increase
b. Decrease
c. Decrease
d. Increase
e. Increase (0.4 each)

REFERENCE
BVPS Thermo Manual chapter 6 pg. 20

3.5 039 000 A 1.05 3.2
3.2 002 000 K 5.11 4.2

ANSWER 5.09 (2.50)

a. As moderator temperature increases, the migration and thermalization lengths of neutrons in the core increases, therefore more neutrons will migrate to the control rods (0.5) thus increasing their worth (0.5).

b. Reactor power would remain constant (0.5). The negative reactivity inserted by the dropped rod would be countered by positive reactivity inserted by MTC (0.5) since Tave would be lower (0.5).

REFERENCE
BVPS Rx Theory Manual chapter 6 pgs. 16, 20
chapter 9 pg. 3

3.1 001 000 K 5.10 4.1

K 5.29 3.9
000 003 EK 1.16 3.2

ANSWER 5.10 (3.00)

a. Reactor power will increase (0.5) to match secondary power from positive reactivity inserted by MTC (0.5) which will be countered by negative reactivity from power defect as power increases (0.5).

b. Q = U A (Tave - Tstm)
70% = U A (568 - 521.3) = 70/85 U A (Tave - 518.7) (0.5)
{521.3 and 518.7 are from the steam tables for their corresponding pressures in Figure 3} (0.5)
Tave = ~~570.9 F~~ (0.5)
575.4 F

REFERENCE
BVPS Thermo Manual chapter 7 pgs. 1-4

3.5 039 000 K 5.08 3.6
A 2.05 3.3

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 6.01 (2.00)

a. The ratio of gamma to neutron flux is greater (0.5)

b. PR comparitor deviation
NIS PR high setpoint rod stop block rod w/D
NIS PR high setpoint neutron flux high
NIS PR neutron flux rate high
PR low setpoint flux deviation or auto defeat
Computer alarm rod deviation / SEQ NIS PR Tilts (5 of 6, 0.3 each)

REFERENCE
BVPS OM 2.1 pg. 16
2.2 pg. 7
NS-8 figures 8, 9

3.9 015 000 K 3.01 4.3
K 6.02 2.9

LP-SQS-2.1 3,4g

ANSWER 6.02 (2.50)

a. Allow for the decay of N-16 (0.4)

b. Steam generator blowdown sample monitor
AFWP turbine exhaust monitor
Main steam safety valve effluent monitor
Steam generator blowdown tank discharge monitor (0.4 each)

c. The condenser air ejector discharge will be diverted to the containment (0.5).

REFERENCE
BVPS OM 43.1 pgs. 9, 15, 21
AOP-42 pgs. 1, 2

Appendix pg. A-6 2.7
3.3 000 037 EK 2.02 2.4
3.3 000 037 EK 3.10 3.7
3.9 073 000 K 1.01 3.9

2336 RCS 6
2353 MSS 10

ANSWER 6.03 (2.20) component temperature alarms ~~(.35)~~ (.35) and surge alarm. (.35)

a. ~~Flow indicators parallel to the leak will indicate a lower than normal flow and abnormally high component temperatures. OR affected component temperatures would be much lower than normal if the leak was downstream of the component or much higher than normal if the leak was upstream of the component (0.7).~~

b. High flow will cause RCP thermal barrier CCR outlet valves to close
Pressure buildup will seat check valve
Piping between valves is designed for 2485 psig (0.5 each)

REFERENCE
AOP-20 pg. 1
BVPS OM 15.1 pg. 16

3.10 008 000 K 3.01 3.5
3.3 000 009 EK 3.15 3.2

LP-SQS-6.3 1

ANSWER 6.04 (2.00)

a. To ensure required adequate back pressure in the RCP seals (0.5)

b. Key interlock will only allow one breaker to be racked in at a time (0.5)

c. For load follow and permits the dilution of water to follow the initial xenon transient (0.5) but using it adds large amounts of non-hydrogenated water to the RCS (0.5).

REFERENCE
BVPS OM 7.1 pg. 38
7.2 pgs 1, 3

3.1 004 000 K 1.06 3.1
K 2.03 3.5
K 5.01 3.3
3.2 002 000 K 1.06 4.0

LP-SQS-7.1 3,5

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 6.05 (2.00)

Control rods will automatically move outward (0.4) due to temperature error and power mismatch error during the transient (0.4). With a small MTC, reactor power will rise (0.4) causing a (C-2) overpower rod stop (0.4) and power overshoot results in an OTdT or OPdT trip despite Doppler feedback (0.4).

REFERENCE
BVPS NS-10 pgs 3 to 12
NS-8 pg.39

3.1 001 000 A 1.02 3.4
3.5 045 010 K 4.21 3.2
3.9 017 000 K 4.02 4.3

2352 MSS 3

ANSWER 6.06 (2.70)

a. Letdown pressure control valve (~~MOV-CH-142~~) (0.4)

b. Will not open at RCS pressure > 430 psig
Will auto close at RCS pressure > 630 psig
Will not open if pressurizer vapor temperature > 475 F (0.5 each)

c. Manually placing two keylock switches in their automatic position (Enables the PORVs' low pressure setpoint) (0.4)

d. The backup supply are two nitrogen filled accumulators (0.4)

REFERENCE
BVPS OM 10.1 pgs. 2, 15
10.2 pgs. 6, 7
6.1 pgs. 52, 53

3.2 006 000 K 4.08 3.5
3.4 005 000 K 4.01 3.2 ; K 4.07 3.5
3.3 010 000 K 4.03 4.1
3.8 078 000 K 3.02 3.6

LP-SQS-10.1 4 RCS PZR Pressure relief system 7

ANSWER 6.07 (2.50)

a. Ensures that secondary system pressure will be limited to within its design pressure during the most severe transient (0.5).

b. 1. Minimize positive reactivity effects of RCS cooldown associated with the blowdown (0.5)
   2. Limit pressure rise within containment during a steam break in containment (0.5)

c. Hot Zero Power (.25) because of the greatest mass in the SG results in the largest RCS cooldown (.25)
   EOL (.25) because MTC is at its maximum negative value (.25)

REFERENCE
T/S B 3/4 7-1
T/S B 3/4 7-3
FSAR 14.1-35 to 38

3.5 000 040 EK 3.01 4.5
EK 2.01 2.5
EK 1.05 4.4
3.5 039 000 K 4.05 3.7

Objectives PGS-10-17

ANSWER 6.08 (2.30)

a. See sketch (1.5)

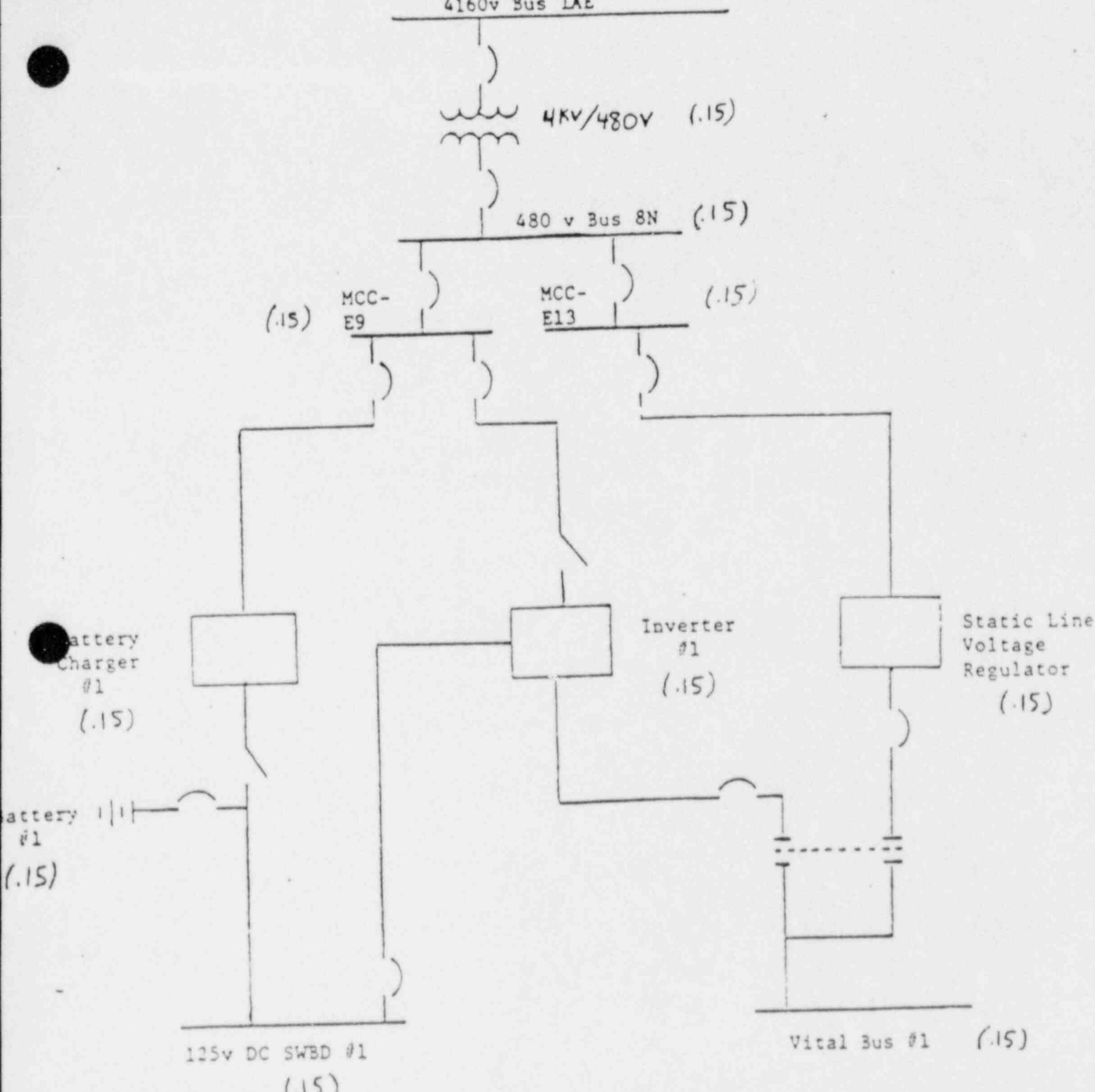
b. To preserve power to the Diesel Generator Auto Loading Sequence Circuits (0.4)

c. Permissive signal from associated undervoltage devices (0.4)

REFERENCE
BVPS OM 37.1 pgs 78, 79
BV exam Bank Question 6-4 a

3.7 062 000 K 4.09 2.9
K 4.03 3.1

LP-SQS-36.1 2.7


4160v Bus 1AE
4KV/480V (.15)
480 v Bus 8N (.15)
MCC-E9
(.15)
MCC-E13
(.15)
Battery Charger #1
(.15)
Inverter #1
(.15)
Static Line Voltage Regulator
(.15)
Battery #1
(.15)
125v DC SWBD #1
(.15)
Vital Bus #1
(.15)

ANSWER 6.09 (3.20) –

a. Associated quench spray pump running (0.4)
   RWST low low level (0.4)

b. Reduces quench spray flow to minimize negative pressure when containment returns to subatmospheric pressure following a LOCA (0.8)

c. Only out leakage can occur and dilution of borated water by river water in containment is not possible which ensures necessary shutdown margin (0.8)

d. ~~No (0.3). CIB initiate pushbuttons must be depressed before the pumps will automatically restart (0.5).~~

(0.3) quench spray pumps

REFERENCE
BVPS OM 13.1 pgs. 2, 8, 12, 19

3.6	103	000	K	1.08	3.8
	026	000	K	4.04	4.1
	026	000	K	1.02	4.1
	026	020	K	4.03	4.3

?-SQS-13.1 3,4,5

NSWER 6.10 (3.60)

a. Higher pressure would tend to increase the amount of accumulator water carried out the break (0.4). Lower pressure results in less rapid delivery of accumulator water to the reactor tending to delay core recovery (0.4).

b. SI signal (0.2) and 2/4 low level on RWST (0.2)

c. 1. Containment sump to LHSIP's suction valves (SI-860A,B) open
2. LHSIP's miniflow isolation valves (SI-855A,B,C,D,) close
3. LHSIP's discharge valves to HHSIP's suction (SI-863A,B) open
4. HHSIP's suction from RWST (CH-115B,D) closed (0.3 for valves
5. LHSIP's suction from RWST (SI-862A,B) closed 0.5 for order)

d. Manually align an outside recirculation spray pump (0.4)

REFERENCE
BVPS OM 11.1 pgs. 3, 7; Fig 11-12 ; BV Exam Bank Question 6-11 a,b
BVPS NS-13 pgs. 9-13

3.2 006 000 K 6.02 3.9
K 4.06 4.2
? 1 005 000 K 3.05 3.8

LP SQS-11.1 3,6

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 7.01 (1.70)

a. Decay Tanks sampled and analyzed (0.5)
Independent verification of release rate calculation and valve line-up (0.5)

b. Activity should decrease as soon as water in the pipes from the discharge of the pumps is purged ~~(0.4)~~. ~~If not, then the release should be stopped (0.3).~~ (0.7)

REFERENCE
TS pg. 3/4 3-63
BVPS OM 19.4 pg. 9
17.4 pg. 67

3.11 068 000 Sys gen 4 3.3
071 000 Sys gen 5 4.0

ANSWER 7.02 (2.50)

a. While performing E-0 if total AFW flow < 350 gpm (0.2) all
Heat Sink Red Path - SG narrow range level in ~~least one~~, SG < 5% (.15)
- Total feedflow to SG's < 350 GPM (.15)

b. Wide range level in two SG's < 10% (0.4)
Pressurizer pressure greater than 2335 psig (0.4)

c. RVLIS full range indication (> 61%) (0.4)
At least two RCS hot leg temperatures (< 350 F) (0.4)
Five hottest exit TC's (< 1200 F) (0.4)

REFERENCE
BVPS EOP FR-C.1 pg. 11
FR-H.1 pgs. 1, 2

3.4 000 074 EK 3.11 4.4
3.5 000 054 EK 3.04 4.6

LP-SQS-53A-FR-H EO 2
LP-SQS-53A-FR-S EO 2

ANSWER 7.03 (2.50)

a. (1) Close main steam trip valves ~~(0.3)~~ Runback the turbine
(2) Close non-return valves ~~(0.3)~~ Close the mainsteam bypass valves (0.15 each)

b. (1) Pressurizer pressure < 1845 PSIG
(2) Containment pressure > 1.5 PSIG.
(3) Steamline pressure < 510 PSIG (0.2 each)

c. (1) RCS subcooling criteria met (0.23)
(2) Feed flow to intact SG's(> 350 GPM)or
Narrow Range level in at least one intact SG (0.23)
(3) RCS pressure - (stable or increasing) (0.22)
(4) PZR level(> 5%) (0.22)

d. Reactor trip breakers must be closed (0.4)

REFERENCE
BVPS EOP E-0 pgs. 3, 5, 14, 17

3.1 000 007 EK 3.01 4.6

LP-SQS-53A-E-0 EO 1,3

ANSWER 7.04 (2.50)

a. Letters denote sequencial importance, bullets do not (0.5)

b. As directed in E-0 (0.5)
When transferring out of E-0 (0.5)

c. 2,3,1,4 (1.0)

REFERENCE
EOP Ex Vol pgs. 3,6,8

SWPWGKA 22 4.3

LP-SQS-53A-Intro EO 1b, 2a

ANSWER 7.05 (2.00)

a. 5(N-18) = 50 REM (0.2)
Total lifetime to date = 48 + 1 = 49 REM (0.2)
Total lifetime available = 50 - 49 = 1 REM (0.2)
Total this quarter available = 3 - 1 = 2 REM (0.2)
Lifetime is more restrictive than quarterly limit

0.85 REM/HR gamma + 0.30 REM/HR neutron =1.15 REM/HR dose rate
1.0 REM/1.15 REM/HR = 0.87 HRS = 52 MIN (0.4)

b. 25 REM whole body one time exposure (0.4)
Emergency Director (0.4)

REFERENCE
10 CFR 20.4; 101
BVPS RCM pg. 9

System wide and plant wide generic K&A (SWPGK&A) 10 3.9

ANSWER 7.06 (2.50)

a. The actual pressurizer water level is less than 60% or
The secondary water temperature of each SG is less than 25 F above each of the in-service RCS cold legs temperature (0.5 for either)

b. Return bank to 500 pcm below ECP and recalculate ECP (0.5)

c. Ensures that:
1. The moderator temperature coefficient is within its analyzed temperature range
2. The protective instrumentation is within its normal operating range
3. The pressurizer is capable of being in an operable status with a steam bubble
4. The reactor vessel is above its minimum NDTT temperature
(0.3 each)

REFERENCE
BVPS OM 50.4 pg. 30& 10
TS pg. B 3/4 1-2
TS 3.4.1.6, BVPS OM 6.4 pg. 3

3.4 000 000 Sys Gen 5 3.9
3.1 001 010 A 2.07 4.2

LP-SOS-6.3 5

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

2336 RCS 8 -

ANSWER 7.07 (2.00)

a. Abnormal pressurizer pressure and level responses to charging and spraying (0.5)
Indication of departure from subcooled conditions (0.5)

b. If the potential for interruption of core cooling with hydrogen in the vessel exists (0.5)

c. If the pressurizer bubble is interferring with the ability to maintain pressure control (0.5)

REFERENCE
BVPS OM 6.4 pgs. 121, 126

3.6 028 000 K 5.01 3.9
3.3 000 009 EA 2.01 4.8
EA 2.38 4.3

LP-SQS-6.9 5

ANSWER 7.08 (3.30)

a. 1. Prevent excessive inventory loss (0.3)
2. Preclude core uncovery from RCP's tripping at a later time (0.3)

b. 1. Highest RCS SG D/P < 145 PSI (0.4)
2. No CCR Flow to RCP's (0.4)

c. 1. Unexpected increase in S/G narrow range level.
2. High S/G sample radiation.
3. High S/G steamline radiation.
4. High S/G blowdown line radiation. (0.25 each)

d. Containment pressure > 5 psig or containment radiation > 100000 R/hr or integrated containment radiation > 1000000 R (0.9)

REFERENCE
BVPS Exec Vol E-0 step 23 pg. 33
BVPS EOP E-3 pgs. 2, 3
E-0 Attachment 6

3.3 000 038 EK 3.06 4.5

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

LP-SQS-53A-E-1 EO 3
LP-SQS-53A-E-3 EO 1

ANSWER 7.09 (3.00)

a. Containment purge exhaust and supply dampers close
Containment purge to exhaust fan damper closes
Main filter bank bypass dampers close
Main filter bank inlet dampers open (0.4 each)

b. Ensure that a single failure will not result in a loss of heat removal capability (0.6)

c. May be stopped (for one hour per eight hour period) for core alterations in the vicinity of the hot legs (0.8)

REFERENCE
AOP-22 pg. 1
TS pgs. B 3/4 9-2, 3/4 9-8

3.4 000 025 Sys gen 5 3.9
3.11 034 000 K 3.01 2.9

LP-SQS-10.1 9

ANSWER 7.10 (3.00)

a. Plant heatup, plant cooldown, abnormal pressure/temperature transients
Chemical shock (0.33 each)

b. Preclude potential high airborne and increased radiation levels in the auxiliary building (0.5)

c. Xenon and iodine (0.25 each) or Krypton

d. ~~Operations may continue (up to 48 hours) provided that operation under these circumstances shall not exceed 10% of the unit's total yearly operating time (1.0)~~ Operation may continue provided the activity does not exceed the limit on the Tech spec curve (TS Figure 3.4-1) and is within specification [illegible] within required time

REFERENCE
AOP-43 pgs. 1, 2
TS pg. 3/4 4-18

3.11 000 076 EK 3.01 3.1
EK 3.05 3.6
Sys gen 5 3.6

2236 RCS 5.8

ANSWERS -- BEAVER VALLEY 1&2 -86/07/22-SILK, D.

ANSWER 8.01 (1.50)

a. Interval requirement not exceeded [0.25]. Eight days does not exceed 1.25 times the specified interval [0.5].

b. Interval requirement exceeded [0.25]. The last 3 consecutive intervals exceed 3.25 times the specified interval [0.5].

REFERENCE
TS pg. 3/4 0-2

SWPWGK&A 5 3.9

LP-SQS-11.1 7

ANSWER 8.02 (1.00)

The Fire Brigade shall not include three members of the minimum shift crew necessary for the safe shutdown of the unit or any personnel required for other essential functions during a fire emergency (1.0).

REFERENCE
SAP pg. 8
TS pg. 6-1

SWPWGKA 19 4.2

ANSWER 8.03 (2.50)

(The Technical Specifications require that all LCO's be satisfied prior to entry into an operational mode.) ~~(0.75)~~ 1.0 Since you are about to enter Mode 3 ~~(0.75)~~ the heatup must be discontinued (.75) and Tave held at less than 350 F until Charging Pump 1B is proven operable ~~(0.5)~~ .75.

REFERENCE
TS pg. 3/4 0-1; TS pg. 3/4 5-3

3.4 005 000 Sys gen 5 4.0

LP-SQS-7.1 9

ANSWER 8.04 (2.50)

Modes (.15) 1&2-- Be in (.2) HSB with pressure within (.2) limits in one (.2) hour. (.75)

Modes 3,(.25) 4,5-- Reduce pressure to within (.25) limit in 5 (.25) minutes. (.75)

All Modes-- Notify the (.4) NRC, Manager of Nuclear (.3) Operation, and (.3) ORC (within 24 hours). (1.0)

REFERENCE
TS pgs. 2-1, 6-12

SWPWGK&A 5 3.9

2336 RCS 8

ANSWER 8.05 (2.50)

LCO's indicate lowest performance level of equipment required for safe operation of the facility (0.5). If proper automatic action occurs prior to reaching Limiting Safety System Settings, then Safety Limits will not be exceeded (1.0). If Safety Limits are not exceeded then fuel and RCS integrity will be maintained (1.0).

REFERENCE
TS pgs. B 2-2,3
10 CFR 50.36 c

SWPWGK&A 5 3.9

ANSWER 8.06 (2.00)

a. You may operate for up to two hours with one less than minimum complement (0.75) provided that immediate action is taken to bring the complement up to minimum (0.5).

b. The on-shift STA will have to wait for a relief to come in (0.75).

REFERENCE
TS pg. 6-4

SWPWGKA 23 3.5

ANSWER 8.07 (2.00)

RCS Pressure Isolation Valve Limits exceeded. (1.0)

UNIDENTIFIED Leakage limits exceeded. (1.0)

REFERENCE
TS 3.4.6.2; TS 3.4.6.3

3.2 002 020 Sys gen 5 4.1

2336 RCS 8

ANSWER 8.08 (2.50)

All penetrations required to be closed during accident conditions are either:
Capable of being closed by an operable containment auto-isolation valve system (0.5), or
Closed by manual valves, or blind flanges (0.5)
All equipment hatches are closed and sealed (0.5)
Both doors in each personnel air lock are properly closed unless being used at which time at least one air lock door shall be closed (0.5) and air lock leakage is within limits (0.5)
The containment leakage rates are within limit (0.5)

REFERENCE
TS 1-2

SWPWG K&A 5 3.9

LP-SQS-12.1 5

ANSWER 8.09 (2.00)

a. Shift Supervisor (0.5)

b. In an emergency when this action is immediately needed to protect the public health and safety and no action consistent with the license conditions and Tech Specs that can provide adequate or equivalent protection is immediately apparent. (1.5)

REFERENCE
SAP pg. 49
EPP pg. 5-3

SWPWGK&A 36 4.7

ANSWER 8.10 (3.50)

a. 2 (0.5)

b. Within 15 minutes (0.2)
1. Restore the indicated AFD to within the target band (0.4), or
2. Reduce the thermal power to <90% of rated thermal power. (0.4)

c. Accumulated penalty over the past 24 hours is 89 minutes. (1.0) The penalty will be reduced to 60 minutes at 1618 minutes on 05/13/85 and then power may be increased. (1.0)

85%	0318-0310	=	8			(0.25)
65%	1637-1557	=	40			(0.25)
45%	0310-0148	-	82/2	=	41	(0.5)
					--	
					89 min. total penalty	

05/13/85, from 1557; 81 min left -60 - 21 min -> 1618 05/13/85 (1.0)

REFERENCE
TS 3.2.1; TS pg. B 3/4 2-2

3.9 015 020 Sys gen 5 3.9

LP-SQS-2.1 3.7

ANSWER 8.11 (3.00)

a. Review plant status by inspection of control room instrumentation
Review entries in logs
Conduct a briefing with the off-going NSOF using Shift Relief Turn-over Checklist (0.33 each)

b. The Nuclear Shift Supervisor (0.33) expresses permission via the Equipment Clearance Permit (0.33) and Maintenance Work Request (0.33)

c. No (0.5). Technical evaluation has to be done with OSC concurrence prior to installation (0.5).

REFERENCE
SAP pgs. 12, 27, 23

SWPWGK&A 14 4.0

DUQUESNE LIGHT COMPANY
Nuclear Division
Training Manual

LESSON PLAN

Core Safety Limits	4.0
Course	Course Hours
George Borlodan	January 2, 1984
Instructor	Date
	LP-LRT-II-54
Approved By:	Lesson Plan No. (Sequentially From I)

References To Be Quoted: BVPS Thermodynamics, Fluid Flow, and Heat Transfer Manual; Explanation of Reactor Core Thermal and Hydraulic Safety Limit Curves; Steady-State Thermal-Hydraulic Considerations; BVPS Technical Specifications.

Items Issued: (Attach copy of all passouts, quizzes, etc.)
Handout attached.

Introduction:

1. Purpose:
   Review basic characteristics of convection heat transfer, clarify basic concepts associated with DNB; review core safety limit curve and core protection parameters.

2. Motivation: (Discuss how you plan to motivate students)
   Explain how the limits provided by the nuclear control and protection systems ensure that core safety limit curve limitations are not violated.

3. General Outline: (List detailed outline Section I)
   Introduction; Coolant Heat Transfer; Boiling Heat Transfer; Reactor Core Safety Analysis; Reactor Core Safety Limit Curves; Appendix.

4. General Student Goals: (List detailed student objectives Section II)
   The student shall have an indepth understanding of the DNB phenomenon, the application of the term DNBR, and the development, basis, and use of the core safety limit curve.

MODULE III

BEAVER VALLEY POWER STATION

UNIT I

1983/84 LICENSE RETRAINING

CORE SAFETY LIMITS

TABLE OF CONTENTS

PERFORMANCE OBJECTIVES

Terminal Objective: The nuclear power plant operator shall have an indepth understanding of the term "Departure from Nucleate Boiling (DNB)", the application of the term "Departure from Nucleate Boiling Ratio (DNBR)", and the development and basis for the core safety limit curve, including the use of this curve in ensuring that the minimum DNBR requirement is met.

Enabling Objectives: Upon completion of this lesson on Core Safety Limits the student shall be able to:

1. Define the following terms: film coefficient; bulk fluid; laminar flow; turbulent flow; linear heat rate; heat flux; critical heat flux; DNB; and DNBR.

2. Draw the boiling curve, labelling the various regions and describing in detail the mode of heat transfer in each region.

3. Use the heat flux (Q") versus channel enthalpy change (Δh) curve to differentiate between the two boiling crisis mechanisms: DNB and dryout.

4. State the minimum requirement for DNBR and discuss how this requirement allows for safe operation of the plant.

5. Sketch a typical fuel cell and a thimble cell and state which is more limiting from a DNB standpoint.

6. Briefly discuss the design bases in consideration of both fuel temperature and core flow.

7. Discuss important characteristics of the core safety limit curve and explain the use of this curve in ensuring that the minimum DNBR requirement is met.

The student shall be able to accomplish the above with at least an 80% accuracy on either an oral or written examination.

CORE SAFETY LIMITS

The purpose of this discussion is to clarify several basic concepts which are often misunderstood or misconstrued by students of nuclear power technology:

1. The definition, and more important, the precise physical implications of the term "Departure From Nucleate Boiling (DNB)".

2. The definition and application of the term "Departure From Nucleate Boiling Ratio (DNBR)".

3. The development and basis for the safety limit curve, including the use of this curve in ensuring that minimum DNBR requirements are met.

Prior to addressing the DNB phenomenon, it is necessary to briefly review the basic characteristics of heat transfer from a heated, stationary solid to a fluid flowing adjacent to its surface.

COOLANT HEAT TRANSFER (Figure 1)

Coolants remove heat from fuel elements in the core, from thermal shields, pressure vessels, etc. and transfer this heat to an intermediate or secondary coolant or to the working fluid in a heat exchanger. The heat transferred, $\dot{Q}$ (Btu/hr), between the coolant and a surface (of a fuel element, heat exchanger tube, etc.) is given by Newton's law of cooling:

$$\dot{Q} = h_{film} A (T_w - T_b)$$

where:

h_{film} = coefficient of heat transfer by convection, Btu/hr - ft^2°F;

A = surface area, ft^2, across which heat flows;

T_w = wall surface temperature, °F;

T_b = fluid bulk temperature, °F.

Figure 1 illustrates the temperature and velocity profile for a reactor coolant channel. In this particular instance, A would be the total surface area of the exposed cladding, T_w would correspond to the clad surface temperature, and T_b is the temperature in the coolant mainstream where the flow is well mixed. The value of h_{film} represents the magnitude of resistance to the flow of heat from the clad surface, through the various boundary layers, and into the coolant mainstream (bulk fluid). A large film coefficient indicates very little resistance to the flow of heat, or in other words, a large $\dot{Q}$ for a relatively small ΔT. A small film coefficient means the resistance to the flow of heat is great, and thus to achieve the same $\dot{Q}$ as above, ΔT would have to be much greater. Because of size limitations, as well as the large thermal output of power reactors, values of h_{film} much higher than those encountered in conventional engineering practice are required.

In most reactor applications the coolant is force-circulated and forced convection heat transfer coefficients apply. Turbulent flow conditions also predominate in such cases. Thus, h_{film} is a function of coolant physical characteristics (thermal conductivity, specific heat, viscosity, etc.) as well as operating conditions (speed, pressure) and flow channel geometry. This dependence on operating conditions and geometry necessitates the development of special correlations to yield the appropriate heat transfer coefficient for the coolant under specified flow conditions. There are various flow conditions which are of particular interest with regard to a nuclear power reactor, and consequently empirical correlations (experimentally arrived at) have been developed to evaluate the heat transfer coefficient corresponding with these conditions. An example of such a correlation is one developed by the Westinghouse Electric Corporation for determining the heat transfer coefficient at a precise location with subcooled water being force-circulated parallel to rod bundles (as is the case in water-cooled reactors using fuel rods). The correlation takes the following form:

$$h_F = \left(\frac{k}{D_e}\right) \left(0.042 \frac{S}{D} - 0.024\right) \left(\frac{D_e V\rho}{\mu}\right)^{0.8} \left(\frac{c_p \mu}{k}\right)^{1/3}$$

where:

k = fluid thermal conductivity, Btu/hr · ft · °F;

D_e = channel equivalent diameter, ft^2;

S = fuel rod pitch (distance between rod centerlines);

D = fuel rod diameter, in;

V = fluid velocity, ft/sec;

ρ = fluid density, lbm/ft^3;

μ = fluid viscosity, lbm/hr · ft;

c_p = fluid specific heat capacity, Btu/lbm · °R; and

$1.1 < \frac{S}{D} < 1.3$

Of the parameters listed above, it is important to note that the values of k, ρ, and c_p are temperature/pressure dependent, while μ varies not only with temperature/pressure, but also with flow conditions. Coolant velocity, on the other hand, is dependent on flow conditions, channel geometry, and on the nature (characteristics) of the heated solid surface.

It is not important that the student have an understanding of the basis and development of this correlation, or any other of the correlations utilized in the thermal-hydraulic design of a commercial power plant. Due to the complexity of the flow conditions in the various heat exchangers in the plant, not to mention in the core itself, an indepth discussion of such correlations is beyond the scope of this lesson. However, as will be seen later, a semi-empirical correlation developed by the Westinghouse Electric Corporation and commonly referred to as the "W-3 Correlation" was used to predict DNB throughout the core area, as well as the limitations on minimum DNBR. It is therefore vital that the student have an appreciation of the complexities and tremendous uncertainties associated with experimentally derived convection heat transfer correlations, and hence the conservativism of the limits which are imposed to ensure fuel and clad integrity.

Three terms commonly used in fluid dynamics problems (and which students might be more familiar with) are the Nusselt number (Nu), the Reynolds number (Re), and the Prandtl number (Pr), where:

$$Nu = \frac{h_F \ D_e}{k}$$

$$Re = \frac{D_e \ V\rho}{\mu}$$

$$Pr = \frac{c_p \ \mu}{k}$$

Using these quantities, the sample correlation presented earlier can be expressed as:

$$Nu = (0.042 \ \frac{S}{D} - 0.024) \ Re^{0.8} \ Pr^{1/3}$$

The student need not be concerned with the physical significance of the Nusselt, Reynolds, and Prandtl numbers, however, the Appendix to this lesson relates supplementary material which may be of interest.

BOILING HEAT TRANSFER

Many of the heat exchangers in the plant are involved directly in thermodynamic processes and experience a change in phase of one or both of the fluids involved. The steam generator and condenser are prime examples of this mode of heat exchange. It should be noted that in such instances the expression

$$\dot{Q} = \dot{m} \ c_p \Delta T$$

is useless since the temperature of a boiling liquid will not change; also, c_p will change drastically with the phase change. To evaluate these cases, an energy balance utilizing enthalpies is required on the boiling fluid.

Under certain circumstances, boiling can take place on the wall of the heat source. This occurs continuously on the secondary side of the steam generator tubes, and to some extent in the core area between the reactor coolant and the fuel rod cladding (0.5% voiding, nominal, at 100% power). A common misconception of students is that all boiling is dangerous and thus undesireable, however, some types of boiling can greatly aid in heat transfer with no undesireable consequences or drawbacks. On the other hand, certain types of boiling can cause severe damage to components and are highly undesireable. It is necessary to first address the characteristics of boiling heat transfer in general prior to proceeding with a discussion of the concerns for various types of boiling in the core region.

The Boiling Curve (Figure 2)

Consider the typical example of a heated tube submerged in a saturated water coolant. As the tube surface temperature rises above that of the saturated water, small bubbles form at nucleation sites on the tube wall. These bubbles increase in size and break loose from the surface because of buoyant forces, thus disturbing the boundary layer and resulting in a significant heat removal capability for a small ΔT. As the ΔT increases due to a higher wall temperature, the bubbles formed are large and appear at a greater rate, so that heat removal is further enhanced. However, beyond this point a higher ΔT will caus the bubbles to coalesce and form patches of vapor on the tube surface before the bubble mass is replaced by the liquid film reestablishing itself. While the vapor patch is covering the surface, heat removal is significantly reduced in that area because conduction through steam and radiation at low temperatures are not very efficient heat transfer mechanisms.

If the wall temperature is increased still further, the vapor blanket on the tube surface stabilizes and heat transfer is severely restricted. This condition may be regarded as a potential failure condition for the heat exchanger. The tube wall will increase in temperature to that of the hot fluid, and will be extremely susceptible to thermal shock when the liquid film again wets the outside surface. Figure 2 shows the heat transfer capability from a heated surface to water with boiling effects included. The curve is a plot of the heat transfer coefficient, h_F, versus the temperature difference, ΔT , between the heated surface and the bulk fluid. It is generally referred to as the "boiling" curve because the following regimes of boiling can be identified:

A. In this range, heat transfer is by pure convection. Subcooled liquid will rise to the surface of the water pool and evaporation will take place.

B. In this range, bubbles begin to form on the surface of the heated element at locations referred to as "nucleation sites", hence the term "nucleate boiling". The bubble formation occurs due to the surface temperature of the element exceeding the saturation temperature of the liquid. At the lower end of the nucleate boiling range, the bulk liquid is subcooled and the bubbles will be condensed after breaking loose from the heated element. This phenomenon is known as subcooled nucleate boiling and is the type of boiling which occurs in the core area under normal operating conditions. Heat transfer from the element to the bulk liquid is enhanced for two reasons:

   1. As the vapor bubbles break through the boundary layer, they create turbulence in the layer which improves the ability of the boundary layer to transfer heat.

   2. The latent heat of vaporization that was absorbed during bubble formation is deposited directly into the bulk liquid when the bubbles are condensed, which adds to the amount of heat transferred.

In summary, subcooled nucleate boiling is desireable from a heat transfer standpoint.

At the top end of the nucleate boiling range, the bubbles no longer collapse in the bulk liquid since its temperature is now at, or above, saturation. This type of boiling is termed saturated nucleate boiling or bulk boiling, and is the type of boiling which occurs in the Boiling Water Reactor (BWR) by design. Due to the disturbance of the boundary layer as the bubbles break away from the surface of the heated element, bulk boiling is also desireable from a heat transfer standpoint.

C. Due to the very high surface temperature, unstable vapor patches (vapor gets displaced by liquid, which is then displaced by vapor, and so on) form on the surface and heat transfer becomes less efficient. This is known as transition boiling or partial film boiling. The peak of the curve in Figure 2 represents the precise point where we would leave the saturated nucleate boiling regime and enter into partial film boiling with any further increase in ΔT, and thus it is referred to as the point of Departure from Nucleate Boiling, or DNB. The vital significance of this point will be addressed in greater detail when we examine a more commonly used curve: the fuel rod heat flux versus clad/coolant ΔT curve.

As surface temperature continues to increase, a stable film is formed and heat transfer capability decreases further. This is referred to as stable film boiling, or simply film boiling. It will be seen later that once we reach the DNB point, surface temperature increases suddenly and dramatically (i.e. ΔT rises very quickly) due to impaired heat transfer out of the heated surface, and thus we instantaneously "jump" into the stable film boiling regime. At some point, the surface temperature becomes so high that radiation heat transfer through the vapor film becomes a significant mechanism, and heat transfer capability begins to increase. Eventually, a high rate of heat removal can be established, but now radiation is the mechanism, and surface temperature is extremely high. Generally, if a high rate of heat transfer is required, such as when the heated surface contains a heat source (as is the case with a fuel rod), the temperature at the surface will exceed the melting point of the material.

Departure from Nucleate Boiling (Figure 3)

While the boiling curve as presented in Figure 2 is applicable to the low pressure side of all heat exchangers, it is also applicable to heat transfer from the fuel rods in the core. For this application, the "h_F" axis will be relabeled the Q", or heat flux, axis since now a certain rate of heat removal is required for both power generation and fuel cooling. Again, the same regions of the curve can be identified, only now the transition to film boiling takes on new significance.

Point A on Figure 3 is once again the Departure from Nucleate Boiling point, or DNB point, and the corresponding heat flux is referred to as the DNB heat flux, or more commonly, the Critical Heat Flux (CHF). This will be discussed in more detail in the next section. Since the curve represents the behavior of a physical system, the operating point must lie on the curve. If some heat flux higher than the DNB heat flux is produced in a fuel rod, then the operating point shifts from DNB over to point B on the curve, and the corresponding surface temperature rises to several thousand degrees F. This high temperature will probably damage the fuel rod: the cladding may soften, melt, or develop pinholes through which the fuel material may come in contact with the coolant. Obviously the DNB heat flux is to be avoided, and accordingly nuclear cores operate at some point lower on the curve. Just how far below the DNB point a plant operates, and how close it approaches DNB during plant transients, is a major portion of core protection analysis.

Critical Heat Flux

The heat flux Q" which a fuel rod can produce is equal to the cladding/coolant temperature difference (ΔT) times the heat transfer coefficient (h_F). This coefficient is not a constant. As the heat flux increases from lower values up to the point where nucleate boiling occurs, h_F tends to increase. More heat can be removed from the fuel rod per square foot and per hour because the boiling action stirs the water. In fully developed nucleate boiling, the wall temperature is determined by the heat flux and pressure, but is insensitive to coolant velocity.

Figure 3 shows roughly how the heat flux ($Btu/hr \cdot ft^2$) is related to the cladding/bulk coolant temperature difference, ΔT. It should be noted by tracing the curve how the beginning of nucleate boiling boosts the heat flux per unit ΔT increase. At some critical flux value, the steam being produced forms an unstable insulating layer over the cladding surface resulting in a sharp decrease in the heat transfer coefficient, and consequently, an increase in the clad surface temperature. Point A on the Figure 3 curve is the vital point. At this point, the transition from nucleate boiling to film boiling is being made, as described earlier. The heat flux corresponding to this point is called the Critical Heat Flux (CHF) and the situation is known as the boiling crisis. Recall that CHF is also referred to as the DNB heat flux, and the boiling crisis is also known as the point of Departure from Nucleate Boiling, or DNB.

Since the ability of the film to transfer heat has markedly decreased, then using the expression

$$\dot{Q} = h_F \ A \ \Delta T$$

it can be seen that ΔT must increase drastically to maintain the same $\dot{Q}$ once CHF is reached. It must be realized, however, that behavior at the boiling crisis is dependent on flow conditions, and that there are two entirely different mechanisms which can occur once CHF is reached. Keep in mind that both of these mechanisms represent the transition from the nucleate boiling regime to the film boiling regime since we are at the critical heat flux.

Boiling Crisis Mechanism 1 - DNB (Figure 4)

In the case of subcooled flow conditions initially, also known as bubbly flow, the bubbly boundary layer flows parallel to the cladding surface, with a subcooled liquid core flowing at the center of the coolant channel. This is the flow pattern that occurs in the PWR core under normal operating conditions. In this type of flow the boiling crisis appears to be associated with the cloud of bubbles, adjacent to the surface, which reduces the amount of incoming water. When this crisis occurs, the surface temperature rapidly rises to a high value. A local void fraction peak occurs near the wall. When the boundary layer separates from the wall, a stagnant fluid forms under the layer. Due to the high heat flux at the surface, this stagnant fluid evaporates, resulting in a vapor blanket on the heated wall. Saturated nucleate boiling (bulk boiling) will prevail in the center of the channel under these conditions. The boiling heat transfer rate is reduced suddenly as the flow stagnation occurs, hence, this type of boiling crisis is called Departure from Nucleate Boiling (DNB). Recall that this is the term we previously assigned to any situation where CHF is reached, however, this particular mechanism is also referred to as DNB. Figure 4 depicts this situation, and from the figure, four distinct regions can be identified if the heat flux were to be increased in the direction of flow.

Region I - The wall temperature is at or slightly below saturation temperature for the fluid and single phase heat transfer takes place.

Region II - The heat flux is great enough to cause the wall temperature to be greater than saturation temperature and subcooled nucleate boiling will begin. As the heat flux is increased, the amount of nucleate boiling increases. This causes the film coefficient to increase as more and more turbulence exists in the boundary layer. Prior to reaching Region III, the film coefficient will reach some maximum value and level off. Even with increased nucleate boiling, the boundary layer has been disrupted to the maximum extent such that the film coefficient can no longer increase.

Region III - The heat flux is high enough to cause large bubbles to form and adhere to the wall surface for a longer period of time. At this point CHF is reached and the film coefficient decreases. This region is referred to as partial film boiling, or transition boiling.

Region IV - At this point the heat flux is so high that a stable vapor film covers the channel wall. The major mode of heat transfer when this occurs is by radiation heat transfer. Two phase annular flow can exist with a subcooled liquid in the center of the channel and a vapor film along the channel wall.

If CHF was reached and the heat flux held constant, it would take approximately 2-3 seconds to reach Region IV. This would result in a temperature difference of approximately 10,000°F between the wall and the bulk liquid.

Boiling Crisis Mechanism 2 - Dryout (Figure 5)

In this case, we shall consider the initial condition of annular flow, where a liquid annulus flows parallel to the wall with a vapor core flowing at the center of the channel. Bulk boiling is already predominate in the channel since the bulk fluid is at saturation temperature. This is the flow pattern that occurs in a BWR core under normal operating conditions. Since the flow is annular, i.e., the center of the channel is relatively high quality, the boiling crisis occurs at a lower heat flux than in the previous case. A local void fraction peak occurs in the center of the channel. Once the annular liquid film adjacent to the wall becomes sufficiently thin, a dry patch may form on the heated surface, which results in the rise of the wall temperature. Since the velocity of the vapor in the channel center is high, the heat transfer rate is much better than in the preceding low-quality case, and the resulting wall temperature rises are lower and less rapid. Downstream, the entire channel will be completely voided if the heat flux is sufficient, and thus this type of boiling crisis is commonly called dryout. This is the most likely mechanism to occur in the various heat exchangers in the plant, particularly in the case of the steam generators. Referring to Figure 5, three distinct regions can be identified if the fluid bulk temperature is initially at saturation.

Region I - Single phase heat transfer will take place until the wall temperature is greater than saturation temperature.

Region II - As with DNB, nucleate boiling takes place since the heat flux is great enough to cause the wall temperature to exceed saturation temperature. When the bubbles leave the wall and enter the coolant channel they do not collapse since the bulk coolant is also at saturation temperature. As more and more nucleate boiling takes place, the bubbles join together in the channel to form large bubbles.

Region III - When the heat flux is great enough and sufficient bubbles have accumulated in the coolant channel, the entire channel and wall surface will become liquid deficient. At this point, CHF has been reached and dryout has occurred. The film coefficient decreases immediately to a very low value, and once again the major mode of heat transfer is by radiation.

Strictly speaking, each of these mechanisms is a Departure from Nucleate Boiling, since DNB is defined as the point on the heat flux (Q") versus ΔT_{film} curve which corresponds to CHF. Although the boiling proceeds differently in each of the two cases, the end result of both DNB and dryout is the same (i.e., extremely high wall temperatures). Thus, it is common (and acceptable) throughout the industry to refer to both of these mechanisms as simply DNB Nucleate boiling has ceased and the predominant mode of heat removal is via radiation heat transfer.

<u>Effect of Channel Enthalpy</u> (Figure 6)

Figure 6 is a plot of critical heat flux versus channel enthalpy. Although the entire solid line represents a departure from the nucleate boiling regime, the two mechanisms discussed earlier are labelled on the curve to illustrate the effect of channel enthalpy and heat flux on the nature of the boiling that is taking place in a given situation. As the channel approaches saturation temperature, the enthalpy increases. As channel enthalpy increases, the boiling crisis will occur at a lower CHF. For example, in the case of dryout, the bulk fluid temperature is at saturation initially, therefore the channel enthalpy is relatively high as compared to subcooled flow. This being the case, a lower CHF is required for this type of boiling crisis to occur. What would actually happen inside the core under accident conditions would be a combination of both mechanisms occurring simultaneously, the extent of each mechanism being dependent on the nature of the accident.

REACTOR CORE SAFETY ANALYSIS

Various attempts have been made to develop wholly or partly analytical approaches to the prediction of the boiling crisis. The greatest success has been obtained with the boiling crisis at high-quality annular flow. However, none of the pure analytical models to date have lead to correlations which are entirely satisfactory for design use. Reliance must still be placed on empirical correlations of experimental data.

Departure from Nucleate Boiling Ratio (DNBR)

In the interest of maintaining the integrity of the fuel, heat fluxes must be limited to some value below the critical value which corresponds to DNB conditions. For this purpose, a simple ratio is utilized. The ratio of the critical heat flux to the actual heat flux at any point is called the critical heat flux ratio or DNB ratio (DNBR). It is one of the most important factors in reactor design, and operating licenses generally specify a minimum value which must be maintained.

Obviously, if the actual heat flux is less than CHF at some point in the core, DNB will not occur there. To establish an acceptable confidence level, DNBR is limited to a minimum value of 1.30 during steady state operation, normal operational transients, and anticipated transients. This value of DNBR corresponds with an actual heat flux that is 77% of the critical heat flux. A DNBR of 1.30 provides a 95% probability at a 95% confidence level that DNB will not occur and is thus chosen as an appropriate margin to DNB for all operating conditions. During normal operations, DNBR will actually be some value greater than 1.7.

Status of DNB Technology

Early experimental studies of DNB were conducted with fluid flowing inside single, heated tubes or channels and with single annulus configurations with one or both walls heated. The results of the experiments were analyzed using many different physical models for describing the DNB phenomenon, but all resultant correlations are highly empirical in nature. These correlations include the W-3 correlation, which is in wide use in the pressurized water reactor industry.

As testing methods progressed to the use of rod bundles instead of single channels, it became apparent that the bundle average flow conditions cannot be used in DNB correlations. Test results showed that correlations based on average conditions were not accurate predictors of DNB heat flux. This indicates that a knowledge of the local subchannel conditions within the bundle is necessary.

To determine the local subchannel conditions, a special computer code was developed. In this code, a rod bundle is considered to be an array of subchannels, each of which includes the flow area formed by four adjacent rods. The subchannels are also divided into axial steps such that each may be treated as a control volume. By solving simultaneously the mass, energy, and momentum equations, the local fluid conditions in each control volume are calculated. The W-3 correlation, developed from single channel data, can be applied to rod bundles by using the subchannel local fluid conditions calculated by the computer code. This approach yielded conservative predictions, particularly in rod bundles with mixing vane grid spacers. Hence, a correction factor was developed to adapt the W-3 correlation to rod bundles with spacer grids. This correction factor was developed as a multiplier on the W-3 correlation from rod bundle DNB test results conducted in the Westinghouse high-pressure water loop at Columbia University. These tests were conducted on nonuniform axial heat flux test sections to determine the DNB performance of a low parasitic, top-split mixing vane grid design. The testing was done over a wide range of simulated reactor conditions applicable to present and future PWR reactor designs. These conditions were:

Axial Grid Spacing	-	20, 26, and 32 inches
Local DNB Quality	-	-15 to +15 percent
Local Mass Velocity	-	1.6×10^6 to 3.7×10^6 lbm/hr · ft^2
Local Inlet Temperature	-	440°F to 620°F
Pressure	-	1490 to 2440 psia
Local Heat Flux	-	3×10^5 to 1.1×10^6 Btu/hr · ft^2
Axial Heat Flux Distribution	-	Nonuniform
Heated Length	-	8 and 14 feet
Heated Rod OD	-	0.422 inches

These rod bundle DNB data have been analyzed and the previously mentioned correction factor has been developed to conservatively incorporate the mixing vane grid benefit for both typical fuel and thimble cells. The predicted heat flux at some point incorporates the modified spacer factor for typical cells (four fuel rods) and for thimble cells (three fuel rods and a guide tube). The expressions developed are as follows:

$$Q''_{PRED} = Q''_{DNB,N(W-3)} \times F'_{s}$$

where $Q''_{DNB,N(W-3)}$ is the predicted nonuniform DNB heat flux using the W-3 correlation, and

$$Q''_{PRED} = Q''_{DNB,N,\ CW\ (W-3)} \times F'_{S}$$

where $Q''_{DNB,N,\ CW\ (W-3)}$ is the predicted nonuniform DNB heat flux with the flow cell having a cold (unheated) wall evaluated with the W-3 cold wall correlation. F'_S is the correction factor discussed earlier, and is the same in both equations for both typical fuel cells and thimble cells.

As stated previously, the thermal design of a reactor core is limited by the heat transfer mechanism, i.e., DNB heat flux, therefore the core design criterion is stated in terms of a DNB ratio (DNBR), which is defined as:

$$DNBR = \frac{Q''_{DNB,N} \times F'_S}{Q''_{local}}$$

$$Q''_{DNB,N} = \frac{Q''_{DNB,EU}}{F}$$

where $Q''_{DNB,EU}$ is the uniform heat flux as predicted by the W-3 DNB correlation, F is the flux shape factor to account for nonuniform axial heat flux distributions, F'_S is the correction factor, and Q''_{local} is the local heat flux.

The student should realize that an indepth presentation of the W-3 correlation is beyond the scope of this lesson, however, information on this subject is readily available in commercial literature. It is important to note that Westinghouse has developed a more sophistocated correlation than the W-3, with slight improvements in several key areas. However, the W-3 correlation was the one utilized in the Beaver Valley design effort.

Thermal-Hydraulic Design

The performance of a nuclear reactor core can be limited by thermal and hydraulic considerations. The limitations imposed by the departure from nucleate boiling (DNB) require that the thermal analysis be capable of accurately predicting the local flow and enthalpy in the high power channels of the reactor core. DNBR is used as one measure of the margin of safety existing in the core. The objective of reactor core thermal design then is to: 1) determine the maximum heat removal capability in the hottest subchannel, where the pertinent engineering and nuclear effects are compounded into a single flow channel having the highest integrated power output; and 2) design the core to operate at a safe power level below this maximum. A subchannel is referred to as either a single flow channel formed by four adjacent fuel rods, or by three fuel rods and a guide tube thimble.

Early attempts to analyze this problem provided a conservative estimate of the hot subchannel behavior since these methods employed a simple product of the effects of coolant mixing, local variations in dimensions, power generation, and flow redistribution. The product method of combining the flow and power effects on the hot subchannel is unduly conservative since these effects are considered to act independently. Therefore, the resulting engineering uncertainty was an accumulative effect on the hot subchannel enthalpy rise. This problem was alleviated in part by statistically combining those engineering uncertainties relating to fuel and fuel rod fabrications. Nevertheless, some repetition of the engineering uncertainties remained in these early attempts to describe the hot subchannel performance.

A realistic method which can correctly account for the various hydraulic and nuclear effects on the hot subchannel enthalpy rise is one in which the entire core is analyzed as a three-dimensional array. The behavior of the hot assembly is then determined from the effects of the core radial and axial power distributions on the inlet flow profile, with allowances for flow mixing and flow redistribution between assemblies. The average assembly velocity and enthalpy as well as the crossflow and energy exchange at the assembly boundaries are obtained as a function of elevation for the hottest assembly from a core-wide, assembly-by-assembly analysis. Local variations in rod-to-rod power, fuel rod and pellet fabrication, and the coolant mixing characteristics within the hottest assembly are then used to determine the conditions in the hot subchannel.

Special computer programs were developed and utilized to carry out these calculations. Calculational procedures were then compared with experimental data to demonstrate the suitability of the computer programs for use in PWR core design calculations.

Design Basis in Consideration of DNB

To satisfy design criteria, certain design bases have been established for the thermal and hydraulic design of the reactor core. The design criteria that have been established as regards DNB are: departure from nucleate boiling will not occur on at least 95 percent of the limiting fuel rods during normal operation or operational transients, and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level. Historically, this has been conservatively met by limiting the minimum departure from nucleate boiling ratio (DNBR) to 1.30. For this application, a minimum DNBR of 1.30 will continue to be used.

The above design criteria, in general, provide a safeguard as discussed below. By preventing departure from nucleate boiling, adequate heat transfer is assured between the fuel cladding and the reactor coolant, thus preventing cladding damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis since it will be within a few degrees of the coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients associated with Condition II events, including overpower transients. There is an additional large DNBR margin at rated power operation and during normal operating transients.

Design Basis in Consideration of Fuel Temperature

The design criteria established regarding fuel temperature are: during modes of operation associated with Condition I and Condition II events, the maximum fuel temperature shall be less than the melting temperature of UO_2. The UO_2 melting temperature for at least 95 percent of the peak kw/ft fuel rods will not be exceeded at the 95 percent confidence level. By precluding UO_2

melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations.

The fuel temperature design bases contribute to overall design safeguards as follows: fuel rod thermal evaluations are performed at rated power, maximum overpower, and during transients at various burnup. These analyses assure that this design basis as well as the fuel integrity design bases are met. They also provide input for the evaluation of Condition III and IV faults.

Design Basis in Consideration of Core Flow

The design basis for core flow is as follows: a minimum of 95.5 percent of the thermal flow rate will pass through the fuel rod region of the core and be effective for fuel rod cooling. Coolant flow through the thimble tubes as well as the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A maximum of 4.5 percent of this value is allotted as bypass flow; this includes RCC guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

Design Basis in Consideration of Hydrodynamic Stability

The criterion for hydrodynamic stability design is that modes of operation associated with Condition I and II events shall not lead to hydrodynamic instability.

Reactor Core Thermal and Hydraulic Safety Limit Curves (Figure 7, 8, 9)

Figure 7 presents a family of curves of average temperature versus core power for various pressures. The curves depict three-loop operation and indicate that the plant must be operated in the temperature region below the particular curve for a given pressure. Each "curve" in Figure 7 consists of three straight-line segments. These segments denote three operational parameters that place temperature restrictions on core power operation.

At low core power, section (A) of the curve (shown in the left-hand portion) limits the average coolant temperature at the reactor vessel exit (hot leg nozzle) to below the saturation temperature of the water at that pressure. If the RCS liquid in the hot leg is at the saturation temperature and additional power is produced by the reactor, then the RCS liquid would be transformed into steam with no change in RCS temperature. This is referred to as latent heat of vaporization. This situation creates a problem because the RTD manifold which samples the hot leg water would not yield an accurate measure of reactor core power. RTD manifolds measure both hot leg and cold leg temperature and, in turn, the ΔT between the two is used to give a measure of the reactor core thermal power. ΔT is also used for reactor protection, i.e. OTΔT and OPΔT (recall, of course, that $\Delta T = T_H - T_C$). There is no change in ΔT due to latent heat of vaporization, therefore, reduced reactor protection is a result.

If the RCS water enters an RTD manifold at saturation temperature, it no longer yields an accurate measure of reactor core thermal power. This is due to the fact that if the RCS water is at saturation temperature (at the PWR operating pressure), then any additional power produced by the reactor only serves to convert the RCS liquid into vapor and does not change its temperature. When this happens, then the RTD's see no ΔT change. It is very important to avoid this sort of situation where ΔT is no longer proportional to reactor core power since reactor control, protection, and safeguard systems use these RTD signals as inputs.

Section (C) of this graph (shown on the right-hand portion) denotes a DNBR of 1.3 in a thimble cell within the reactor core. The reactor protection system is designed so as to prevent DNB from being reached. The DNB ratio of 1.3 has been chosen as a sufficient margin above DNBR = 1.0 to prevent DNB from occurring. Recall that DNB is a local phenomenon. It can occur in a hot channel where the combination of heat flux and coolant enthalpy rise (increase in coolant Btu/lbm) produces the DNB condition. Section (B) of this curve will be discussed later. It was stated previously that coolant channels, or cells, are of two types in a PWR:

1. Normal fuel cell - consists of four adjacent fuel rods.

2. Thimble cell or "cold wall cell" - consists of three adjacent fuel rods and one instrumentation thimble or RCCA thimble.

Figure 8 illustrates each of these types of flow channels. The thimble or cold wall cell has a smaller cross-sectional area, which tends to restrict the flow in that channel to approximately 75% of that in a standard fuel cell. The heat generated in the thimble cell is approximately 75% of that in a standard fuel cell. At first glance, it appears that both types of channels should be roughly equivalent in a thermodynamic sense, however, it has been established that the thimble cell (or cold wall cell) is usually limiting from a DNB or critical heat flux point of view. The reasoning behind this standpoint is as follows:

1. Consider the coolant flow axially up through the reactor core. As the coolant flows up through the core in a particular channel, the assumption is that the change in coolant pressure (as measured from the bottom of the core up to source height), Z, along this axial flow path is the same for all coolant channels in the core. That is to say, the axial ΔP should be the same for every channel at any height, Z, where the ΔP is measured.

   Therefore, the axial ΔP at any height in a standard fuel cell is also the same as the axial ΔP for a thimble cell at that same height. The thimble cell or channel, however, has a smaller cross-sectional area and, hence, a more restricted flow. Therefore, to have the same axial ΔP as the standard fuel cell, water must flow radially outward from the thimble channel into adjacent channels for this to occur. The net result is that coolant is removed from the thimble cell.

2. If one examines a standard fuel cell in the radial direction, one finds the coolant temperature distribution to have a general shape such as shown in Figure 9A. Average coolant temperature is not very different from the maximum coolant temperature. On the other hand, in the thimble cell, the radial temperature distribution looks like that depicted in Figure 9B. For the same average coolant temperature, the fuel rod clad temperature is clearly higher in the thimble cell.

Therefore, this combination of coolant leaving the thimble channel along with the higher clad temperature results in this channel usually having the limiting CHF or DNB for a given average coolant temperature.

Section (A) of the Figure 7 curve limits coolant temperature so as to keep the RCS hot leg liquid temperature below the saturation temperature. If the RCS liquid temperature reaches the saturation temperature, then the liquid changes to vapor with no change in RCS hot leg temperature. Hence, the RTD's do not yield an accurate measure of the reactor core power. Section (A) of the curve decreases with reactor power increase. This is because the curve is plotted using Tavg, but we are concerned with the core exit temperature, and core exit temperature is at T_H, i.e. the hot leg temperature. Section (B) of the graph, as shown by the middle straight line segment of the "curve", limits the enthalpy rise in the reactor coolant to 15% steam quality (15% steam, by volume exiting the reactor core).

This is a limitation because the input data into the W-3 DNB correlation is based on data where coolant steam quality is no higher than 15%. It may not pose an actual danger to operate with a higher core exit steam quality, but Westinghouse does not have data to prove that higher steam quality at the core exit is actually safe. So, at the present time, Section (B) of the "curve" imposes a restriction on three-loop operation that is somewhat more limiting than the vessel exit saturation temperature limitation given by Section (A) of the "curve".

Note that the production of steam within certain coolant channels is a localized effect. It does not mean that the overall RCS liquid is at saturation temperature. It only means that certain localized areas within the reactor core have reached saturation temperature.

Section (C) of the curve denotes a limitation by the DNBR = 1.3 in the thimble cell once again. Present data shows that even though the thimble cell DNBR is more limiting than the standard fuel cell, the temperature limitations imposed by both the standard fuel cell and thimble cell are very nearly the same for three-loop operation.

Various mechanisms are incorporated into the plant design to ensure that operations do not extend into the region above the curve. First, since the upper pressure curve is a 2400 psia line, the 2385 psig setpoint of the high pressure trip helps prevent operation above that point. Notice also that this pressure is well below the pressure limit of 2735 psig based on the pressure vessel and the pressurizer design. However, the high pressure trip is really designed to limit the range of the OTΔT trip protection and is somewhat of a backup to that trip. The high pressure trip protection is reinforced by the pressurizer safeties which, although not preventing operation outside of the safety limit curve, do protect against exceeding the pressure limit of 2735 psig. Another protection, though not obvious, is the low flow trip, set at 90%. The curve is based on a nominal core flow of 90% so operation should not be allowed with core flow less than that value. Since the curve also is based on an $F^N_{\Delta h}$ of less than 1.55, the low pressure trip, in conjunction with the low flow trip, help prevent operation above section B and C, the curves representing DNB concerns.

Steam generator safety valves provide protection from exceeding the A section of the curve. 1075 psig at 0% power corresponds to a Tave of 555°F and to 615°F at 100% power. If this line were plotted on Figure 7, it would show that operation above the 2250 psia line is prevented. In effect, five lines could be plotted, one above the other, to illustrate all 5 safety settings.

The high flux trip at 109% power guards against operation to the right of the C curve. At low Tave values (i.e., < 577°) it is backed up by the OTΔT trip and at temperatures above 577°F, the OTΔT line (if one were plotted ignoring rate changes) curves inward to the left and becomes the first line of protection, being backed up by the high flux trip. The OPΔT trip, though not credited in accident analysis, plays a part also in protection and again, a simple line could be plotted showing its line of protection.

The OTΔT trip setpoint equation is:

$$\Delta T(S.P.) = \Delta T_o[[K_1 - K_2 (\frac{1 + \tau_1 S}{1 + \tau_2 S}) (T - T')] + [K_3 (P - P')] - f\Delta I]]$$

where:

$\Delta T(S.P.)$ = ΔT setpoint

ΔT_o = Rated ΔT (i.e., 60°F)

K_1 = 1.18, the setpoint with no corrections (i.e., 118% of 60°F)

K_2 = adjustable gain, .01655/°F

$$\frac{1 + \tau_1 S}{1 + \tau_2 S} = \text{lead-lag unit.}$$

It produces an output proportional to the input, in this case the Tave deviation from its rated value, plus the rate of change of the input. Note that this will always have an output unless T - T' is zero regardless of whether or not the difference is changing at some rate. The purpose of this term is to compensate for piping delays involved in temperature measurement meaning that core temperature could be different than that measured.

τ_1 = 30 seconds

τ_2 = 4 seconds

T = Tave

T' = rated Tave, 576.3°F

T - T' = This term lowers the setpoint when Tave exceeds 576.3°F. This is necessary because the coolant's heat capacity is increased at higher temperatures and the margin to DNB (i.e. DNBR) is reduced.

K_3 = Adjustable gain, .000801/psig

P = Pressurizer pressure

P' = rated pressure, 2235 psig

P - P' = This term decreases the setpoint when pressure is below 2235 psig as the lower pressure decreases the margin to DNB.

fΔI = This term which reduces the setpoint to reflect an increase in hot channel factors. The best power distribution is a cosine function and results in equal power production in the upper and lower core regions. Significant ΔI deviations (i.e. > + 11% or - 23%) generate this term to lower the setpoint. For each percent > -23% the ΔT setpoint is reduced by 1.54% and for each % ΔI greater than +11%, the setpoint is lowered by 1.91%.

The OPΔT trip setpoint equation is:

$$\Delta T(S.P.) = \Delta T_o \; [[K_4 - K_5 \; (\frac{\tau_3 S}{1 + \tau_3 S})] - [K_6 \; (T - T")]]$$

where

$\Delta T(S.P.) = \Delta T$ setpoint

ΔT_o = Rated ΔT (i.e., 60°F)

$K_4 = 1.07$, the setpoint at rated conditions.

K_5 = adjustable gain, .02/°F. It is low limited to 0 meaning there are no temperature credits for lower values of Tave.

$\frac{\tau_3 S}{1 + \tau_3 S}$ = Rate - lag unit. It produces an output proportional to the rate of change of the temperature input. In this case, even though there may be a temperature deviation, there will be not output unless the deviation is changing.

τ_3 = 10 seconds

T = Tave. This combined with the rate lag unit produces the rate of change output.

K_6 = Adjustable gain, .00128/°F. This gain is also low limited to 0 such that no temperature credits are given.

T" = Rated Tave (i.e., 576.3°F)

T - T" = This term reduces the setpoint at Tave values above 576.3°F. Because K_6 is low limited, no credits are assessed at Tave values below 576.3°F.

Core Thermal and Hydraulic Safety Limit Involving DNBR (Figure 10, 11)

The Beaver Valley Specifications state: "The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This corresponds to a 95% probability at a 95% confidence level that DNB will not occur..." What does this last statement mean?

As stated throughout this lesson, Westinghouse uses the W-3 DNB correlation, which, in effect, predicts the heat flux required to produce DNB at any point in the core. Figure 10 shows how well the predicted flux agrees with the actual, measured flux at 809 different points in some PWR core. (Actually, a core was not used, but the principle is the same).

Assuming, then, that Figure 10 is understood, lines are drawn on it through the origin of slopes 0.6, 0.7, 0.8, 0.9, 1.0, 1.1, etc. The number of points lying above 0.6 are counted and this fraction of the 809 is plotted on a graph like Figure 11. Similarly, the fraction of points lying above 0.7 is plotted against 0.7. And so on for 0.8, 0.9, 1.0, 1.1, etc. It turns out (as predicted by a theorem from statistics called the Central Limit Theorem) that the points so plotted lie on a straight line.

For another sample of 809 (or whatever) points, a different line would be obtained. To account for this uncertainty, a correction factor is applied to the line. The factor is obtained from a set of tables that enable determination of probabilities based on sample parameters at a given confidence level. Westinghouse designers use a 95% confidence level. Thus, Figure 11 has been corrected to a 95% confidence level, which means that this curve will roughly approximate the experimental curve in 95 out of 100 experiments.

If in all points in the core, the actual (i.e., the measured) DNB heat flux is $>$ 0.77 times the predicted DNB heat flux, then when the actual DNBR is equal to 1.0, we are assured that the predicted:

$$\text{DNBR} = \frac{\text{predicted DNB flux}}{\text{actual local flux}} > \left(\frac{1.0}{0.77}\right) \left(\frac{\text{actual DNB flux}}{\text{actual local flux}}\right)$$

$$= \left(\frac{1.0}{0.77}\right) (1.0)$$

$$= 1.30$$

From Figure 11 it can be seen that the probability of this situation occurring is 95%. Thus, it can be said that if the predicted DNBR is greater than or equal to 1.30, then the probability is 95% (at a 95% confidence level) that the actual DNBR is greater than or equal to 1.0.

APPENDIX

HEAT TRANSFER COEFFICIENTS IN GENERAL

The value of the convection heat transfer coefficient, h_{film}, is governed by many factors, as mentioned earlier. These include various operating factors (geometrical shape of the channel, the flow rate of the coolant, the heat flux, the system temperature, etc.) and the physical properties of the coolant fluid.

In most reactor work, the flow of the fluid is forced and turbulent. The value of h_{film} in forced convection is governed by the thermal conductivity of the fluid as well as by those factors representing turbulence and operating conditions. In the correlations for h_{film}, it is given as part of the Nusselt number, a dimensionless group which includes the thermal conductivity of the fluid and the equivalent diameter of the channel. The Nusselt number is a function of the Reynolds and Prandtl numbers, a fact that can be proved by theoretical analysis, dimensional analysis, or by experiment. Thus,

$$Nu = f(Re, Pr)$$

where:

$$Nu = \text{Nusselt number} = \frac{h_{film} D_e}{k} \text{ (dimensionless);}$$

$$Re = \text{Reynolds number} = \frac{D_e V \rho}{\mu} \text{ (dimensionless);}$$

$$Pr = \text{Prandtl number} = \frac{\upsilon}{} = \frac{c_p \mu}{k} \text{ (dimensionless);}$$

$$\upsilon = \text{kinematic viscosity of fluid} = \frac{\mu}{\rho}$$

= thermal diffusivity of fluid = $\frac{k}{\rho c_p}$

The Nusselt number is numerically equal to the ratio of the temperature gradient at the wall to a reference temperature gradient.

The Reynolds number is a measure of the ratio of inertial to viscous forces. A low Re means that any disturbances arising in the fluid are dampened and that the flow is laminar. At a critical value of Re, such disturbances are no longer dampened and the transition from laminar to turbulent flow takes place. In turbulent flow, the energy transport (heat transfer) mechanism is aided by eddies, which in effect cause lumps of fluid (acting as energy carriers) to mix with the rest of the fluid. A high value of Re therefore means a high degree of turbulence, a high rate of mixing, and a large heat transfer coefficient by convection.

The Prandtl number is a ratio of two molecular transport properties --- the kinematic viscosity, which is a measure of the rate of momentum transfer between molecules and affects the velocity gradient, and the thermal diffusivity, which is a measure of the rate of heat transfer to energy storage by molecules and affects the temperature gradient. Pr therefore relates the temperature gradient to the velocity gradient. Values of Pr much lower than unity mean that near the wall the temperature gradient is less steep than the velocity gradient.

Of the above, only the Prandtl number is made up entirely of physical properties (it is sometimes tabulated as a physical property). It is a function of the temperature of the fluid and, to a lesser extent, of its pressure.

Therefore, for one fluid operating at a given temperature, the Nusselt number is primarily governed by the Reynolds number. However, different coolants operating at the same Reynolds number exhibit heat transfer characteristics which are strongly dependent on the Prandtl number. This dependence governs the type of correlation of h_{film}.

FIGURES

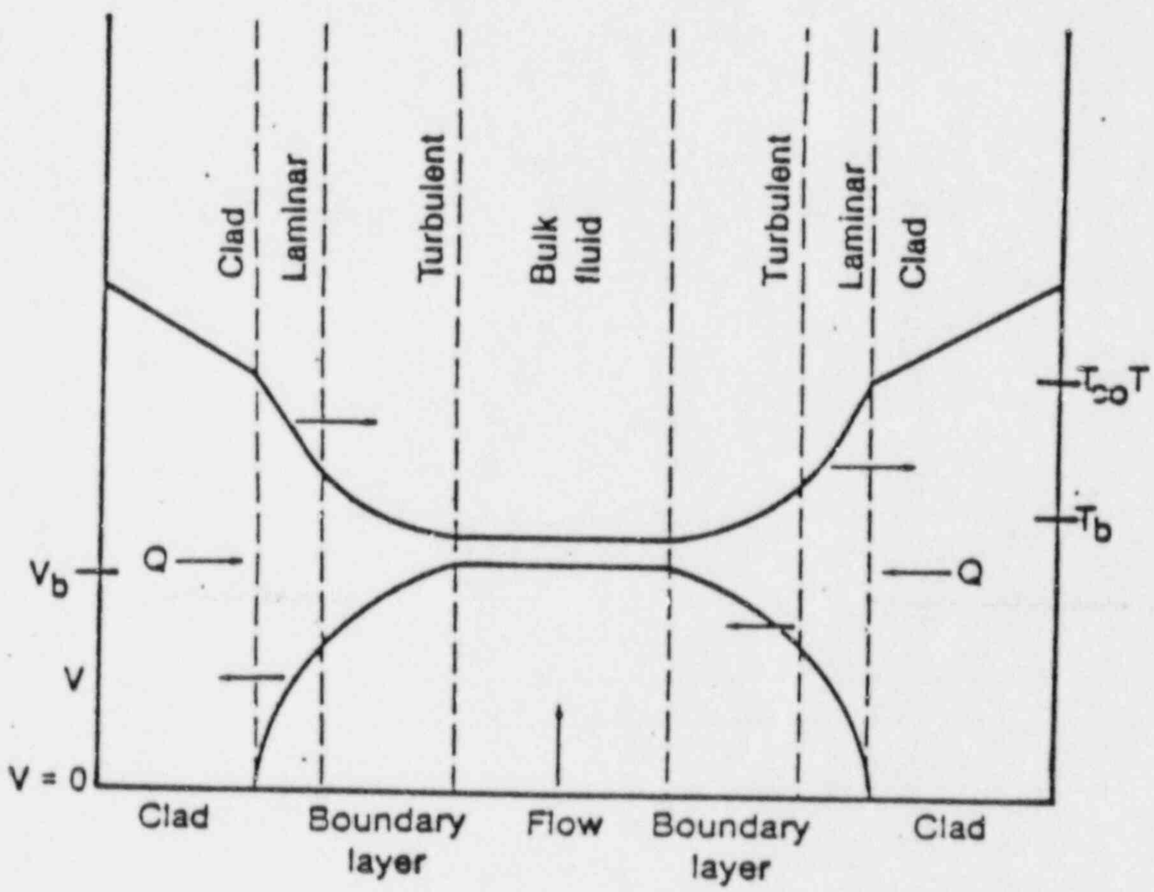


Figure 1. Temperature and Velocity Profiles from Clad Surface to Bulk of Coolant

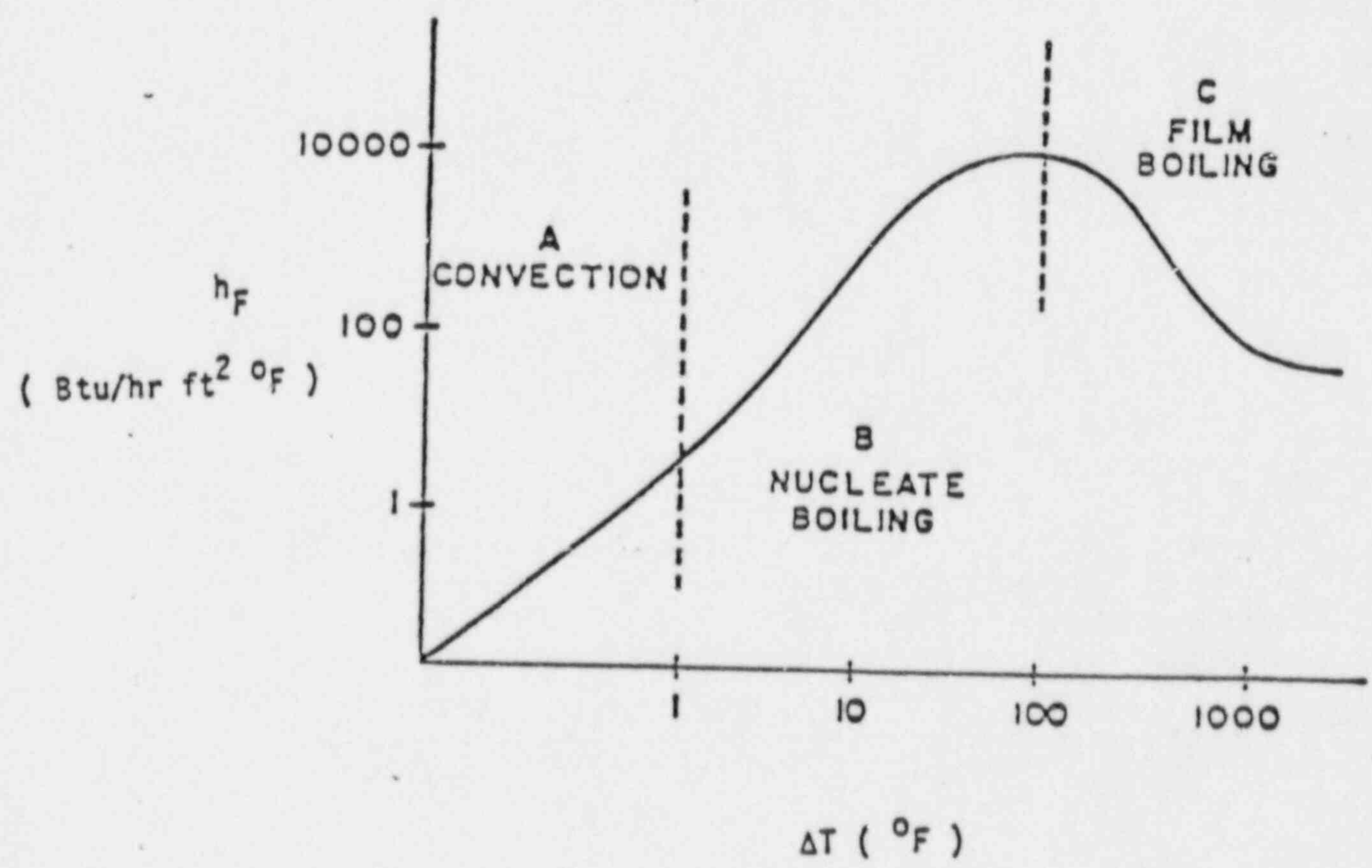


Figure 2. Heat Transfer Coefficient of Water at Saturation Temperature

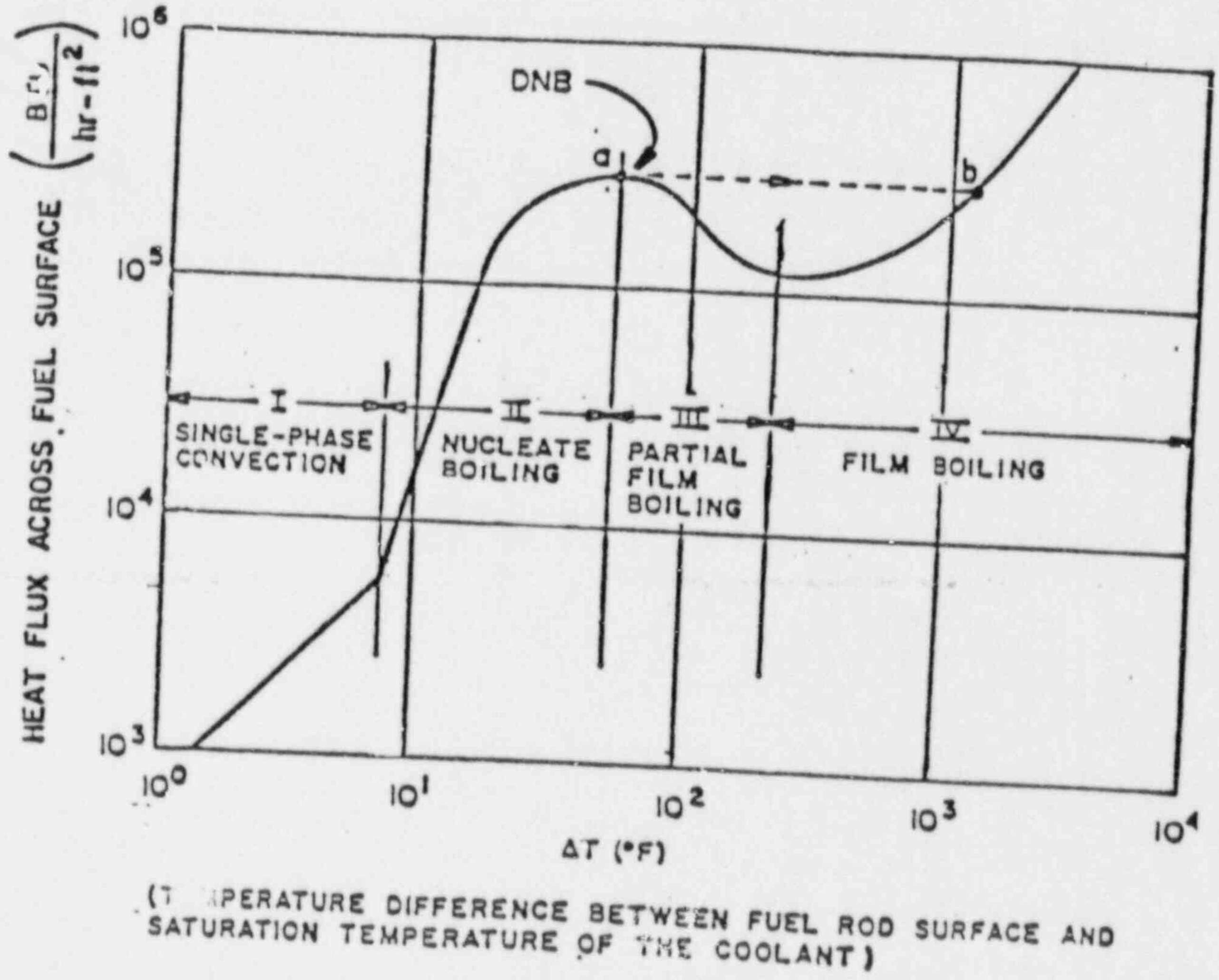


Figure 3. Departure from Nucleate Boiling Curve

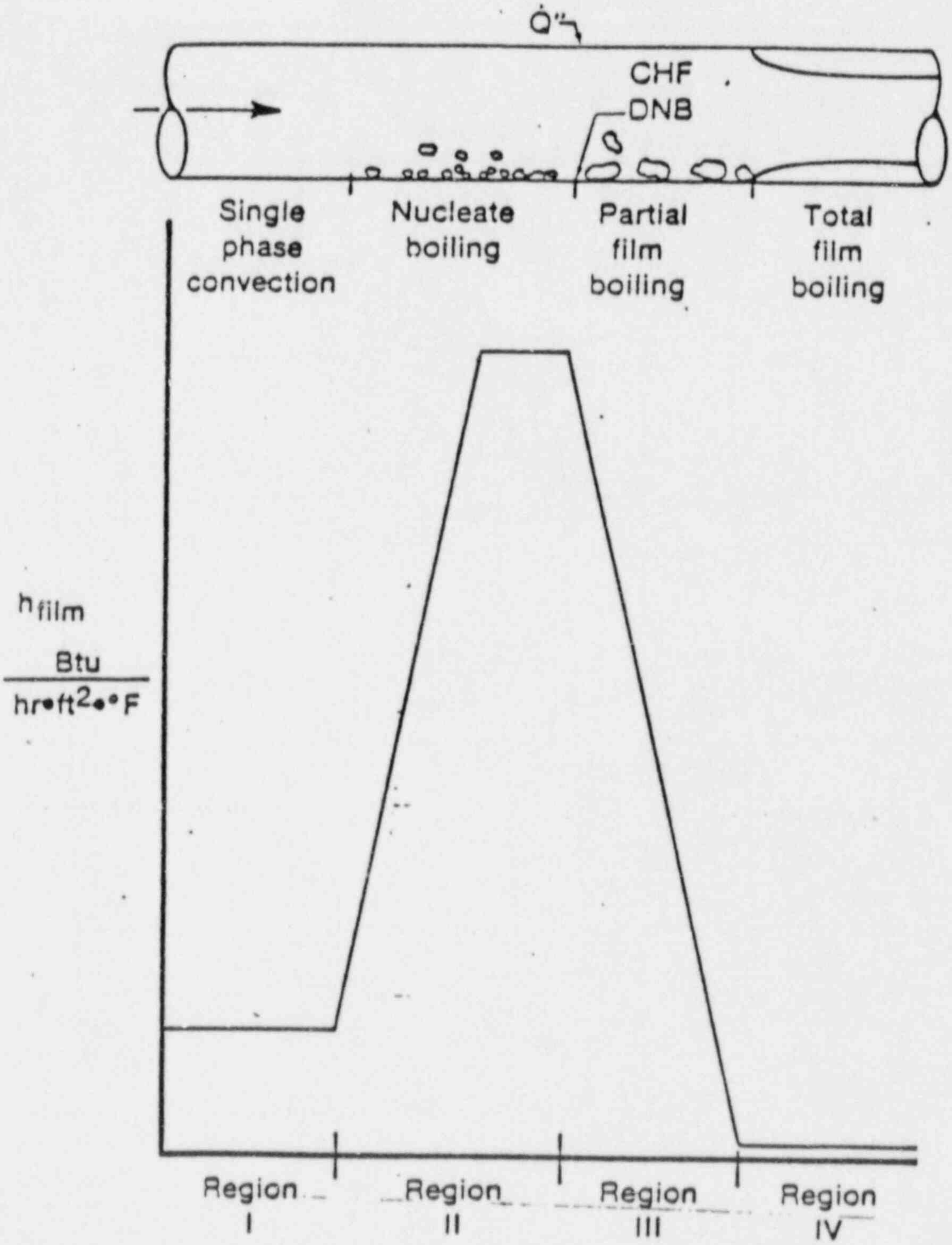


Figure 4. Boiling Crisis Mechanism 1 - DNB

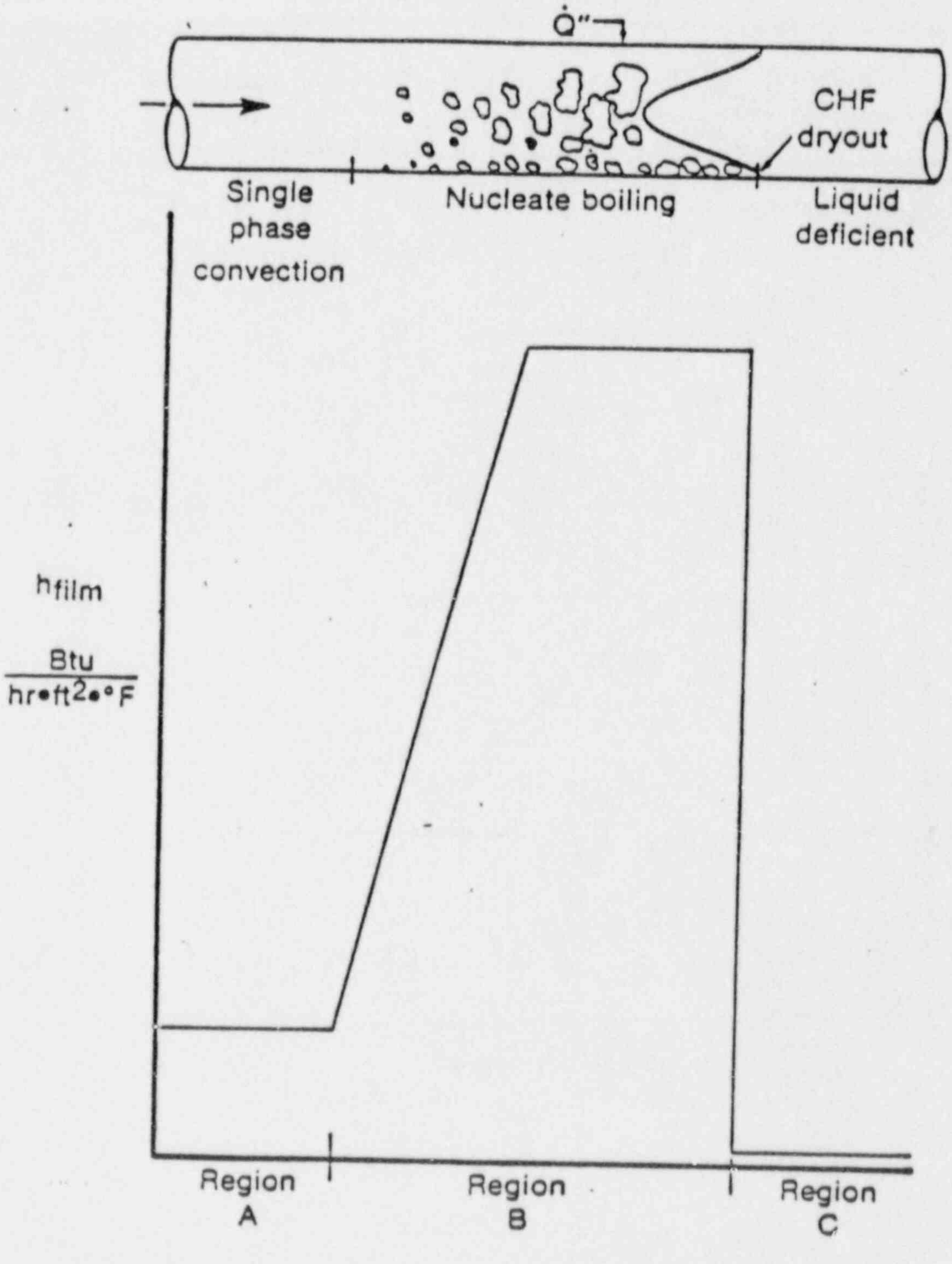


Figure 5. Boiling Crisis Mechanism 2 - Dryout

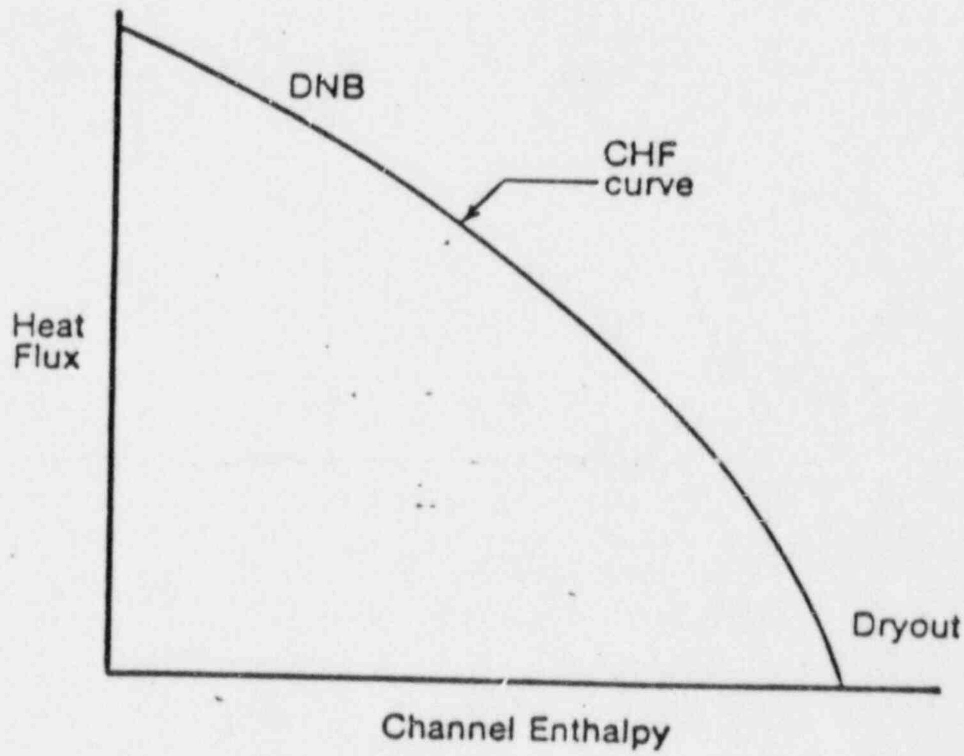


Figure 6. Critical Heat Flux Curve

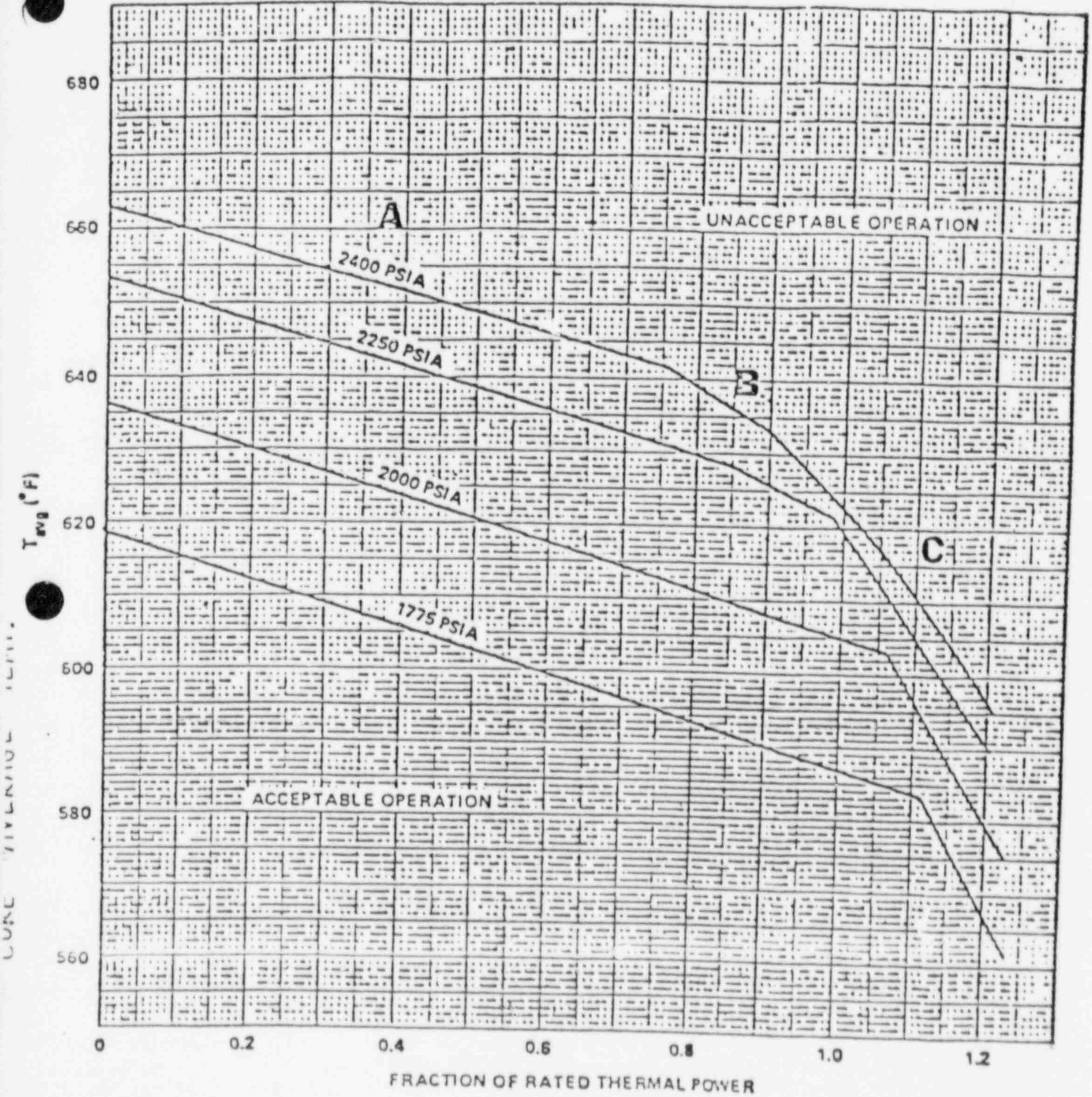


Figure 7. Reactor Core Safety Limit: Three Loops in Operation

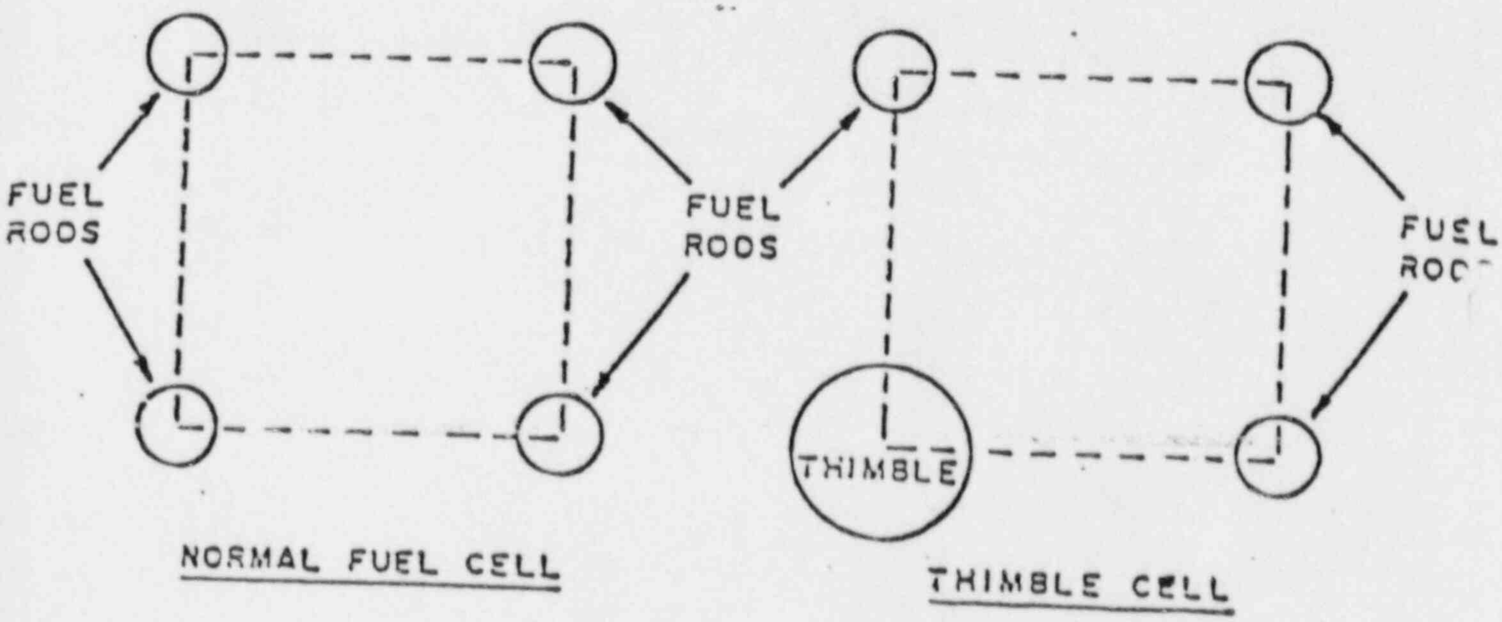


Figure 8. Overhead Cross-Sectional View of Normal Fuel Cell and Thimble Cell

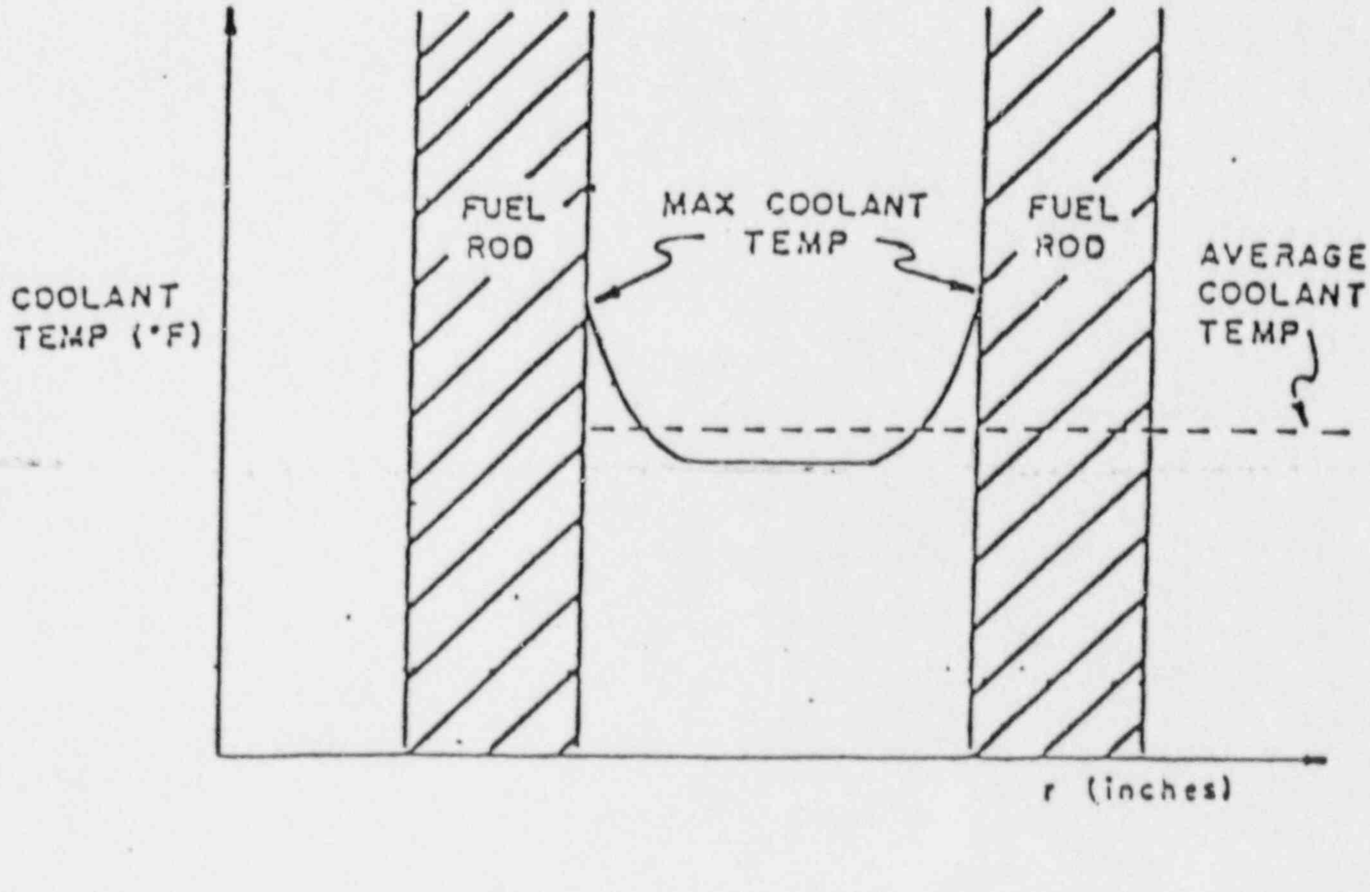


Figure 9A. Temperature Profile for Typical Fuel Cell Channel

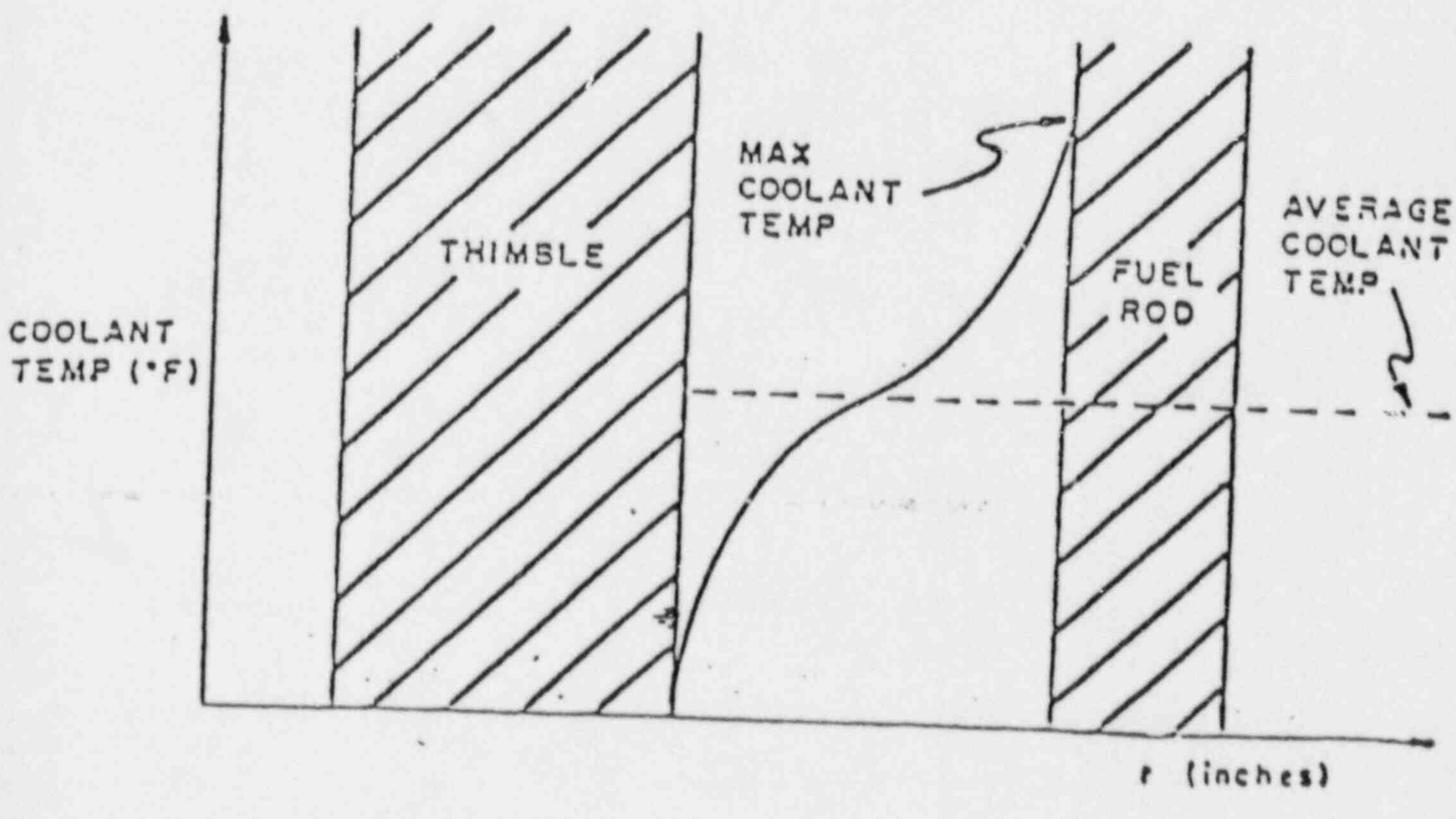


Figure 9B. Temperature Profile for Typical Thimble Cell Channel

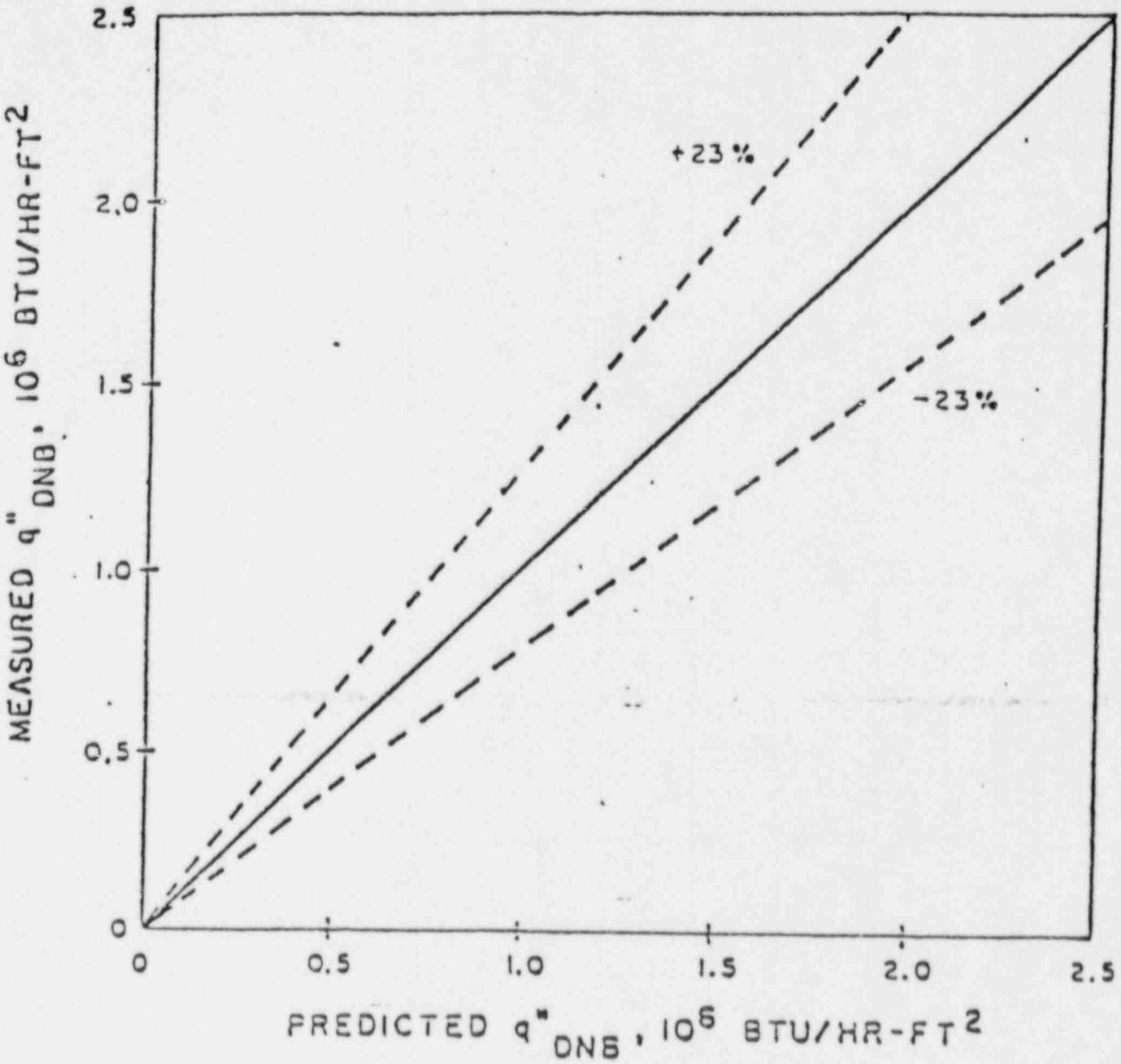


Figure 10. Comparison of W-3 Prediction and Uniform Flux Data

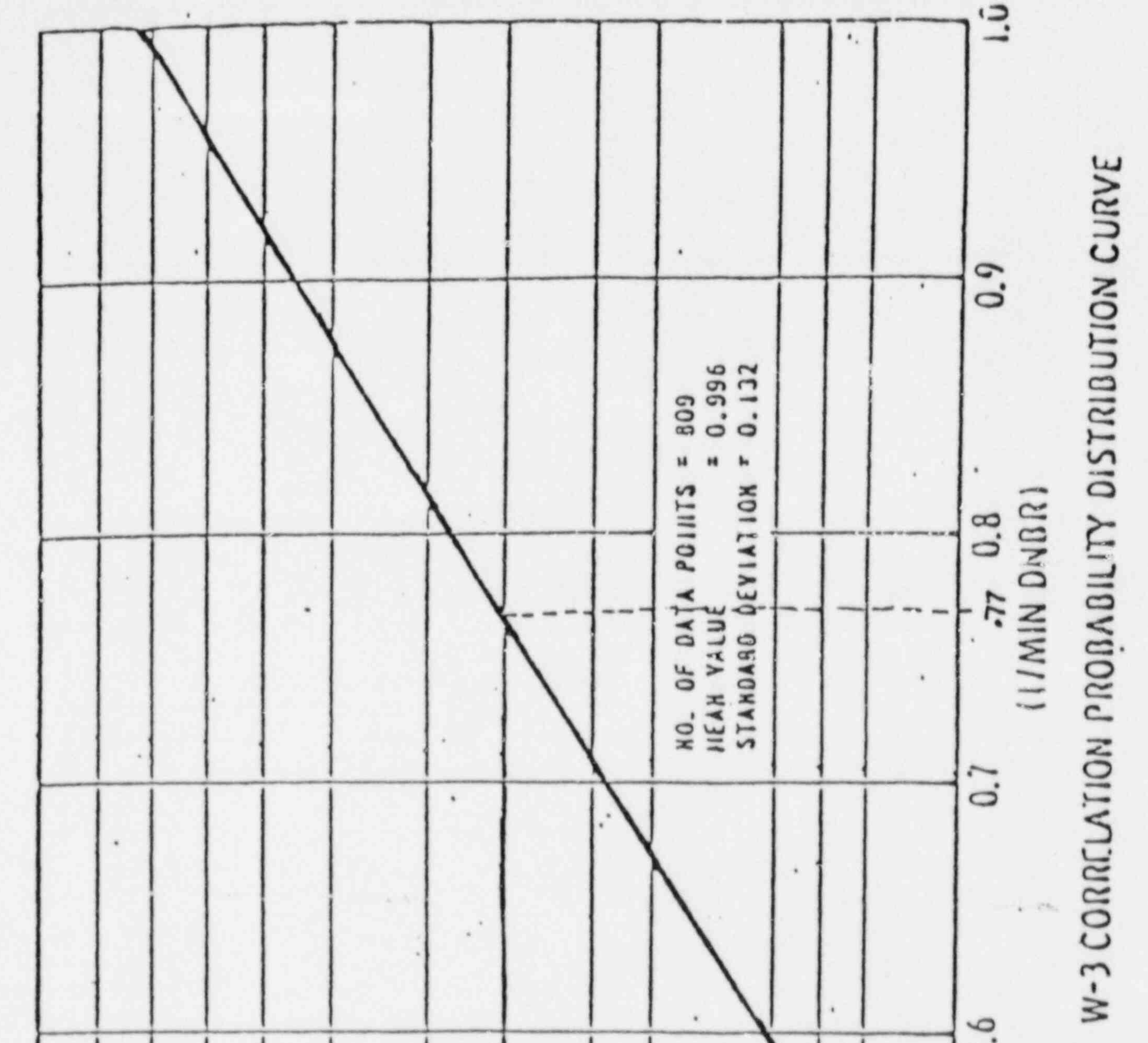


Figure 11. W-3 Correlation Probability Distribution Curve

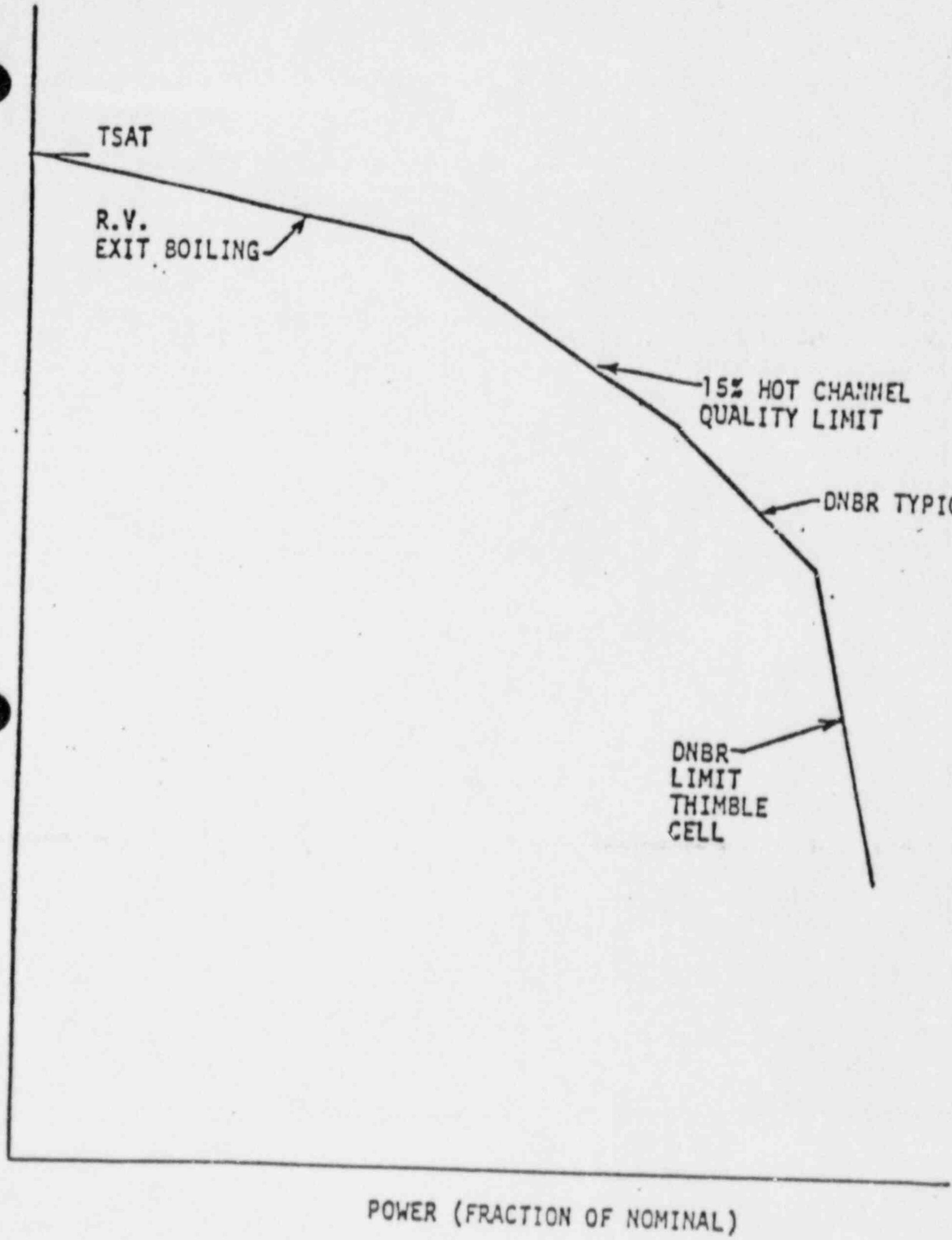


Figure 12. Typical Core Limit

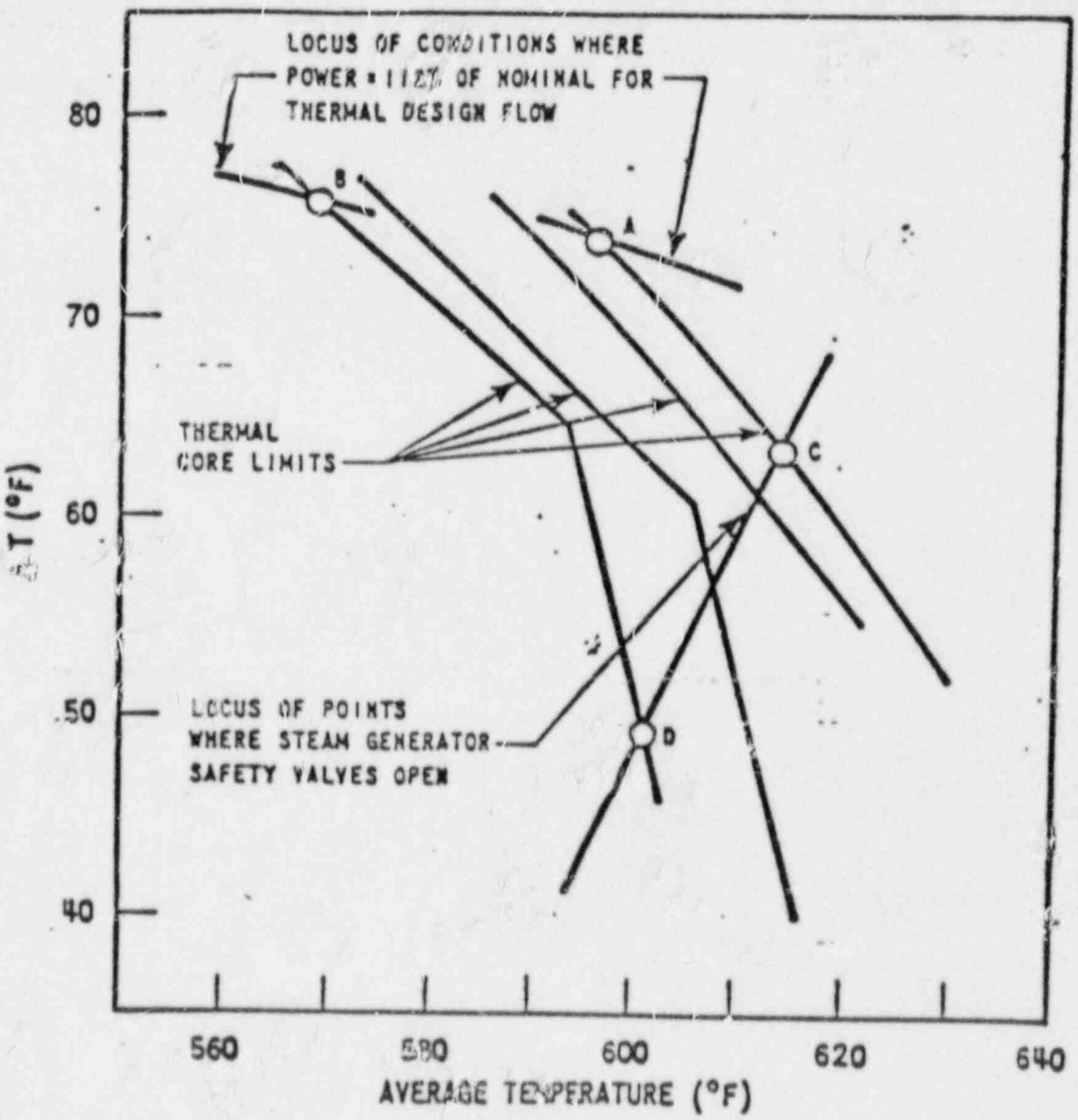


Figure 13. Intersection Points Used to Determine Overtemperature ΔT Protection

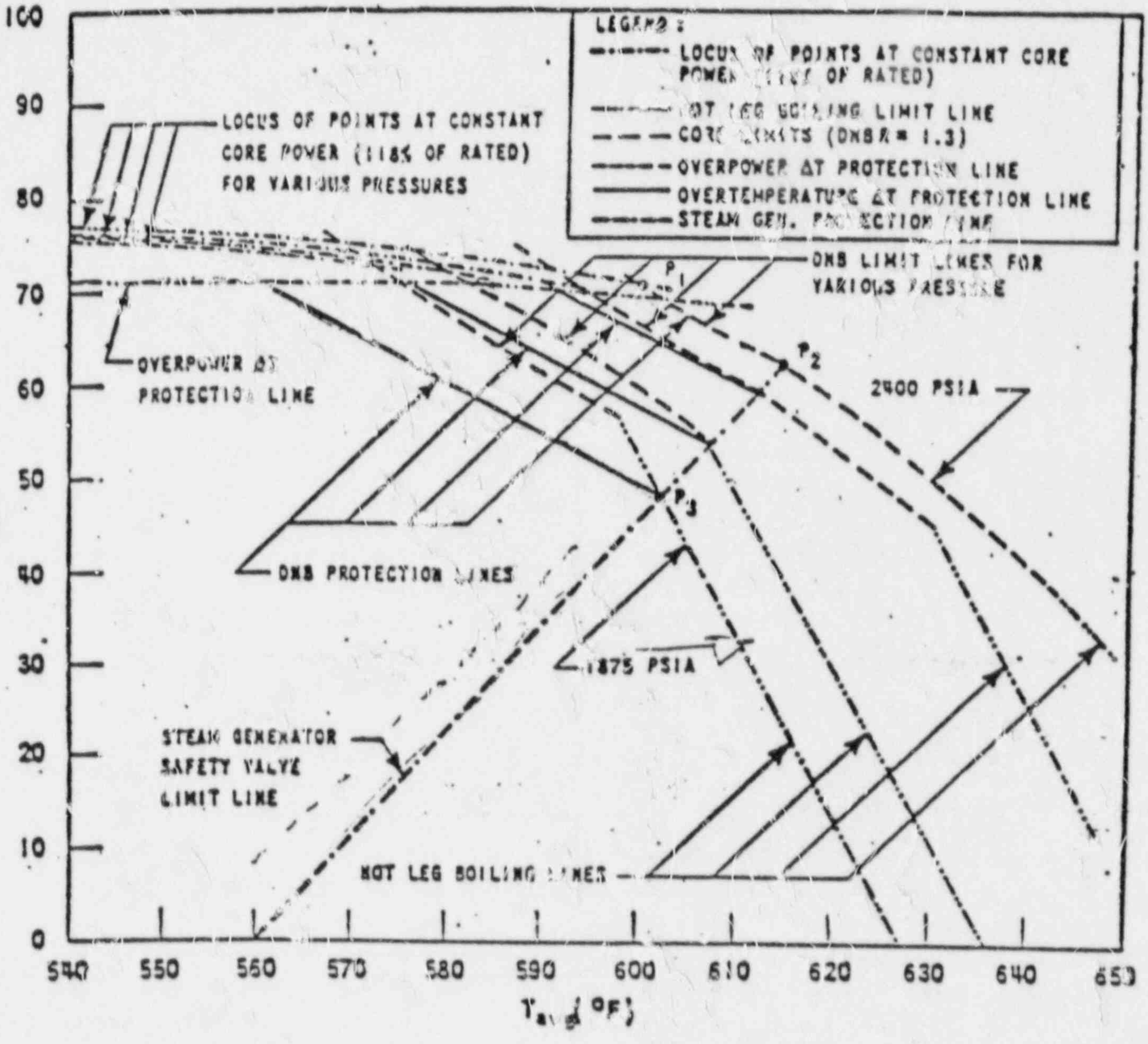


Figure 14. Illustration of Overpower and Overtemperature ΔT Protection

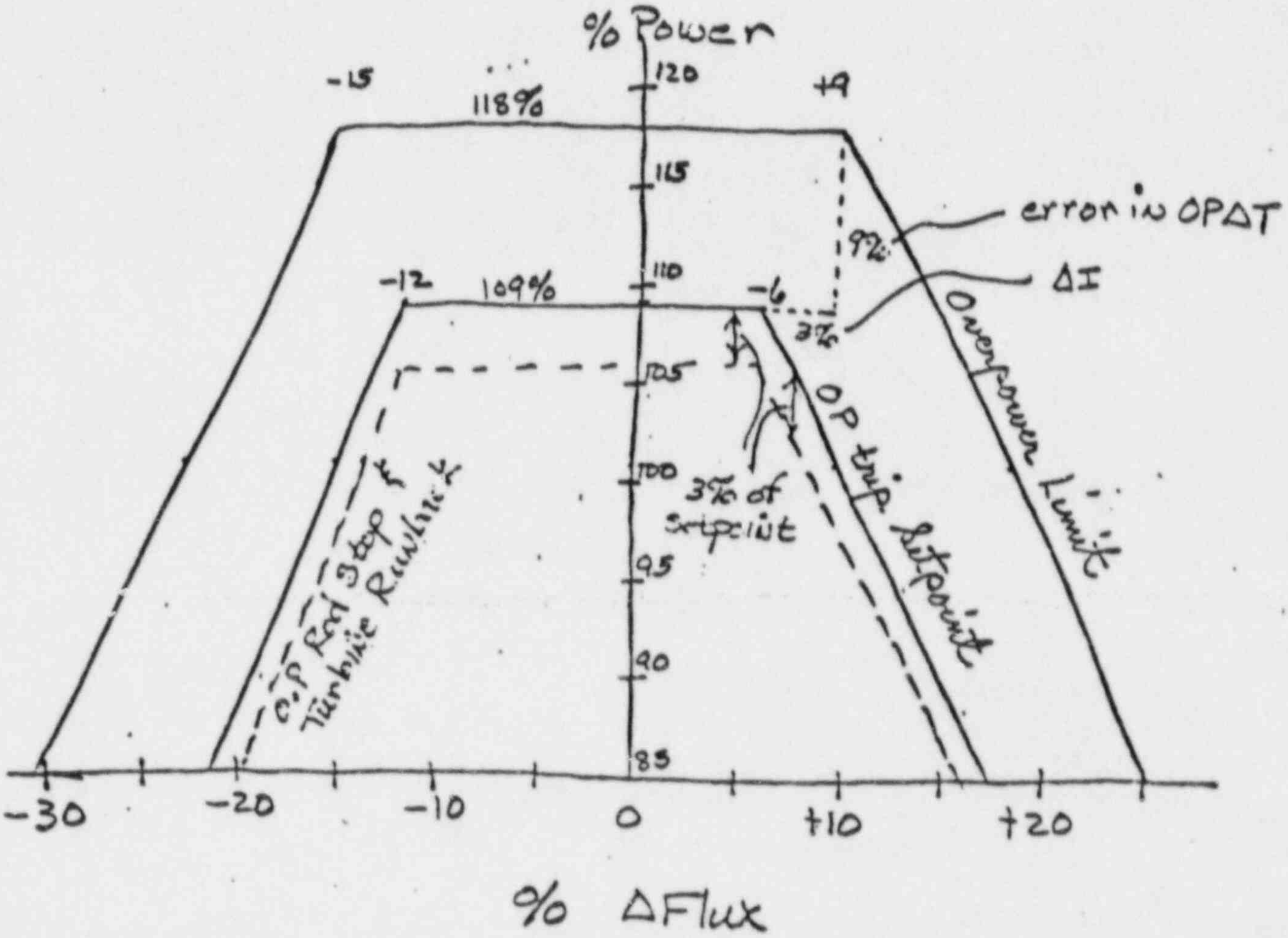


Figure 15. Overpower ΔT Protection -Allowable Power Versus Axial Flux Difference (ΔI)

EXHIBIT K - ATTACHMENT 1

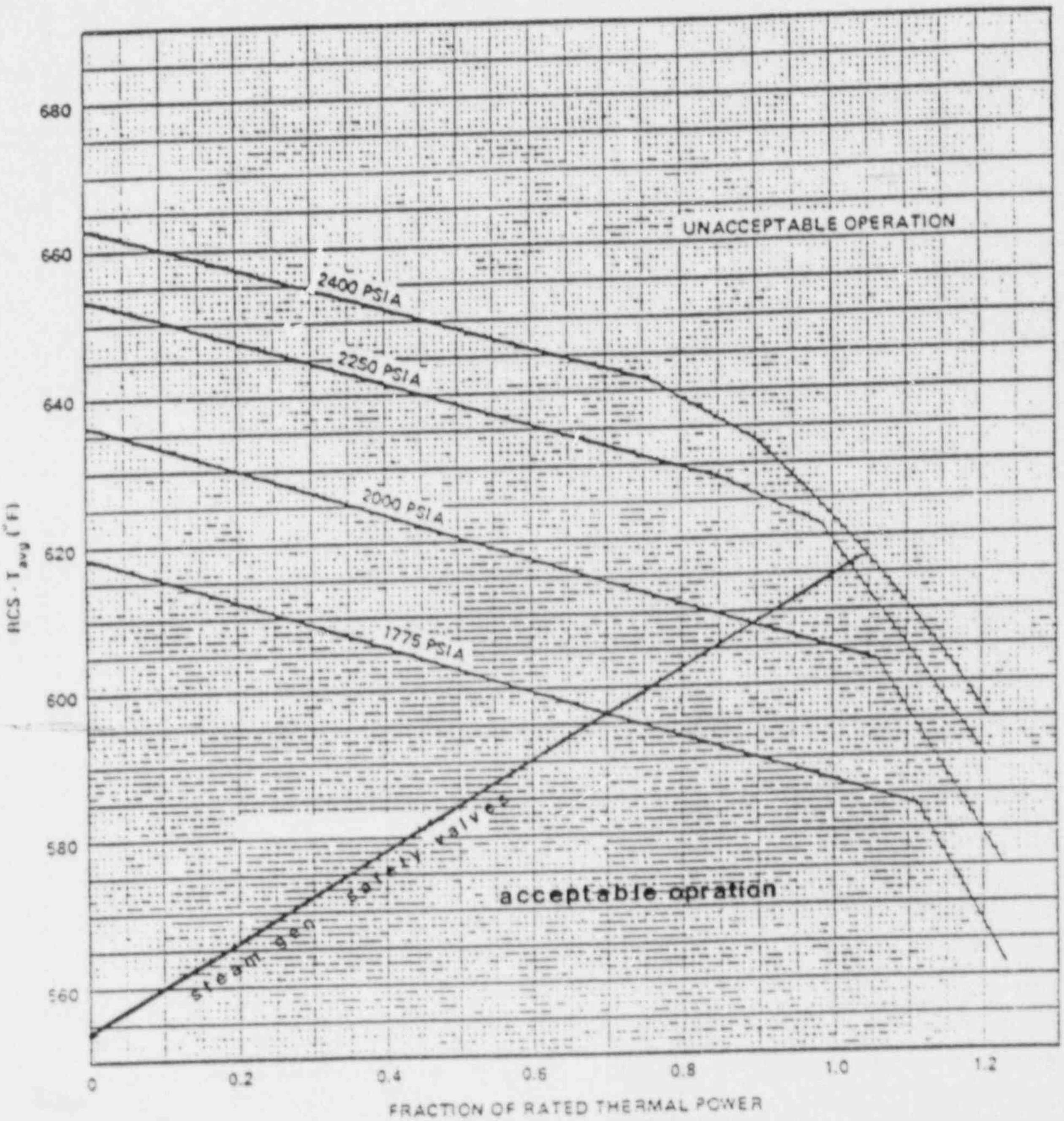


FIGURE 2.1-1 REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

NOV 12 1986

Docket No. 55-60755

Mr. Alfred J. Morabito
685 Tulip Drive
New Brighton, Pennsylvania 15066

Dear Mr. Morabito:

This is in response to your letter of September 11, 1986 requesting a review of the results of your Senior Reactor Operator license written and simulator examinations which were administered at Beaver Valley Unit 1, the week of July 21, 1986. We have reviewed your request for a hearing concerning the denial of your application for a Senior Operator license, under 10 CFR Part 55 of the Commission's regulations. Copies of your examination results were forwarded to you by letter on August 27, 1986.

Based upon our review of the information you provided, a complete regrade of the written examination and a complete reevaluation of the simulator examination was conducted. No adequate basis was found for reversing our original determination. Accordingly, the denial of your application for an operator license remains in effect. Attachments to this letter contain a summary of the regrades, resolution of the comments you submitted, and additional grading changes that resulted from this review.

You may file a new application after two months from the date of your denial, August 27, 1986, in accordance with Section 55.12(a) of 10 CFR Part 55. Under the regulations, each examination and reexamination consists of both written and operating tests. However, the Commission may, at its discretion, under Section 55.12(b), excuse an applicant from reexamination on the part of the examination which he has previously passed, if sufficient justification is presented. A request to be excused may be made at the time of reapplication.

In connection with the above, if you wish to continue to pursue a hearing on the denial, please inform Mr. W. T. Russell, Director, Division of Human Factors Technology, Nuclear Regulatory Commission, Washington, D.C. 20555, within 20 days of the date of this letter. You may submit additional information to support your request at that time. If we do not hear from you within the 20 day period, the matter will be considered closed.

Sincerely,

William F. Kane, Director
Division of Reactor Projects

Enclosures: Attachments 1 and 2

<u>ATTACHMENT 1</u>

<u>REGRADE OF WRITTEN EXAMINATION</u>:

A complete independent regrade of the written examination was conducted. An unmarked copy of the original examination was used for the regrade. A review of the answer key, facility review comments, Mr. Morabito's review comments, and the utility training material was completed prior to beginning the regrade. The results of the regrade are as follows:

	5	6	7	8	Total	
NRC Initial Grade	87.6	59.7	85.3	96.3	82.2	Fail
NRC Regrade	87.6	67.6	78.8	88.6	80.6	Fail

The following are the resolutions to Mr. Morabito's comments.

6.03.b No references were supplied by Mr. Morabito to support his statements. The references provided in the key support the answer in the answer key. Only one of the three design features in the answer key was provided on the written examination. No change.

6.07.b No references were supplied by Mr. Morabito to support his statements. The references provided in the key support the answer in the answer key. FSAR Chapter 14.2 on Page 16 indicates the MSIV closure is required to terminate the blowdown in order to reduce the amount of positive reactivity added by the cooldown of the reactor coolant. Lesson Plan LP-SQS-53A-E-2 on Page 4 indicates the MSIV closure should be verified to assure the rupture is isolated or to identify that the leak cannot be isolated. The facility literature provides many varied reasons for closing the MSIVs and, therefore, there is no definitive answer to the question and the question was deleted.

6.08.c No reference was supplied by Mr. Morabito to support his statements. The reference provided in the key indicates the undervoltage device is on the 480 volt emergency bus, not the 4KV emergency bus. No change.

6.09.b The reference provided in the key supports Mr. Morabito's statements Add 0.8 points.

6.09.c No reference was supplied by Mr. Morabito to support his statements. The reference in the key supports the answer in the answer key. There was no indication in Mr. Morabito's answer that he was aware of the safety concern of maintaining Shutdown Margin. No change.

6.10.a The references provided by Mr. Morabito support his answers. This question was taken directly from the Beaver Valley Examination bank, question number 6-11. Either Mr. Morabito did not retain the information as it was presented in the training program, or the information in the training program is inaccurate. Add 0.4 points.

The following are justifications for additional point deductions.

6.06.b Incorrect setpoint for pressurizer vapor temperature. Minus 0.2 points.

7.02.c The candidate did not specify the "five hottest" thermocouples or two RCS hot leg temperatures. He did not treat RVLIS indication as an independent parameter. Minus 0.6 points.

7.03.c The candidate did not indicate what parameters must be checked to verify that a heat sink is available. Minus 0.2 points.

7.09.b The candidate did not indicate that the requirement for the operability of two RHR loops when water level was less than 23 feet above the vessel flange was based on single failure criteria. Minus 0.3 points.

7.10.b The candidate did not indicate that concerns for securing specific equipment on high RCS activity centered on the conditions in the auxiliary building and not just "release of high activity water from the containment." Minus 0.3 points.

7.10.d The candidate did not include the limit that activity must always be within the bounds of the Technical Specification curve on dose equivalent I-131. Minus 0.2 points.

8.04 The candidate did not correctly identify the modes for the applicable action statements for the RCS pressure safety limit. Minus 0.3 points.

8.05 The candidate did not specify that the safety limits were to maintain the integrity of the fuel and the RCS, he simply stated the limits were to prevent "release of radioactivity." Minus 1.0 points.

8.06.a The candidate did not state that actions for restoring the Technical Specifications staffing levels were required to be conducted immediately. Minus 0.3 points.

8.11.a The candidate did not include reviewing logs as part of shift change. Minus 0.33 points.

ATTACHMENT 2

1. REVIEW OF ORAL EXAMINATION:

Mr. Morabito's statements concerning the evaluation comments on NRC Form 157C are not addressed since the statements do not affect the pass decision made for the oral examination.

2. REVIEW OF SIMULATOR EXAMINATION:

A review of Mr. Morabito's statements concerning the simulator examination and discussions with the examiners involved in the simulator examination were conducted. Mr. Morabito's statements do not challenge the validity of the observations of the examiner concerning the occurrence of incorrect actions. Mr. Morabito's statements attempt to minimize the significance of the incorrect actions by references to good crew interaction and teamwork.

Mr. Morabito was being examined as an individual and his contribution to the overall crew response was evaluated. As a result of the review, no justification was found for overturning Mr. Morabito's failure of the simulator examination.

The following are detailed responses to Mr. Morabito's statements.

ES-301-11, Attachment 1/4, comment 1:

The candidate incorrectly diagnosed the problem. Candidate took actions to "commence reducing power to achieve Mode 3 conditions ..." without reference to a procedure. These facts indicate the candidate's initial diagnosis was incorrect and that he operated the plant in abnormal conditions without reliance on procedures. These facts by themselves do not provide a basis for failure but support the case that the candidate was unable to correctly diagnose instrument and component failures and did not rely on procedural guidance.

ES-202-11, Attachment 1/4, comment 2:

The candidate displayed an incorrect thought process which was corrected by an operator.

ES-302-11, Attachment 1/4, comment 3:

Agree. The candidate did not have sufficient time to consult the alarm response procedure to verify that his corrective actions were proper.

ES-302-11, Attachment 1/4, comment 4:

The comment indicates that the candidate was unable to correctly respond to plant transients without reliance on procedures. This is a significant point when the candidate has previously stated that he did not need to rely on procedures due to his knowledge of procedural requirements. [See candidate's comments ES-305, 5.2.a and ES-302-11, Attachment 1/4, comment 3.]

ES-302-11, Attachment [illegible], comment 1:

The candidate made an incorrect diagnosis and took incorrect action. His reliance on other operators to prevent his incorrect actions is not indicative of a safe operator.

ES-302-11, Attachment 2/4, comment 2:

The examiner incorrectly stated in his report that there were position indication lights available for the residual heat release valves. However, the candidate was unable to determine how to properly complete a verification step in an Emergency Operating Procedure. The candidate relied on another operator to make a decision on how to complete the verification step.

ES-302-11, Attachment 2/4, comment 3:

The candidate incorrectly positioned the wrong switch for the containment sump pump. Another operator provided assistance in locating the proper switch. The candidate's reliance on another operator is not indicative of a safe operator.

ES-301-11, Attachment 2/4, comment 4:

The candidate incorrectly reset CIA during the scenario. The discussion held after the scenario was conducted to make the candidate aware of his mistake. The CIA and CIB annunciator will not clear unless both trains have been reset. The candidate still does not know what plant indications are available for verifying proper plant response to his actions.

ES-302-11, Attachment 3/4, comment 1:

The feedwater regulating valve bypass valve was failed open at 1130. The mismatch of feedwater flows was noted and an I&C technician was called to investigate instrumentation. At 1154 the candidate noticed the failed open bypass valve as part of an Emergency Operating Procedure verification. This is an example of the candidate being unable to properly diagnose a component failure.

ES-302-11, Attachment 3/4, comment 2:

The operating crew did not properly diagnose the cause of unbalanced feedwater flow.

ES-302-11, Attachment 4/4, comment 1:

The candidate incorrectly transitioned to ECA-0.0. The candidate correctly identified the DF bus as deenergized but incorrectly diagnosed the cause. The DF bus being deenergized was a normal consequence of the sequence of events. By his actions, the candidate did not demonstrate a correct understanding of plant response to a loss of offsite power.

ES-302-11, Attachment 4/4, comment 2:

The use of hand signals to indicate the amount an instrument is reading above normal is unsatisfactory. It is uncertain whether the candidate had sufficient information to conclude that the radiation monitor was not reading normally.

ES-302-11, Attachment 4/4, comment 3:

The candidate displayed an incorrect thought process which was corrected by an operator.

The following incorrect actions were identified during the review process.

During the dilution accident, the RO incorrectly initiated normal boration when the low rod insertion alarm limit was reached. Emergency boration, in accordance with Operating Manual Chapter 7, Section 4S, is required by the alarm response procedure. The RO incorrectly secured boration. The RO incorrectly pulled rods with Tave above the programmed level. The candidate, who was the shift supervisor during these events, took no action to correct the RO or to exercise supervisory control of his operator.

As an operator, the candidate expected the shift supervisor to provide detailed supervision of his actions to identify and prevent any incorrect manipulations or diagnosis. As the shift supervisor, the candidate was unable to provide this type of supervision and diagnosis for his operators as demonstrated by his inactions during the dilution accident, the feedwater flow imbalance condition, and the pressurizer level reference failure.

valves and connecting piping have been analyzed for the dynamic forces imposed during steam relief.

10.3.1.2 Description

Steam from each of the three steam generators is conducted in 32 inch OD x 0.9297 inch minimum wall thickness, carbon steel pipe through swing disk-type trip and nonreturn valves, located in an enclosure immediately outside the reactor containment, to a 36 inch OD manifold which is located in the turbine building. Connections for the turbine steam bypass, turbine steam sealing system, reheater supply and auxiliary steam supply are provided at the manifold. A steam flowmeter interconnected with a three-element feedwater control system is provided in the main steam line between each steam generator and its main steam isolation valve. From the 36 inch OD manifold, the steam passes to the turbine throttle/stop valves and governor control valves.

The nonreturn valves automatically prevent reverse flow of steam in the case of accidental pressure reduction in any steam generator or its piping. If a steam line breaks between a nonreturn valve and a steam generator, the affected steam generator continues to blowdown while the nonreturn valve in the line prevents blowdown from the other steam generators. This steam line break accident is discussed in Section 14. The main steam trip valves provide backup for the nonreturn valves to prevent blowdown from intact steam generators through a ruptured pipe between a nonreturn valve and another steam generator. 6.01 b.&

The swing disk-type trip valve in series with each main steam isolation valve contains a free swinging disk normally held up out of the main steam flow path. If a pipe ruptures (Section 14) downstream of the trip valve, a signal derived as indicated in Section 7 causes all three valves to trip closed, thus stopping the flow of steam through the pipe rupture. Maximum closing time for the trip valves is approximately 5 seconds, including instrument time. Valve closure checks the sudden and large release of energy in the form of main steam, thereby preventing rapid cooling of the reactor coolant system. Trip valve closure also ensures a supply of steam to the turbine drive for the turbine-driven steam generator auxiliary feedpump.

Five ASME Code safety valves are located in each main steam line outside the reactor containment and upstream of the nonreturn and trip valves. These safety valves are sized to pass steam flow resulting from a complete load rejection or other shutoff of main steam flow without a direct reactor trip. This is considered the most extreme accident condition.

Excess steam generated by the sensible heat in the nuclear steam supply system (NSSS), immediately following loss of load, is bypassed directly to the turbine condenser (Section 10.3.6) by means of two turbine steam bypass lines, which provide a total bypass capacity of 85 percent of full load steam flow. Each

Steam is supplied from each main steam line upstream of the trip valve to the turbine drive for the turbine-driven steam generator auxiliary feedpump (Section 10.3.5). The piping is arranged so that any steam generator can supply the turbine drive for this pump.

Check valves are provided in the steam supply line from each steam generator to the turbine drive to ensure the availability of driving steam in the event of failure of a steam generator or a line break upstream of a main steam nonreturn valve. Two trip open valves in parallel are located in the inlet of the turbine drive. Steam pressure is available at the inlet of these valves at all times. Indications of all operating conditions are available in the main control room to enable the operator to adjust feedwater flow by throttling valves at the pump discharge. Additional description of steam generator auxiliary feedpump operation is contained in Section 10.3.5.

Steam leaving the high pressure turbine passes through four moisture separator-reheater units in parallel to the inlets of the main low pressure turbine cylinders. Each of the four steam lines between the reheater outlet and low pressure turbine inlet is provided with a reheat stop valve and a reheat intercept valve in series. These valves, operated by the turbine control system, function to prevent turbine overspeed. An ASME Code safety valve is installed on each moisture separator reheater to protect the separators and reheat system from overpressure. The safety valves are designed to pass the flow resulting from closure of the reheat stop and intercept valves with the main steam inlet valves wide open. These valves discharge to the condenser.

10.3.1.3 Performance Analysis

If a main steam line pipe rupture occurs (Section 14.2.5), a 2 out of 3 channel low pressure signal from any main steam line causes the swing trip valves in all three main steam lines to trip closed. Maximum closing time for the trip valves is approximately 5 seconds. If the rupture occurs downstream of the trip valve, valve closure stops the flow of steam through the pipe rupture, thus checking the sudden and large release of energy in the form of steam. This prevents rapid cooling of the Reactor Cooling System (RCS). Trip valve closure also ensures a supply of steam to the turbine drive of the turbine-driven steam generator auxiliary feedpump, as described in Section 10.3.5. 6.07b.1.

If a main steam line breaks between a trip valve and a steam generator, the affected steam generator continues to blowdown. The nonreturn valve in the ruptured line prevents blowdown from the other steam generators. This is the worst steam break accident and is discussed in Section 14.2.5.

ANALYSIS AND SYSTEM MODIFICATION FOR RECIRCULATION SPRAY AND LOW HEAD SAFETY INJECTION PUMPS NET POSITIVE SUCTION HEAD

FINAL REPORT

November 17, 1977

BEAVER VALLEY POWER STATION - UNIT NO. 1
DOCKET NO. 50-334
LICENSE NO. DPR-66

2.2 SYSTEM MODIFICATION

It was determined in August 1977 that the available net positive suction head (NPSH) for the RS and LHSI pumps may be less than previously specified under postulated LOCA conditions.

Interim plant modifications described in the report entitled "Analysis and System Modification for Recirculation Spray and Low Head Safety Injection Pumps Net Positive Suction Head," dated September 9, 1977, were implemented during the recent station outage. These modifications are temporary and will be replaced by the permanent modifications described in this report during the first scheduled refueling outage.

Under certain pipe break conditions, it is possible for the energy distribution between the containment atmosphere and the sump water to be different from that originally assumed in the accident analysis. Also, the containment pressure transient after LOCA can be such that NPSH (available) for the RS and LHSI pumps is further reduced below the original design value specified. This could result in degradation of the pump performance.

The most significant assumption in determining the available NPSH to the various pumps is the distribution of energy released from the reactor coolant system between the containment atmosphere and water in the sump. The condition which results in the potential for lowest available NPSH is a sufficiently large opening in the reactor coolant system to allow the low head portion of the safety injection system to operate at full flow. Full LHSI flow transfers the energy as sensible heat directly to the sump. This causes rapid depressurization of the containment and elevation of the sump water temperature, resulting in less than the desired NPSH.

The required NPSH may be reduced by a reduction in the pump flow, or the NPSH available at a given flow rate may be increased by the injection of cold water in the pump suction. The cold water injection lowers the temperature and, therefore, the vapor pressure of the water entering the pump. Both approaches will be utilized to eliminate any potential for less than adequate NPSH to the RS and LHSI pumps. In addition, several changes to the RWST and QS systems are proposed to ensure the containment remains subatmospheric after it depressurizes. The appropriate NPSH solution for each pump, as well as any additional changes to control containment pressure, will be described in the following paragraphs.

In order to provide adequate NPSH for the RS pumps, cold QS water will be diverted to the pump suctions (see Figures 2.2-1 and 2). Approximately 150 gpm will be diverted to the inside RS pumps, and approximately 300 gpm (this additional flow above the interim flow of 250 gpm provides additional margin) will be diverted to the outside RS pumps. Orifices will be employed to provide the necessary flow split between each set of outside and inside RS pumps.

The RS pump suction fluid vapor pressure is reduced by the QS injection into the suctions of the RS pumps. The consequent increase in available NPSH for both the inside and outside RS pumps permit their operation at the full flow rate of 3,400 gpm. Based on the tests conducted at VEPCO's North Anna Site as discussed in Section 2.3.1.1 of the September 9, 1977 report, the NPSH required curve for the RS pumps shows an NPSH required of 9.8 ft at 3,400 gpm. The minimum NPSH available with QS injection for the RS pumps will always be ≥ 11.0 ft as shown in Section 2.3 of this report.

As shown in Figure 2.2-1, a new 4-in. Sch. 40 distribution system will direct the required subcooling flow from the QS pump discharge line inside containment to the RS pump suctions in the sump area. Orifices in each subcooling flow path will ensure the correct flow split. Details of the distribution piping in the sump are given in Figures 2.2-2a and 2.2-2b.

Since the outside RS pumps take suction from the containment sump through a 12-inch line and the inside RS pumps draw directly from the containment sump, a different cold water distribution method is employed for the two pump arrangements. This distribution method promotes mixing of the cold QS water with the warmer sump water before entering the pumps' inlet bell. Figure 2.2-2b shows the direct injection method used for the suction connections for the two outside RS pumps. Figure 2.2-2a shows the distribution arrangement to deliver cold water to the annular inside RS pump suctions.

The piping has been stress analyzed for operating, seismic, and water hammer conditions consistent with the QS system. Material and installation will be consistent with the material and installation of the QS system. The QS pump providing injection water to the distribution piping for a given train of RS pumps is powered from the same emergency bus as the RS pumps.

In order to provide the additional flow required for injection into the RS pumps, each QS pump will be equipped with a larger impeller installed in its existing casing. Design QS pump flow to the spray ring headers of 2,000 to 2,200 gpm has been maintained.

Once the containment returns to subatmospheric pressure following a LOCA, QS flow will be reduced to approximately 1,100 gpm per QS train to minimize subatmospheric peak pressure. This reduction is accomplished by the addition of an orifice in parallel with a motor-operated valve on each QS pump discharge (see Figure 2.2-1). The normally open motor-operated valve permits full QS flow following a LOCA until the containment is again subatmospheric. Then, on an appropriate level signal from the RWST level instrumentation, the motor-operated valve will close, and the QS flow will automatically be throttled to approximately 1,100 gpm by the orifice.

The RWST weir will be removed and elbows installed inside the RWST in order to increase the volume of the water available to the QS system. In conjunction with the RWST weir removal, eductors will be installed in the QS system to replace the gravity-feed caustic delivery system presently employed (see Figure 2.2-1). The new eductor-driven caustic delivery system will be designed to minimize the time delay for caustic addition to the QS. The system will also provide QS and ultimate sump pH control, in accordance with NRC Standard Review Plan 6.5.2.

Additional RWST water is also made available by increasing the minimum RWST fill level to elevation 787 ft-6 in. (see Figure 2.2-3). All non-Seismic, non-Category I lines entering the RWST below this level will be relocated to an elevation above this new fill level.

The LHSI pumps will be automatically throttled during all operating modes (initial injection, cold leg recirculation, and hot leg recirculation) using cavitating venturis similar in operating characteristics to those provided by Fox Valve Development Co., Inc. A cavitating venturi is designed to reduce pressure below the fluid vapor pressure and thus choke the flow at a specific flow rate. Pressure recovers downstream of the venturi to near its initial value. The total LHSI flow for one pump operation will be limited to a maximum of 3,200 gpm. For both pumps operating, the flow per pump is less than 3,200 gpm due to higher pressure losses in the discharge piping. The corresponding minimum NPSH available is always ≥ 12.0 ft as shown in Section 2.3 of this report. This limitation assures adequate NPSH is available to LHSI pumps during all modes of operation.

Tests run at VEPCO's North Anna Site verify the LHSI pump performance at the proposed flow and NPSH. Test results are given in Section 2.3.1.2 of the September 9, 1977 report. Based on these test results, the NPSH required by the LHSI pumps is 10.6 ft at 3,200 gpm.

2.3 CONTAINMENT NPSH ANALYSES

2.3.1 Analyses, Methods, and Procedures

The LOCTIC computer program is used to calculate the available net positive suction head (NPSHA) for the inside and outside containment RS pumps and the LHSI pumps. Since the most limiting assumptions for an NPSH analysis are different from those for containment integrity and depressurization analyses, these analyses must be performed for any system modification under consideration. The former ensures that the required net positive suction head (NPSHR) to support the flow values used in the analysis is available during the entire accident transient, and the latter ensures that the peak design pressure and subatmospheric containment criteria of depressurizing in less than one hour and remaining depressurized thereafter, are satisfied.

All NPSH values are referenced to the centerline of the first stage of the pump impeller.

The flow and minimum NPSH required for the pumps are as follows:

1. Inside and outside RS Pumps - 3400 gpm at 9.8 feet

2. LHSI Pump - 3200 gpm at 10.6 feet

A number of sensitivity studies were performed to determine the limiting case for NPSH.

Table 2.3-1 lists the preliminary sensitivity study input parameters used in the most recent studies, and Table 2.3-2 lists the original containment analysis input parameters. The heat sink input parameters are provided in Table 14.3-8 of the FSAR. Mass and energy release rate tables are provided in Appendix A.

<u>NPSH Sensitivity Studies - Single Failure and Break Location</u>

A single failure study was made (Table 2.3-1 lists the input parameters) in order to identify the limiting single failure cases for RS pump and LHSI pump NPSH. Table 2.3-3 shows the minimum NPSHA calculated for both the pump suction double-ended rupture (PSDER) and the hot leg double-ended rupture (HLDER) with each single failure analyzed. For the RS pumps, the HLDER with normal engineered safety features (Norm. ESF) is the worst case. For the LHSI pumps, the PSDER with min. ESF is the worst case. However, all NPSHA values are greater than, or equal to, the NPSHR values given in items 1 and 2, above.

NPSH Sensitivity Studies - Tagami

The sensitivity of NPSH to the condensing heat transfer coefficient (Tagami) is shown by comparing a calculation which uses a condensing heat transfer coefficient with a peak value equal to 4 times Tagami. The calculated NPSHA for the LHSI Pump (PSDER with min. ESF) is 12.1 feet for both 4 times Tagami and Tagami. This demonstrates NPSH is insensitive to this parameter.

NPSH Sensitivity Studies - River Water and RWST Water Temperatures

NPSHA was calculated for both the PSDER and HLDER assuming Min. ESF for the following temperature conditions to show their effect on NPSH:

1. 55 F RWST Water Temperature and 32 F River Water Temperature
2. 55 F RWST Water Temperature and 80 F River Water Temperature
3. 45 F RWST Water Temperature and 32 F River Water Temperature
4. 45 F RWST Water Temperature and 86 F River Water Temperature

Table 2.3-1 lists the remaining input parameters.

The higher RWST water temperature and/or the higher river water temperature lowers the LHSI pump NPSHA. The higher RWST water temperature and/or the lower river water temperature lowers the RS pump NPSHA. Refer to Table 2.3-4.

NPSH Sensitivity Studies - Break Size

Since Min. ESF was determined to be the most limiting single failure for LHSI pump NPSH, further analyses assuming Min. ESF were done for various break sizes. Table 2.3-5 shows the Min. NPSHA to the pumps for this analysis. The input parameters of Table 2.3-1 apply.

2.3.2 Recirculation Spray (RS) Pumps

The NPSHA analyses for the RS pumps show that NPSHA for a flow of 3400 gpm is always above the minimum required NPSH of 9.8 ft for the entire transient. A Norm. ESF HLDER is the limiting case for the RS pumps NPSH. Table 2.3-1 lists the input parameters pertinent to the RS pump limiting case. The NPSHA calculated for this limiting case is shown on Table 2.3-3. The mass and energy release rates are given in Appendix A and Figures 2.3-1 through 6 show time histories of the limiting case for the following parameters:

1. NPSHA
2. Containment Total Pressure
3. Sump Water Level
4. Sump Vapor Pressure
5. Containment Temperature
6. Sump Water Temperature
7. RS Suction Vapor Pressure
8. RS Cooler Duty
9. QS Flow
10. Condensing Heat Transfer Coefficient

2.3.3 Low Head Safety Injection Pumps

For the 3200 gpm maximum flow, NPSHA analyses for the LHSI pumps show that NPSHA is always above the minimum required NPSH of 10.6 ft for the entire transient. A Min. ESF PSDER is the limiting case for the LHSI pumps NPSH. Table 2.3-1 lists the input parameters pertinent to the LHSI pump NPSH limiting case. except that a river water temperature of 80 F is the maximum allowable for a 55 F RWST water temperature. The NPSHA calculated for this limiting case is shown in Table 2.3-4. The mass and energy release rates are given in Appendix A and Figures 2.3-7 through 12 show time histories of the limiting case for the following parameters:

1. NPSHA
2. Containment Total Pressure
3. Sump Water Level
4. Sump Vapor Pressure
5. Containment Temperature
6. Sump Water Temperature
7. RS Suction Vapor Pressure
8. RS Cooler Duty
9. QS Flow
10. Condensing Heat Transfer Coefficient

2.3.4 Containment Integrity and Depressurization

Table 2.3-6 provides the containment depressurization time and subatmospheric peak pressures for a PSDER and various single failures (Table 2.3-1 lists the input parameters used). The limiting case for containment depressurization is shown to be a PSDER with Min. ESF.

Table 2.3-7 shows the sensitivity of containment integrity and depressurization results to initial conditions.

As shown in Tables 2.3-6 and 7, the containment depressurizes in less than 1 hour and remains depressurized in each case. Also, the maximum containment pressure is less than the containment design pressure of 45 psig in each case.

The Min. ESF PSDER and the initial conditions in Table 2.3-1 is the limiting case for containment depressurization. The mass and energy release rates are given in Appendix A and Figures 2.3-13 through 17 show time histories of the limiting case for the following parameters:

1. Containment Total Pressure
2. Containment Temperature
3. RS Cooler Duty
4. QS Flow
5. Condensing Heat Transfer Coefficient

A HLDER, with the initial conditions in Table 2.3-1, is the limiting case for peak pressure (a Min. ESF PSDER results in an almost identical peak pressure). The mass and energy release rates are given in Appendix A and Figures 2.3-18 through 22 show time histories of the limiting case for the following parameters:

1. Containment Total Pressure
2. Containment Temperature
3. RS Cooler Duty
4. QS Flow
5. Condensing Heat Transfer Coefficient

2.3.5 Technical Specification

The technical specifications (3.6.1.4 and 3.6.1.5) limiting containment pressure and temperature have been revised and according to the analysis described above. These proposed changes to the technical specifications (TS) are provided in Appendix B.

TABLE 2.3-6

SENSITIVITY OF CONTAINMENT DEPRESSURIZATION TO SINGLE FAILURES

PSDER	Depressurization Time (sec)	Third Peak Pressure (psig)	Third Peak Time (sec)	Fourth Peak Pressure (psig)	Fourth Peak Time (sec)
Min ESF*	3550.	-0.09	4380.	-0.05	8220.
Norm ESF	1360.	-1.63	2580.	-1.40	4370.
LHSI Pump	1530.	-1.74	2790.	-1.48	4540.
RS Pump	1590.	-1.15	2640.	-0.74	4490.
QS Pump	2280.	-0.94	3820.	-1.12	7260.

NOTE:

*Worst case depressurization

C. ADHERENCE AND FAMILIARIZATION TO OPERATING PROCEDURES (continued)

which are established for operating personnel is to ensure that these personnel have the technical knowledge and judgement capabilities required to understand and utilize the proper procedures for the various plant operations and to take the necessary actions when procedures are not provided.

1. In the event a procedure cannot be followed as written or a procedure is not available, the activity should not be conducted unless required by an emergency or casualty situation until the procedure has been revised or prepared, approved and issued, or a decision has been made by the Plant Manager that a new or revised procedure is not required.

2. Operating personnel may take reasonable action that departs from a license condition or a Technical Specification in an emergency as per 10 CFR 50.54 (x) when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and Technical Specifications that can provide adequate or equivalent protection is immediately apparent. The guidelines of O.M. 48, Section 1, Procedure F, should be followed when taking the action. This action shall be approved, as a minimum, by a licensed Senior Operator prior to taking the action.

3. All immediate actions (circled step numbers) in the emergency operating procedures must be committed to memory. The first seven Immediate action steps of E-0, and all immediate action steps of FR-S.1 and ECA-0.0 shall be performed in specified sequence. All other immediate action steps in the EOP's are not required to be performed in the specified sequence. In the event of an emergency or casualty not covered by an approved procedure, operating personnel have the responsibility and authority to take whatever action considered necessary to prevent injury to personnel or damage to plant equipment and to place the plant in a safe condition.

4. Deviations from procedure Initial Conditions must be evaluated by Operations supervision to determine if the procedure may be conducted as intended. Normal System Arrangement is determined by present Control Room Logs, Status Boards and/or Station-controlled flow diagrams, however, hands on verification is advisable when practical.

December 16, 1986

Mr. W. T. Russell, Director
Division of Human Factors Technology
Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Russell:

On December 1, 1986, I informed you that I wished to continue pursuit of a hearing on the denial of my senior reactor operator license by Region I, reference Docket No. 55-60755. I said I would provide supporting information in the near future. This letter forwards that information in Attachments 1, 2, and 3. In addition, please consider the following remarks:

1. In each scenario, there were several malfunctions. I correctly diagnosed most of those instrument and component failures. I correctly used procedures when their use was required. In the few cases where I failed to diagnose a problem correctly, I directed action to place the plant in a safe condition. An impartial review of my performance on each scenario after dispositioning the comments in Attachment 3, should lead to a satisfactory grade on each.

2. In the implementation of the new EOP's, diagnosis of problems is not required, though it may be beneficial. My implementation of ECA-0.0 based on the symptom of both emergency busses being deenergized was proper. When, in the course of performing ECA-0.0, the symptom for continuing that procedure was removed, I directed transition to the appropriate procedure.

3. Both INPO and the NRC are strong proponents of teamwork and diagnostic skills training. This type of training has got to be part of a candidate's initial training. During the training of candidates on simulator exercises, communications and teamwork are encouraged. If it continues to be practice for NRC examiners to fail candidates who initially vocalize thoughts and then allow them to be changed based on crew interaction, it will become difficult to convince candidate crews under examination that they must communicate and interact.

4. In at least two instances, the examiner concluded that my actions were not indicative of a safe operator. These instances contributed to the determination of a "fail" grade. My response has been to show that proper actions were taken as a result of good crew interaction. I am not attempting to minimize the actions, but rather to change the perspective in which those actions were viewed.

5. I have over twenty years of nuclear power plant operating experience, including six as a certified operating supervisor at Shippingport. I never had so much as a reactor trip during my on-shift assignments. I personally directed the reactor startup training of many candidates at Shippingport, some of whom later licensed on the Beaver Valley Power Station. Through my various supervisory positions including Station Superintendent, I enforced the requirement for adhering to and following procedures. My ability as a safe, conservative operator has never been challenged. Such a record should certainly reduce the significance of negative observations formed during less than four (4) hours of examination.

I request that all of this information be considered and the grade for both the written and simulator portions of the examination be changed to "pass". If the denial is upheld, I request that copies of all information possessed by Region I concerning my examination, including log sheets and examiner's comment sheets, be forwarded to me so that I can prepare for the next phase of this appeal process.

Sincerely,

A. J. Morabito

A. J. Morabito

AJM:dmd

ATTACHMENT 1

Comments on Written Exam

6.03.b The question read: "What three design features minimize the effects of a rupture. . . " One of the answers given in the NRC key was thermal barrier piping designed for 2485 psig. The way the question is written, I assumed it proposed that the rupture had already occurred. The piping design feature has no bearing on the question since the question leads the reader to the conclusion that the piping is already broken. The answers given in my original examination are more correct than the NRC key. Two-thirds (2/3) credit should be given.

6.07.b Region I recognizes that there are a variety of correct answers to this question. The answers I gave are acceptable. Reinstate the question and give full credit.

6.08.c Where the undervoltage device is located has no bearing on the answer to the question. As shown on the attached supplemental sheets 1 & 2, it is the voltage and frequency of the diesel generator output that provides the permissive signal. This is the same voltage as the 4 KV emergency bus. Full credit should be given.

6.09.c The question asked for the reason recirculation spray water pressure is greater than river water pressure in the recirculation spray heat exchangers. The correct answer to the question is to ensure that spray water leaks into river water rather than vice versa. The major benefit of leakage in that direction is to avoid dilution of the boron in the spray water and thus maintain shutdown margin. That is not the only benefit, supplemental sheet 3 shows that the leakage can be monitored by radiation monitors; thus allowing detection of the leaking component and subsequent isolation of it. My answer gave the correct reason for higher pressure on the recirculation spray side of the heat exchanger and provided valid reasons why that was important. Three-fourths credit should be given.

Comments on regrade:

7.02.c I said, "core exit thermocouples < 1200° . . ." or " core exit thermocouples < 700° . . ." that meant all, not just the five hottest, but certainly it includes the five hottest. Give back 0.2 points.

ATTACHMENT 1

7.03.c The question asked for parameters to terminate SI. My answer answered the question. Give back 0.2 points.

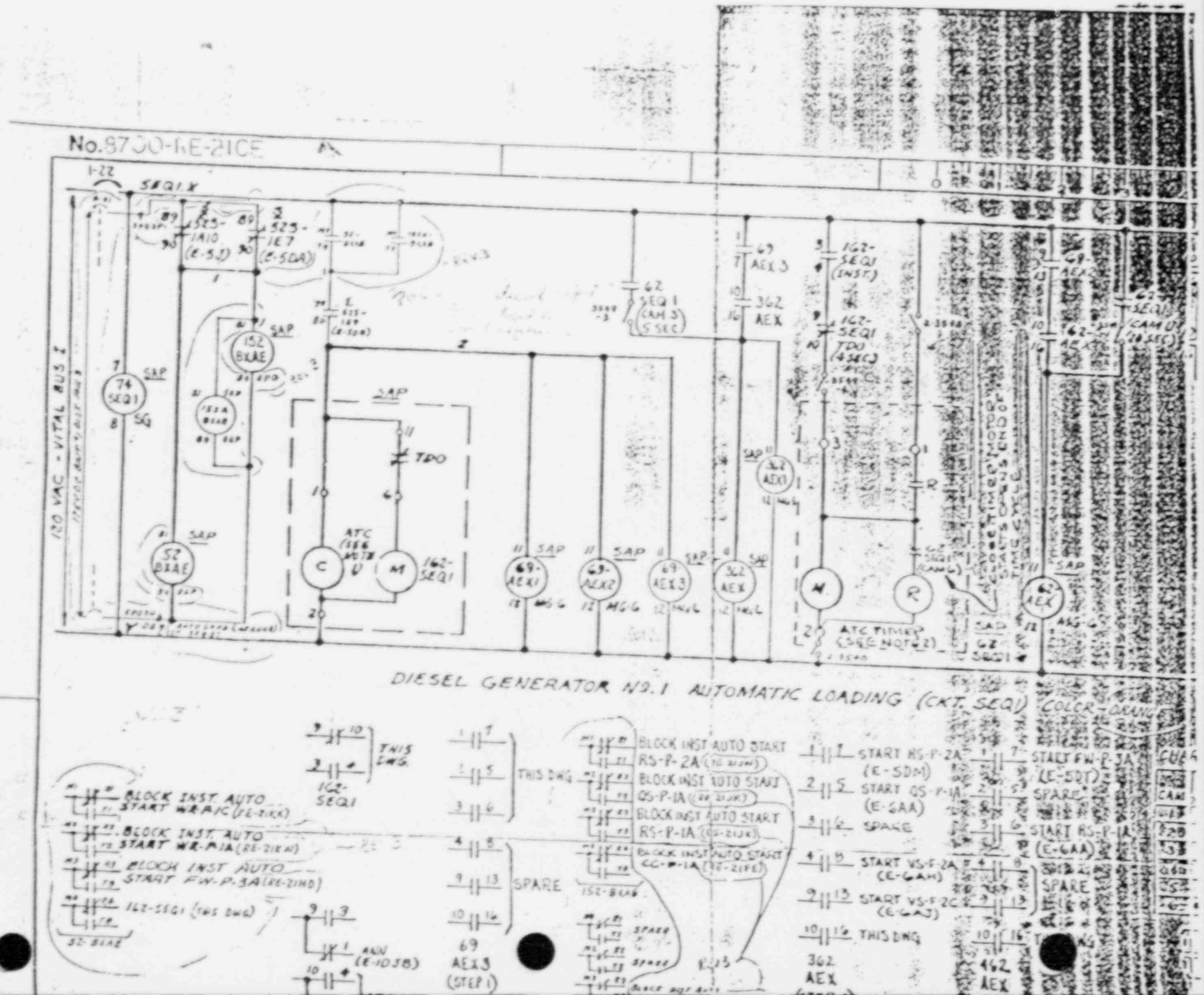
7.10.b The question was worth 0.5 points. My answer was sastisfactory. It was at least worth somewhat more than half credit. Give back 0.2 points.

8.05 The question specifically stated: "Discuss . . . in terms of preventing release of radioactivity to the environment." Give back 1.0 points.

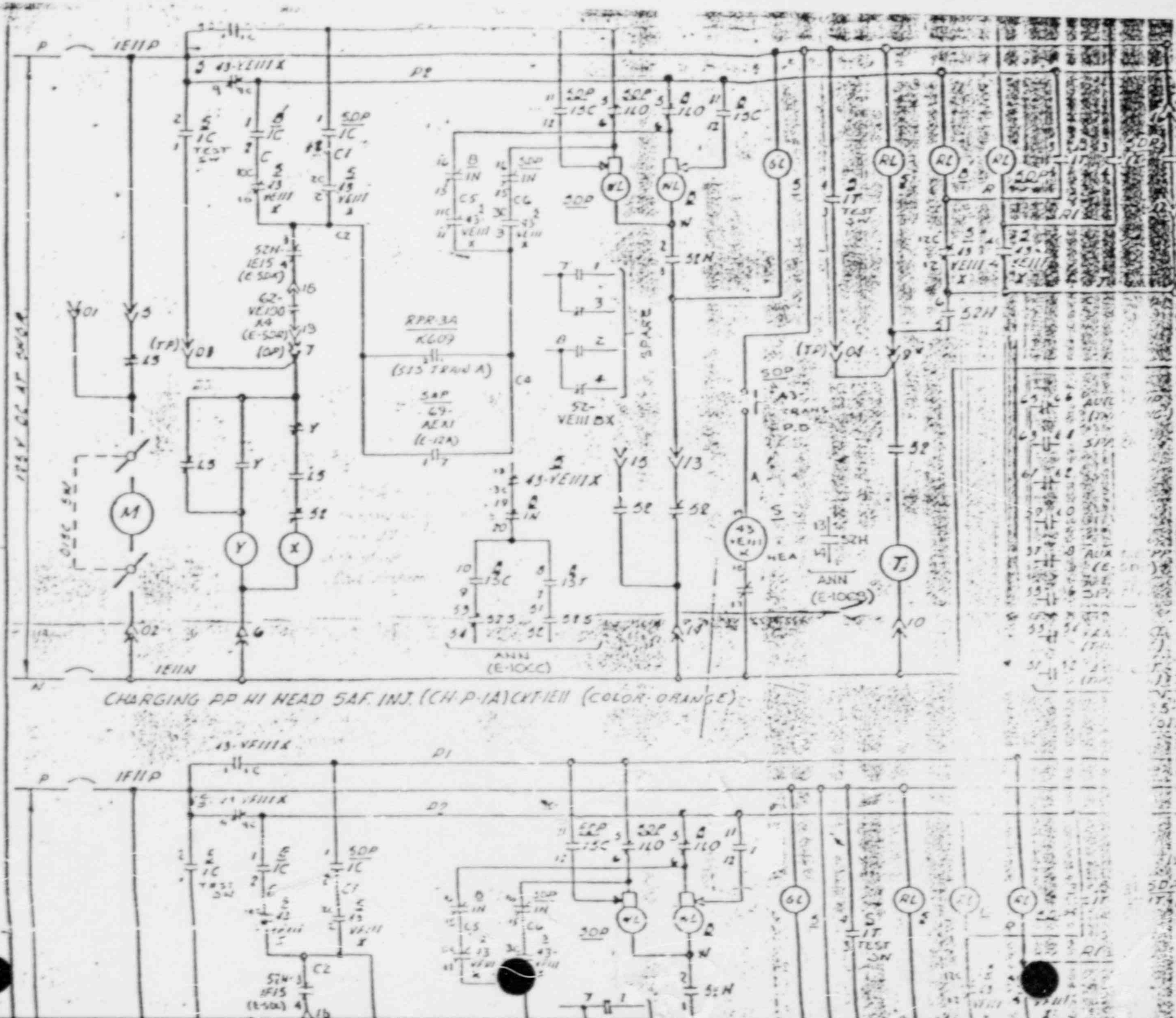
8.06.a. The word "immediately" is not in the answer, but the answer certainly implies immediately. Give back 0.3 points.

8.11.a. My answer said, "complete the shift turnover checklist. . ." Review of logs is one of the items on the checklist. Give back 0.33 points.

Supplemental sheet 1 to Attachment 1



Supplemental sheet 2 to Attachment 1



Supplemental sheet 5 to attachment 1

type of particle could conceivably pass lengthwise through the screen and cause clogging of a spray nozzle. However, since the final screen opening is smaller than the smallest spray nozzle size, which is 0.360 inch, such an occurrence is considered to be highly improbable. For the recirculation spray subsystems, the screen assemblies in the containment sumps are arranged so that no single failure could result in the clogging of all suction points (Section 6.4.2) to the recirculation spray subsystems. A main screen assembly failure, coupled with the plugging or failure of the suction point caps, must occur for any one of the suction points to be lost. Sufficient area has been provided to ensure that system operation during accident conditions is not impaired and entrance flow velocities are low enough to prevent entrainment of most small particles. System overdesign allows for some plugging or loss of function, in spite of the foregoing.

Consideration has been given to the possibility of a reaction between sodium hydroxide and atmospheric carbon dioxide forming a precipitate in the chemical addition tank. If the temperature of the gas space in the tank varies over a range of 60 F each day and all the carbon dioxide which enters due to this breathing reacts, 90 years would be required to react with 1 percent of the stored caustic. The sodium carbonate formed by this reaction would remain soluble at 45 F.

During normal unit operation, the recirculation spray coolers are dry on the shell side and on the tube side are filled with river water with corrosion preventatives as described in Section 9.9.2. For long term operation, on the order of weeks, there may be some fouling of the tubes on the river water side, with resultant loss in heat transfer capability. A fouling factor of zero is assumed because the loss of heat transfer capability will be more than offset by the decrease in necessary heat load due to decreasing decay heat production. One day after a loss-of-coolant accident, the drop in decay heat is such that one pump and heat exchanger has sufficient heat-removal capacity to hold the containment depressurized. With an expected maximum river water temperature of 86 F, the recirculation spray subsystem design is conservative, with a minimum 100 percent backup capacity at the onset of an accident. Within one day after the LOCA, the backup capacity exceeds 400 percent.

The recirculation spray coolers have welded construction at all points where there is a potential for leakage of radioactive recirculation water into the river water. The maximum pressure differential which can occur between the river water and the recirculation water is 150 psi; under these conditions, leakage flow from the recirculation spray subsystems is toward the river water system. The river water is monitored for leakage by means of radiation monitors. The defective subsystems are shut down if leakage above the allowable values (within the limits of 10CFR20) is detected. As a result of the above pressure difference, inleakage of nonborated water into the containment, causing dilution of the borated water in the containment, is not possible.

ATTACHMENT 2

Comments on Oral Exam

The Region did not address my comments on their grading of the oral exam. Their reason for this is that the pass decision would not be affected. Nevertheless, I request that my comments on the oral exam, as submitted to Region I by my letter of September 11, 1986, be evaluated. My purpose here is to show that the judgement of the examiner may have been influenced by his knowledge of items and facts in other plants that are not the same at Beaver Valley.

ATTACHMENT 3

Comments on Simulator Exam

The Region claims that I did not challenge the validity of the examiner's observations but rather attempted to minimize their significance by referring to good crew interaction and teamwork. Let me state it quite clearly in this letter - I do challenge the validity of many of the examiner's observations and request the I be informed of how all of the observations weighed into the determination of a "fail" grade.

The following are my responses to the Region's responses to my comments of September 11, 1986.

ES-302-11, Attachment 1/4, Comment 1:

As stated in my September 11, 1986 letter, I directed a 10% power reduction and stopped it after 5%. I did not need a procedure to reduce power 10%. My concern that two nuclear instruments might be malfunctioniung was caused by observation of 4 recorders, NR-N141, NR-N142, NR-N143, and NR-N144 on vertical board B. The meters record the 8 power range nuclear detector outputs. Each meter records two detector outputs. NR-N141 and NR-N142 record the four top detectors; NR-N143 and NR-N144 record the four bottom detectors. I observed that two meters were showing decreasing traces while two were showing increasing traces; see supplemental pages 1 and 2. I directed a 10% power decrease to determine if all instruments responded to a power change. After a 5% decrease, I observed that all detector outputs responded with precision. I stopped the power decrease at that point.

ES-302-11, Attachment 1/4, Comment 2:

There was no incorrect thought process. Had the scenario progressed to this point quicker, the hot leg temperature could have been greater than 395°. I asked the operator to verify the temperature, I assumed the answer would be yes. When he responded no, I reacted accordingly.

ES-302-11, Attachment 1/4, Comment 4:

There are valid conditions for which procedures are not required, that is the case for ES-302-11, Attachment 1/4, Comment 1. There are other cases when procedures must be followed. It is required that the immediate action steps be verified. They also must be verified quickly to ensure that if certain important components are not operating properly the appropriate actions can be taken. In my September 11, 1986 letter, I asked to be informed how much weight this comment carried. That answer has not been given. I never made the statement that the Region says I made. See original comments.

ATTACHMENT 3

ES-302-11, Attachment 2/4, Comment 2:

My original comment of September 11, 1986, is valid and accurate. It is logical to assume that I was graded harder than I might otherwise have been had the examiner realized that there were no position indicating lights for the residual heat relief valve.

ES-302-11, Attachment 2/4, Comment 4:

The examiner still does not know that there is no CIA annunciator. How much did the examiner's misinformation influence my grade? See original comments.

ES-302-11, Attachment 3/4, Comment 1:

A copy of the 1B steam generator feed flow, steam flow, and level strip charts is attached, see supplemental sheet 3. Note that initially the only symptom of a problem was a spike on feed flow and level. This spike was very similar to spiking observed in the first scenario. Much effort was put into the first spiking problem before it was finally made clear to the crew that this was a simulator problem which should be disregarded. The action to call for I&C investigation, especially in light of the earlier problem, was not un-reasonable.

ES-302-11, Attachment 3/4, Comment 2:

My comment of September 11, 1986, is accurate and valid.

ES-302-11, Attachment 4/4, Comment 1:

My comment of September 11, 1986, is accurate and valid. The new symptom-based EOP's do not require diagnosis. They require action based on the symptom. Because of my quick reaction, the symptoms I saw were both emergency busses de-energized. ECA-0.0 correctly transitioned me to E-0 when the symptoms no longer existed.

ES-302-11, Attachment 4/4, Comment 2:

The examiner is not qualified to say if the use of hand signals is satisfactory or not. The operator and I knew exactly what we were talking about and we used other indications to verify our assumptions. I directed correct action and I expected to be congratulated for it, not criticized. I was aware of how much the subject radiation monitor was reading above the orange mark prior to the event occurring due to my pre-event tour of the control room.

ES-302-11, Attachment 4/4, Comment 3:

Proper actions were taken under my direction.

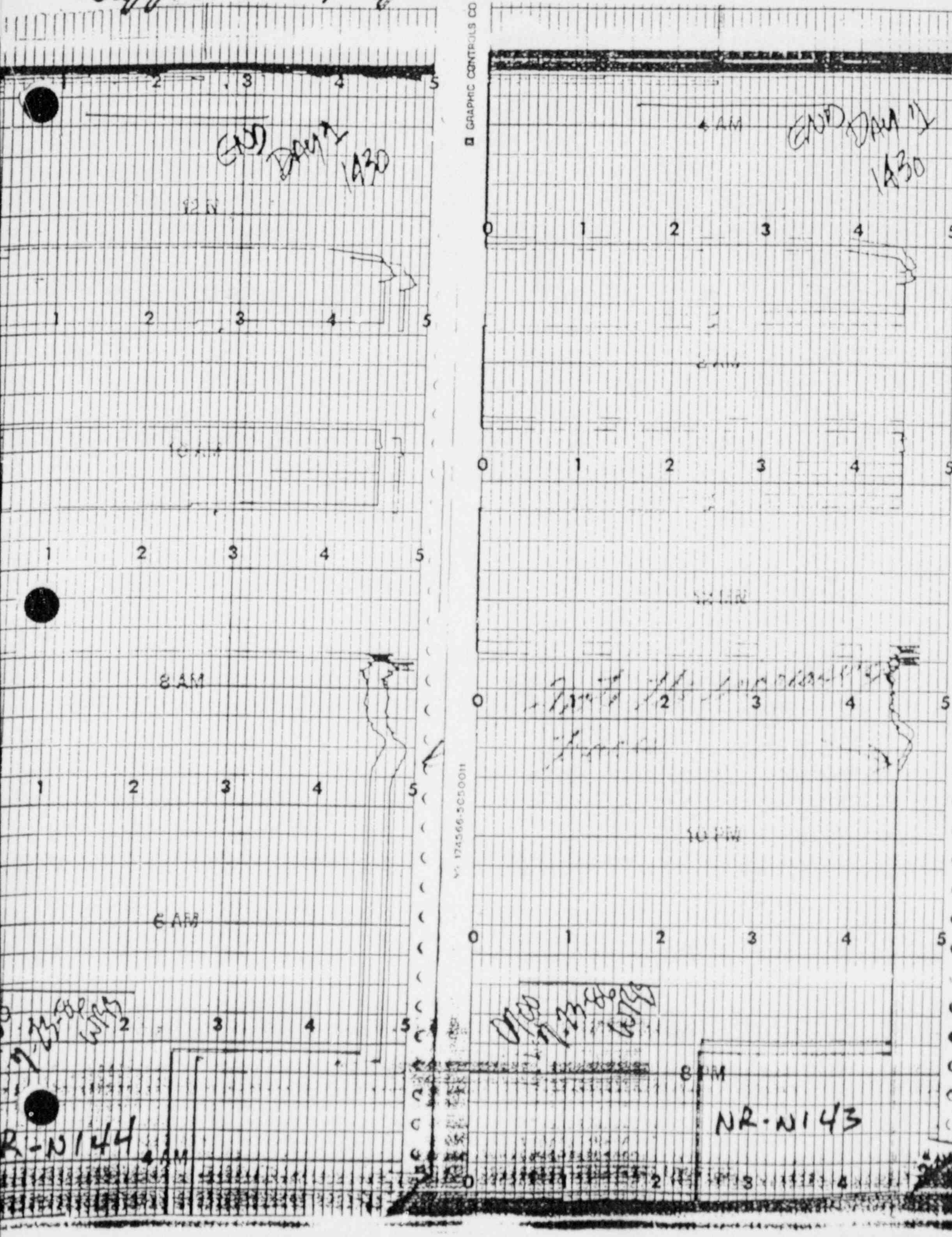
Comments on Additional Comments Made by the Region:

1. Emergency boration per Ch. 7, Section 4S is required for the low-low rod insertion limit alarm. We did not have this alarm.

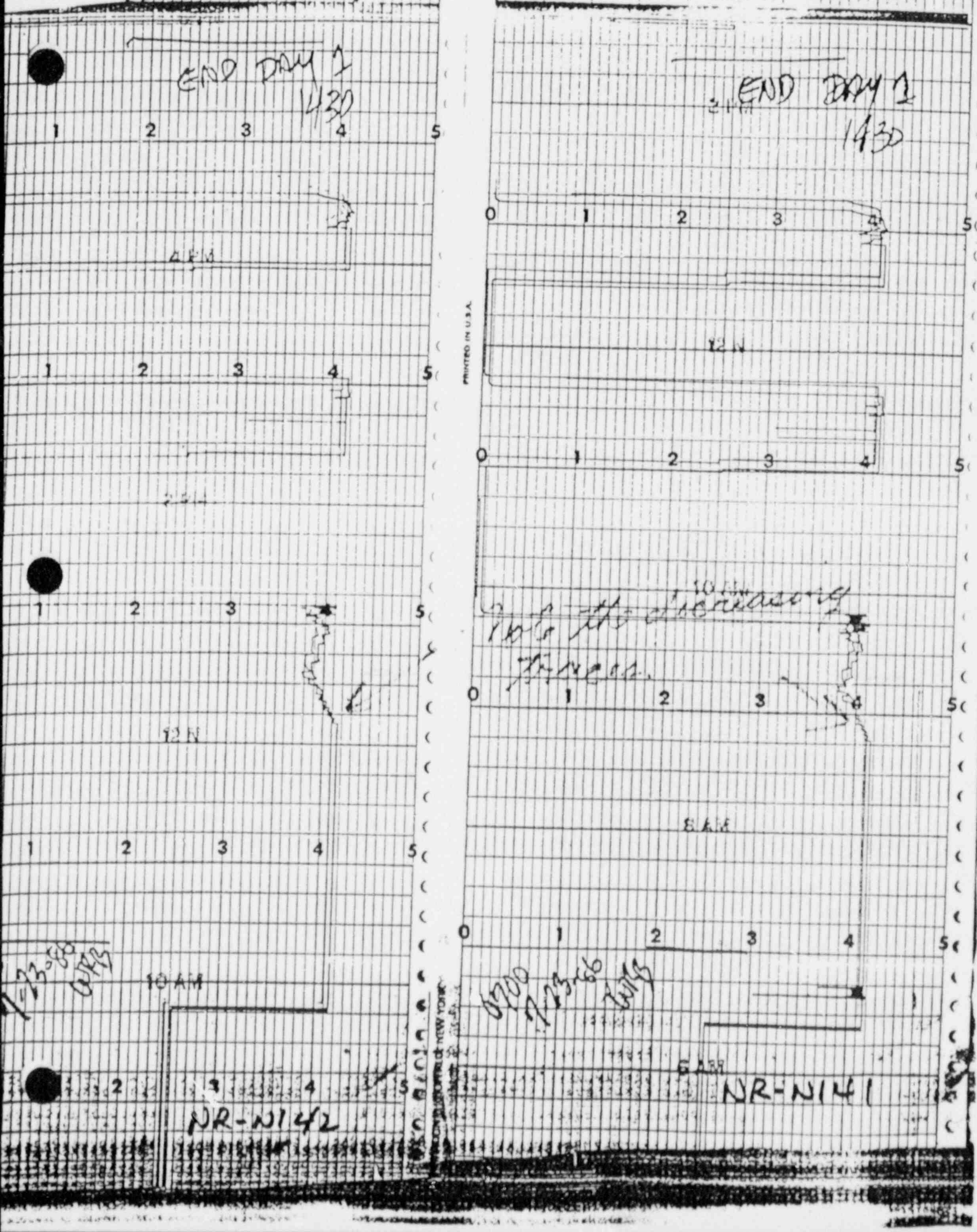
2. The RO has the ability to secure boration anytime he/she feels it is not required; provided the boration was not initiated for a low-low RIL condition. SRO permission is not needed. Several things were occurring simultaneously, including a load reduction at my direction. The plant never got out of control. We continuously moved it in the conservative, safe direction.

3. I continuously communicated with my operators. I listened to their opinions and, when necessary, overrode their objections in order to move the plant in the safe direction as in the case of my directed load reduction during the dilution accident. Excellent supervisory ability and managerial skill was exhibited.

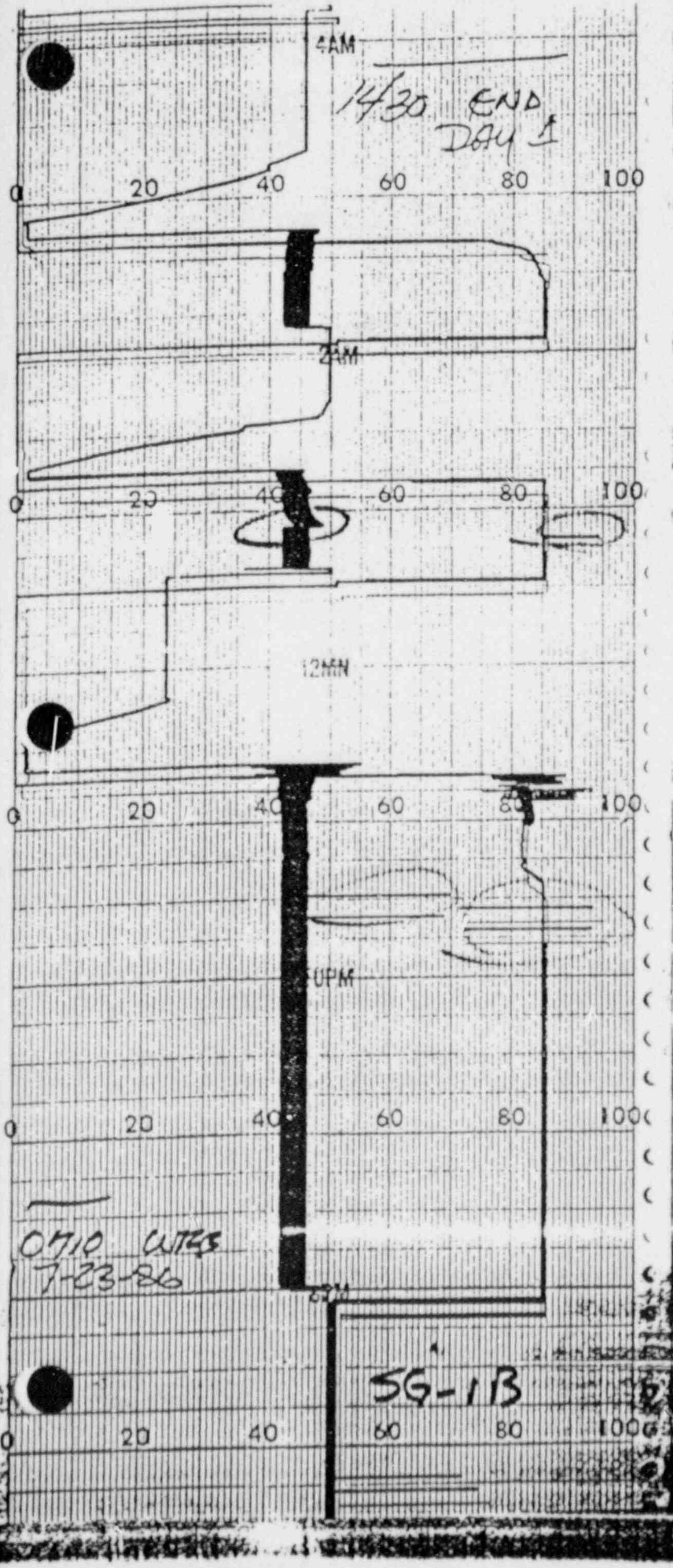
supplemental page 1 to attachment --



Supplemental page 2 to Attachment 3



Supplemental page 3 to Attachment 3



Note that the initial spikes on feed and level were similar to spiking which had occurred in the first scenario. The level indication did stay high for a period of time but the operator wouldn't have known this initially.

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

February 2, 1987

Mr. Alfred J. Morabito
685 Tulip Drive
New Brighton, PA 15066

Dear Mr. Morabito:

In response to your letter to me dated December 16, 1986, we have reviewed the grading of the written examination administered to you on July 22, 1986 and the simulator examination administered to you on July 23, 1986. Our responses to your specific comments are enclosed. Taking your comments into consideration, we find that you failed both the written and simulator examinations. Therefore your appeal of your license denial has been forwarded to the Office of the General Counsel. They will contact you concerning the details of the hearing process. If you have additional concerns on this matter, please contact Ted Szymanski, Acting Chief, Operator Licensing Branch at (301) 492-4368.

Sincerely,

BABoger for

William T. Russell, Director
Division of Human Factors Technology
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: W. Kane, RI

COMMENTS ON THE WRITTEN EXAMINATION

6.03.b No additional credit allowed. Rupture of the RCP thermal barrier is not the same as rupture of cooling water supply or return piping.

6.07.b Agree that the question should be reinstated and award ½ credit (+0.5 points) for his answers. The candidate's second answer is the reason for isolating the leaking (faulted) generator, which is his first answer. Therefore, he should only be awarded credit for one answer.

6.08.c No additional credit allowed. The additional documentation provided was inconclusive in that it did not locate the signal device or explain the permissive signal(s) required to allow sequential loading following a loss of off site power.

6.09.c Award ½ credit (+0.4 points). The documentation provided shows that leak detection allowing for isolation of the heat exchanger is a benefit of maintaining recirculation water pressure higher than river water pressure. The important reason, to prevent dilution of the recirculation spray water and maintain the shutdown margin, was not given in the answer.

7.02.c thru 8.11.a Comments on the grading of these questions are not relevant to the pass/fail decision on the written examination; and, therefore, were not reviewed. The applicant passed sections 7 and 8 of the written examination and had better than 80 percent overall score on the examination. The license denial is based on a score of less than 70 percent (failure) on section 6 of the written examination and failure of the simulator portion of the operating examination.

COMMENTS ON SIMULATOR EXAMINATION

ES-302-11, Attachment 1/4 Comment 1:

The documentation provided, recorder strip chart of nuclear instrument channels, does not challenge the observation or comments of the examiner concerning use of procedures.

ES-302-11 Attachment 3/4 comment 1:

The examiner's comment is that the candidate did not identify an open feedwater regulating valve bypass valve <u>during</u> the diagnosis of apparent mismatch of feedwater flows. The strip chart of steam generator feed flow, steam flow and level provides no basis to challenge this comment.

ES-302-11, Attachment 1/4 comments 2 and 4, Attachment 2/4 comments 2 and 4, Attachment 3/4 comment 2, Attachment 4/4 comments 1 through 3; and comments on additional comments made by the region:

No documentation was provided to challenge the examiner's conclusions, and therefore the reviewers lacked any references on which to evaluate the candidate's comments.

Grade on Section 6 After OLB Review

QUESTION	Point Value	Points Lost
6.01.a	0.5	
b	1.5	-0.6
6.02.a	Deleted	
b	1.6	
c	0.5	
6.03.a	0.7	
b	1.5	-1.0
6.04.a	0.5	
b	0.5	
c	1.0	-0.5
6.05.a	Deleted	
6.06.a	0.4	-0.4
b	1.5	-0.2
c	0.4	
d	0.4	
6.07.a	0.5	-0.5
b	1.0	-0.5
c	1.0	-0.25
6.08.a	1.5	-0.15
b	0.4	-0.4
c	0.4	-0.4
6.09.a	0.8	-0.4
b	0.8	
c	0.8	-0.4
d	0.8	-0.8
6.10.a	0.8	
b	0.4	-0.2
c	2.0	-0.5
d	0.4	
TOTAL	22.6	7.2

15.4/22.6 X 100% = 68.1%

B. Non-Normal Conditions

1. Complement of Shift Personnel Less Than Minimum Functional Complement

Under emergency conditions that may develop during a shift, such as personnel injury or sickness, the following steps must be instituted:

a. Immediately after the emergency is under control, call out the required personnel to return to a minimum functional complement, for the current Operational Mode, within 2 hours if at all possible.

b. Should the personnel required to return to a minimum functional complement not be available for duty for any reason, then the reactor shall be brought to the hot standby condition utilizing normal operating procedures. During the course of this shutdown, if the required number of employees arrive on the site, the controlled shutdown may be terminated and the Station may resume normal operation to provide the Station load requested by the Duquesne Light Company System Operations load dispatcher.

2. Control Room Order

When the Control Room activities become so numerous that, in the opinion of the NSS, the proper amount of attention and review is not available to insure that nuclear safety is being properly implemented, all activities not required for safe operation should be suspended until such time that these functions can be re-initiated in a controlled and disciplined manner.

C. Use of Instruments

1. No measurements shall be obtained or tests performed utilizing measuring and test equipment that has a red calibration sticker which has passed its calibration due date, or is otherwise proved to be out of calibration as described in paragraph VI.D below.

2. All instruments proven to be out of calibration shall be tagged out-of-service or out-of-calibration by a responsible supervisor.

D. Response to Instrument Indications

1. All Station instrument readings are to be considered accurate and reliable, unless demonstrated to be false

September 11, 1986

Harry B. Kister, Chief
Projects Branch No. 1
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, PA 19406

Dear Sir:

In accordance with the provisions of Part 2, Title 10, Code of Federal Regulations, Section 2.103(b)(2), I request a hearing be held concerning the denial of my application for a senior operator license. Attachments A & B to this letter list the questions which were graded incorrectly or too severely. The bases for my opinion are also contained therein.

Yours truly,

Alfred J. Morabito

Alfred J. Morabito
685 Tulip Drive
New Brighton, PA
15066

ATTACHMENT A

Review of 7/22/86 NRC Written
Exam of Al Morabito

6.03.b The question asked for three design features of the component cooling water system that mimimize the effects of a RCP thermal barrier rupture. Two valid design features were given. The automatic isolation of the line supplying the thermal barrier on high flow and the fact that this line is seperate from the rest of the cooling water supplied to the RCP so that all pump cooling is not lost if the thermal barrier ruptures. 1.0 of 1.5 points were deducted. Only .5 points should be deducted since two of three design features were given.

6.07.b The question asked for two reasons why the MSIV's are required to close during a steam line rupture. The answer key contained the bases for the Tech. Spec. on the MSIV's which are to minimize the reactivity addition due to RCS cooldown and to limit the pressure rise in containment. The way that the MSIV's accomplish these bases is by isolating the faulted steam generator and preventing the non-faulted steam generators from blowing down through the break. This was the answer given. The quesion did not ask for the Tech. Spec. bases. Full credit should be given.

6.08.c The question asked for the signal that allows sequential loading of an Emergency Diesel Generator following a loss of offsite power. The answer required by the key was "Permissive signal from associated undervoltage devices". The answer given was "4KV Emergency Bus voltage at rated voltage". Both answers say the same thing. Full credit was deducted. Full credit should be given.

6.09.b The question asked for the purpose of the orifice parallel to the Quench Spray System flow cut-back valves. The answer key gave the purpose of the cut-back valves (i.e; reduce flow to minimize negative containment pressure). The answer given, to allow some flow to cool the sump for NPSH requirements and to allow all NaOH to be added, is the purpose of the orifices. Full credit was deducted and should be given.

6.09.c The question asked for the reason that the recirc. spray side pressure is greater than the river water side pressure in the recirc. spray heat exchangers. The answers given, although not the answer on the key, are valid advantages to this design. Full credit was deducted. Partial credit should be given.

6.10.a The question asked, in part, what problems exist if SI Accumulator pressure is significantly above its limit. The answer given was that N_2 gas could be injected into the RCS during accumulator injection. This was not the answer on the Key and full credit was deducted for this half of the question. Both ECA-0.0 "Loss of All AC Power" and FR-C.1 "Response to Inadequate Core Cooling" limit RCS depressurization to a certain pressure to prevent injecting N_2 gas into the system from the accumulators. These pressures assume that accumulator pressure is within limits. If pressure was significantly above its limit, the pressure limits given in these two EOP's could be invalid and the problem stated in the answer could occur. Partial credit should be given.

	5	6	7	8	TOTAL	
NRC Grade	87.6	59.7	85.3	96.3	82.2	FAIL
Re-grade	87.6	74.3*	85.3	96.3	86.2	PASS

*Assumes 1/2 credit given back for questions 6.09.c and 6.10.a.

ATTACHMENT B

ES-301 8.b. (refers to ES-305 8.B.4)

Upgrade ES-305 8.B.4 from U to M for the following reasons:

1. When asked about the evacuation of personnel, I informed the examiner that I would initiate personnel accountability, perform search and rescue as necessary for personnel not accounted for, and evacuate personnel to an offsite assembly point, Hookstown Grange for example, by the lowest exposure path as determined by Radiation Control.

2. I informed the examiner that the procedure was probably in the Emergency Plan Implementing Procedures. When I couldn't find it after a few minutes of searching, I explained that I would certainly call on an assistant to help find it if I needed it as the Emergency Director.

3. I did locate the procedure in less time than it takes to perform personnel accountability which, I informed the examiner, I would have initiated prior to evacuation.

4. EPP/IP 3.1, Section E, steps 1.7, 1.8, 1.9 verify the accuracy of my response to the examiner that I would implement accountability, search & rescue and radcon monitoring.

5. These procedural requirements are well known to me by virtue of my several years of responsibility for emergency planning at the Shippingport Atomic Power Station in the capacity of Control Center Supervisor and later as Station Superintendent.

ES-305 B.5.1.c. (NRC Form 157c, Page 6)

I do not request a grade change. I do request consideration of the following comments to the extent that the existing grade may have affected my overall grade.

1. The pulling of the source range fuses was a recommended action (and essentially required by the instructors) during my simulator training evolutions.

2. I thoroughly explained the reason for pulling the fuses to the examiner.

3. The fuses are currently not installed following the return to power after the fifth refueling outage. The status of the fuses is carried on the shift supervisor's carryover log. Current plans are to proceduralize the pulling of the fuses.

4. The Emergency Operating Procedures acknowledge the fact that the fuses may not be inserted and direct that they be inserted prior to the need to re-energize the instruments. Refer to procedure ES 0.1 step 10.b.

5. Why assess a candidate knowledge of "normal procedures" on the basis of his performance of an action which is permissible and on which he has been drilled but which is not in a procedure? Does that tell you anything about his knowledge of the actual procedure(s)?

ES-305 B.2.5.H. (NRC Form 157c, page 4)

Upgrade from U to S for the following reasons:

1. My answer was correct. See OM Ch. 13, Section 1.13.3, 1RC valve list, page 3 of 4, which lists the NSA for those valves as "open".

Upgrade from U to S for the following reasons:

1. I did not consider that two power range (PR) instruments were inoperable. I thought that two were reading different than the other two. I did not know why. Later analysis showed that I misinterpreted the implications. Several items were checked such as instrument readings on control panel and on the NIS panel. Rod positions or inadvertant rod motion and delta flux were checked. No reason for the apparent discrepancy could be found. I knew that if two instruments were failed I would have to order a shutdown per Tech Specs. I directed a 10% power decrease to accomplish two things:

   a. Diagnose whether or not there was a real PR instrument problem.

   b. Commence reducing power to achieve Mode 3 conditions if the PR instruments did not respond to the power reduction. This approach was used to gather data using good sound management techniques prior to making a decision to call the instruments inoperable. The power reduction was of the magnitude of 5%. There is no requirement to use a procedure for a small load decrease. In fact, SAP 4, page 41 of 52, paragraph 6, allows small load changes without the procedure being present.

2. As a result of the power reduction, all four PR instruments were observed to track accurately and precisely so I directed that the power reduction be stopped.

3. Subsequent analysis of the scenario with the examiners led to the realization that an undetected dilution was occurring. My action in ordering the load reduction maintained temperature and power level in spec and would have eventually led to the crew discovering that a dilution was occurring if the scenario had not been time-programmed to cause other malfunctions leading to a reactor trip.

ES-305 C.2.5.E

Upgrade from M to S for the following reasons:

1. When I informed the examiner that I did not know if steam would be flowing to the turbine driven Auxiliary Feedwater pump during resetting operations, I also informed him that it would depend on whether or not the operator in the control room had closed the steam supply valves. Depending on whether he did or not steam would or would not be flowing.

2. I informed the examiner that if I were resetting the throttle in response to an overspeed trip, I would be in communication with the control room to determine steam supply valve position.

ES-305 C.7.1.C

Upgrade from M to S or delete for the following reasons:

1. I informed the examiner that I wasn't sure if the diesel powered air compressor was covered by Tech Specs or not. After he and Mr. [illegible] few minutes, I informed him that if the air compressor was not covered by Tech Specs, it was probably covered by an OST. It is. Refer to OST 1.36.12.

2. The examiner has based his entire opinion of my knowledge of Diesel Generator Tech Specs on the fact that I didn't know that a backup piece of support equipment was <u>not</u> covered by Tech Specs.

Delete this grade and do not permit it to influence the overall grade for the following reasons:

1. I do not remember being asked such a question.

2. I have never had difficulty converting PPM to Delta K/K and can produce training records to verify this.

3. An example of my capability in this regard is the satisfactory performance of reactivity changes in the operating plant involving boration, dilution and power changes where some division of reactivity had to be accounted for between boron worth and rod worth.

ES-305 D.8.C.2

Upgrade from M to S or delete for the following reasons:

1. Trainees are specifically restricted from knowing the combination to the NSS key cabinet.

2. My not knowing something that I am not supposed to know or allowed to know doesn't tell the examiner anything about my knowledge of plant security.

3. Security procedures direct that only the operations supervisors on shift have access to the safe combination. Trainees should not know the combination.

ES-301-11, Attachment 1/4, comment 1

This is an invalid comment. See my comments to ES-305, 5.2.A.

ES-302-11, Attachment 1/4, comment 2

This comment is accurate, but the inference that the examiner draws is not. We try to avoid discouraging trainees from thinking aloud and talking amongst themselves even though their thinking may be incorrect. This promotes crew interaction. Crew interaction is exactly what occurred in this case. As a result of the operators answer, I directed the correct operation. This interaction and correct procedure performance should lead to a satisfactory grade.

ES-302-11, Attachment 1/4, comment 3

This comment is incorrect. I immediately detected that channel 459 had failed by noting that it read high compared to the other two channels. I switched it out of the control scheme before any appreciable level or pressure change occurred. These actions are in accordance with the alarm response procedure. The additional actions in the alarm response procedure are follow-ups to an actual level and pressure change. In addition, there were only two (2) minutes between the onset of the channel 459 failure and the order to commence power decrease. During that time level in pressurizer started decreasing and volume control tank low pressure alarm occurred signifying the onset of the S/G tube rupture called for by the scenario. This did not allow any time to refer to an alarm response procedure for a failure for which I had already taken corrective action. My priorities for response were directed to the ensuing tube rupture and subsequent loss of RCS inventory. See the attached alarm printer printout and pressurizer level chart.

ES-302-11, Attachment 1/4, comment 4

How much weight does this comment carry in determining the overall grade?

ES-302-11 attach 1/4 Comment 3

Note the short time interval

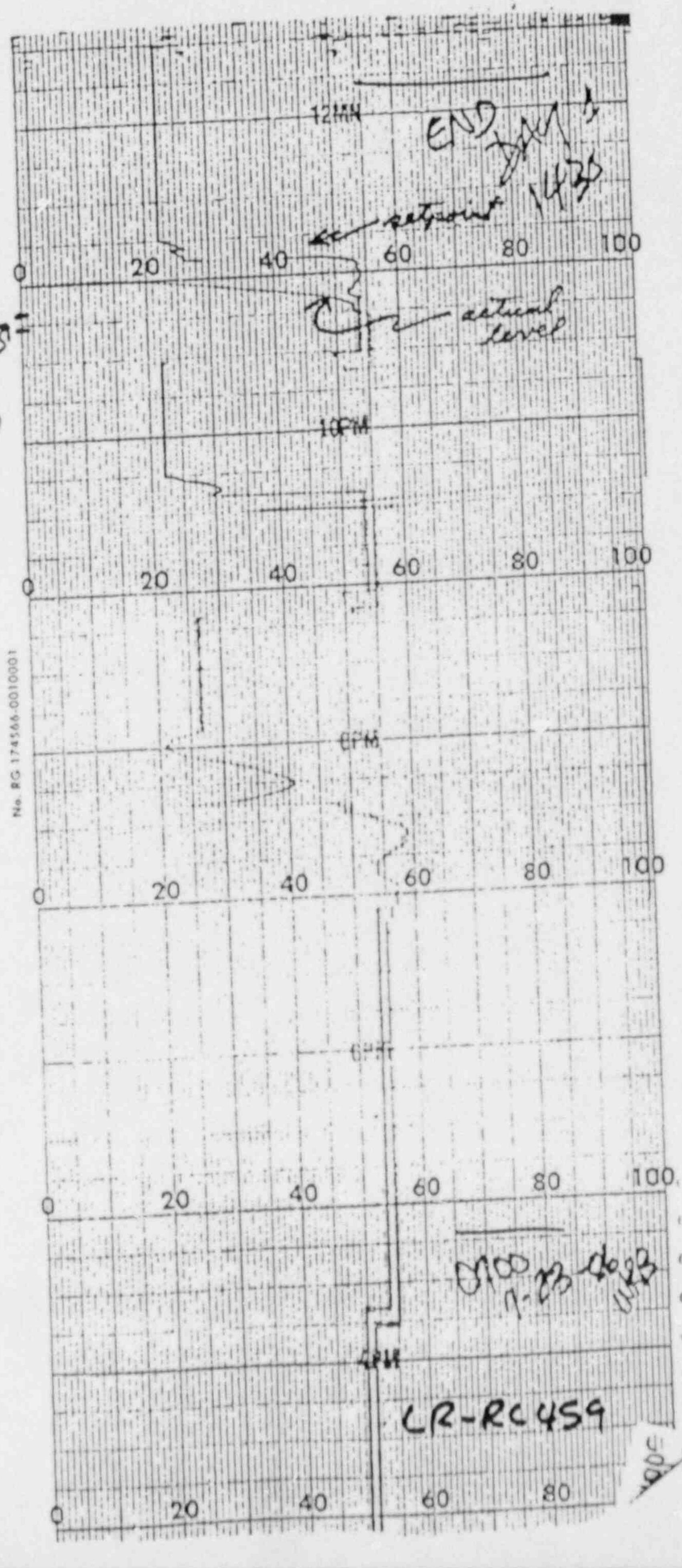
```
1307 C0026  CONT ROD BANK D GROUP 1 POS B08 DEV FROM BANK -   1? STEPS
1307 C0027  CONT ROD BANK D GROUP 1 POS H1? DEV FROM BANK -   1? STEPS
1307 C0028  CONT ROD BANK D GROUP 1 POS P0? DEV FROM BANK -   1? STEPS
1307 C0029  CONT ROD BANK D GROUP 2 POS F06 DEV FROM BANK -   1? STEPS
1307 C0030  CONT ROD BANK D GROUP 2 POS F10 DEV FROM BANK -   1? STEPS
1307 C0031  CONT ROD BANK D GROUP 2 POS K10 DEV FROM BANK -   1? STEPS
1307 C0032  CONT ROD BANK D GROUP 2 POS K06 DEV FROM BANK -   1? STEPS
 1316  RETRN HI  F0134A  AI  NONREGEN HX LTDN OUT F    F-CH150   63.5   H   90.0   GPM
 1317  ALARM CI  L0480D  CI  PRESSURIZER HI L 1 PART RE          TR
 1317  ALARM HI  L0480A  AI  PRESSURIZER 1 LEVEL       L-RC459   100.0  H   60.0   PC
 1318  ALARM LO  P0139A  AI  VOLUME CONTROL TANK P     P-CH117   19.?   L   20.0   PSIG
1318 REDUNDANT MEAS STATUS REPORT  GROUP 21   ALARM
     L0480A PRESSURIZER 1 LEVEL        L-RC45?                 100.000
     L0481A PRESSURIZER 2 LEVEL        L-RC46                   34.835
     L0482A PRESSURIZER 3  LEVEL       L-RC46                   34.835
     K0011  PRESSURIZER LEVEL CLUSTER LIM                        5.000
 1319  RETRN CI  Y0103D  CI  BLEND TO CHRG PMP VLV FCV-CH113BCL
 1320  ALARM HI  F0128A  AI  CHARG PMP DISCH HDR F     F-CH122   150.1  H   150.0  GPM
 1321  ALARM LO  L0112A  AI  VOLUME CONTROL TK LEVEL   L-CH115   15.1   L   16.0   PC
 1322  ALARM CI  Y0103D  CI  BLEND TO CHRG PMP VLV FCV-CH113BNOT CL
 1322  RETRN CI  Y0103D  CI  BLEND TO CHRG PMP VLV FCV-CH113BCL
 1322  RETRN LO  P0139A  AI  VOLUME CONTROL TANK P     P-CH117   24.6   L   20.0   PSIG
 1322  RETRN LO  L0112A  AI  VOLUME CONTROL TK LEVEL   L-CH115   17.7   L   16.0   PC
 1322  RETRN CI  P0493D  CI  PRESUZER LO P 2 SI TR PART BLOCKRESET
 1322  RETRN CI  P0492D  CI  PRESUZER LO P 1 SI TR PART BLOCKRESET
 1322  RETRN CI  P0494D  CI  PRESUZER LO P 3 SI TR PART BLOCKRESET
 1323  ALARM LO  L0482A  AI  PRESSURIZER 3  LEVEL      L-RC461   1?.?   L   12.0   PC
 1323  ALARM LO  L0481A  AI  PRESSURIZER 2 LEVEL       L-RC460   1?.?   L   12.0   PC
 1323  ALARM CI  Y0007D  CI  REACTOR TRIP MAIN BKR B    RC RTBTRIP
 1323  ALARM CI  Y0390D  CI  TB TRIP CAUSE RE                    TR
 1323  ALARM CI  P0397D  CI  TB HYD OIL LO P 2 PART RE           TR
 1323  ALARM CI  Y0394D  CI  TB STOP V D CL PART RE LM-TB???ATR
 1323  ALARM CI  P0486D  CI  PRESSURIZER LO P 3 PART RE          TR
 1323  ALARM CI  P0484D  CI  PRESSURIZER LO P 1 PART RE          TR
 1323  RETRN CI  Y0722D  CI  AUTO STOP LATCH                     TR
 1323  ALARM CI  P0398D  CI  TB HYD OIL LO P 3 PART RE           TR
 1323  ALARM CI  Y0006D  CI  REACTOR TRIP MAIN BKR A    RC RTATRIP
 1324  RETRN HI  P2010A  AI  EXTR STM TO REHTR TB P    P-MS103B  46.8   H   300.0  PSIG
 1324  ALARM LO  T0453A  AI  RCPC MOT UPR RDL BRG T T-RC438A2    99.6   L   100.0  DEGF
 1324  ALARM LO  T0433A  AI  RCPB MOT UPR RDL BRG T T-RC428A2    99.6   L   100.0  DEGF
 1324  ALARM LO  T0455A  AI  RCPC MOT LWR RDL BRG T T-RC438?C    99.6   L   100.0  DEGF
 1324  ALARM LO  P2001A  AI  LP TURBINE 1 STM IN P   P-MS105A   -14.1   L   -1.0   PSIG
 1324  ALARM LO  U0523   CV  RCL ROD MAX DT DEV FR MEAS          2.9        3.0    DEGF
 1325  ALARM CI  Y0003D  CI  NUCLEAR & TB PWR RE TR P7 PERM  SET
 1325  RETRN CI  Y0001D  CI  TB PWR 1 RE TR PART PERM          RESET
 1325  RETRN CI  Y0002D  CI  TB PWR 2 RE TR PART PERM          RESET
 1325  RETRN CI  P0488D  CI  PRESSURIZER LO P CAUS RE          NOT TR
 1325  ALARM CI  Y0103D  CI  BLEND TO CHRG PMP VLV FCV-CH113BNOT CL
```

pg 9A

pg 9A

ES-302-11
Attach. 1/4
Comment 3

note
the short
time interval



Pg 9B

ES-302-11, Attachment 2/4, comment 1

This comment is accurate. It documents a significant error. It alone should not be sufficient to produce a failing grade for the following reasons:

1. We always want to avoid a loss of forced flow cooling. When necessary, however, natural circulation is adequate for removal of decay heat. In this instance, I did establish and maintain adequate natural circulation. As the scenario progressed and the fast pace slowed down, I began reviewing my instrumentation recordings. I discovered on my own that I had tripped the pumps prematurely and was about to announce that fact to the SRO stand-in with a request to re-start at least one pump, but the scenario ended at that time. I then informed the examiner that it appeared I had tripped the pumps prematurely. We then went into a discussion of core cooling mechanisms during which I informed the examiner that we never wanted to lose forced cooling flow if it could be avoided but that, as demonstrated in this scenario, we had the ability to cool the core with natural circulation flow. Also, it must be realized that this is another instance where I did not have the extra help from the SRO that the RO's whom I was tested with would have had from me. It should be noted, and can be verified by Mr. Norris since he watched intently, that while I performed as an SRO I had reactor coolant system data trending on the computer screen and constantly verified the accuracy of information being relayed back to me by the operators. In a case like this, I, as SRO, could have warned the operator to take a second look at his pressure readings before tripping the pumps.

ES-302-11, Attachment 2/4, Comment 2

This comment is invalid. There are no position indicating lights for the Residual Heat Release valve. I was not confused during the performance of that step. I was hesitant to verify the valve closed based on observation of the demand signal alone because we are cautioned against doing that during training. The plant operator concurred with me that the demand indication was the only way to verify valve position from the control room. I then responded appropriately to the SRO that the valve was closed. This is an example of good crew interaction and teamwork. If this comment was allowed to stand, it would hinder the training on crew interaction in the future.

ES-302-11, Attachment 2/4, Comment 3

The examiner has drawn a wrong conclusion. Why didn't he ask about this after the scenario? The other operator did not have to show me where the containment sump pump switch was located. He saw that I had turned the wrong switch in my haste to perform the step and called that to my attention. He then pointed to the correct switch but had he not I would still have quickly located the correct switch. The examiners should realize that each candidate is trying to impress the examiners and in so doing, if they work together to correct errors on their own, that teamwork should reduce the seriousness of any comment like that referenced above.

ES-302-11, Attachment 2/4, Comment 4

This comment should be reconsidered because it is not entirely correct. there is no way to verify CIA reset from the control room other than attempting to cycle the CIA valves. When the scenario ended, the examiner asked how to get RCS samples. After some confusion was cleared regarding the thrust of his question, I explained that the CIA sample valves would have to be opened. He asked me to open them. I opened the train B valves first, and they came open. I then tried the train A valves; they did not open. I immediately realized that train A of CIA had not been reset. I reached over in front of the examiner, pushed the train A reset button and then opened the train A sample valves. This was performed without prodding from the examiner. This is exactly how a failure to reset CIA in the plant would be detected and corrected.

ES-302-11, Attachment 3/4, Comment 1

This comment is erroneous. As the SRO, I was the first to notice the stuck open valve. I called it to the PO's attention. She also observed it and initiated corrective action to close the valve. This is verified by the fact that the RO had no way of knowing that valve was stuck open unless he had heard me and the PO discussing it since he was busy reacting to his control board. Yet, when questioned by his examiner, he knew the valve was stuck open and he acknowledges that he heard my discussion of the problem. This is a good example of crew communication and interaction.

ES-302-11, Attachment 3/4, Comment 2

On the basis of the preceding statements, this comment is unsupported.

ES-302-11, Attachment 4/4, Comment 1

This comment should be reconsidered. During discussion after the scenario, I informed the examiner that I must have mistakenly read the DF bus as de-energized and I was astonished as to how I could have done that. Later consideration of the event leads to the conclusion that I had not misread the DF bus voltmeter. In fact, the DF bus was de-energized when I looked at it. The diesel generator had not yet loaded on the bus. I correctly directed the performance of ECA 0.0 for the indications that I saw at that time. During performance of the ECA-0.0, step 8, I asked the operator to verify the emergency buses energized, which he did and responded that DF bus was energized. I was surprised but, without getting flustered, properly directed transition to step 1 of E-0.

ES-302-11, Attachment 4/4, Comment 2

This comment is not true. I never admitted mistaking the operator's report. I questioned the operator, out loud about how much variation from normal the radiation detector showed. When she held her fingers a small amount open, I said aloud that that was insufficient variation to support transition to E-3. I asked aloud whether there were any other indications of a tube rupture such as erratic steam generator levels. When the operator responded that there were no other indications, I said out loud we are not going to transition to E-3. We will stay in E-1. That was the correct procedure to be in because we had a loss of reactor coolant occurring not a tube rupture. Since TMI, operators have been taught not to rely solely on one instrument, but to verify or discount that information based on redundant readouts or corresponding parameters that monitor the same information. I cannot imagine why the examiner would suggest that I should have been in E-3 when he knew his scenario did not call for a tube rupture. In discussion after the scenario, I showed the examiner the small variation on the radiation detector and explained again that in the absence of other radiation detector variations and with no other indications of a tube rupture, it was not proper to transition to E-3. I then asked the examiner if his scenario called for a tube rupture. He said "no".

Instead of being criticized for my actions, I should be congratulated on having the supervisory fortitude to discount an operators response and proceed along the correct path.

ES-302-11, Attachment 4/4, Comment 3

Please see my response to ES-302-11, Attachment 1/4, Comment 2. In addition, note that the same comment is used to support two different areas of poor performance in the mind of the examiner. It does not support either. The procedure was carried out correctly and it was carried out based on the operators report.

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

MAR 24 1987

Mr. Alfred J. Morabito
685 Tulip Drive
New Brighton, PA 15066

IN RESPONSE REFER
TO FOIA-87-151

Dear Mr. Morabito:

This is in response to your letter dated March 10, 1987, in which you requested, pursuant to the Freedom of Information Act, copies of the originals of your July 23, 1986 examination notes and log sheets and a copy of NRC's letter to Mrs. Susan Neuder.

Enclosed are copies of the requested notes and log sheets. However, we cannot provide a copy of the requested letter at this time.

Although you state that you have express permission from Mrs. Susan Neuder to receive a copy of the requested letter, in order for the NRC to release a record containing a home address and other personal information of a third person, it would be necessary for you to provide to this office written, notarized authorization for us to provide such information to you.

If you do not provide such authorization from Mrs. Neuder to this office by March 31, 1987, we will assume that you have no further interest in this matter and will close our file on your request.

Sincerely,

Donnie H. Grimsley

Donnie H. Grimsley, Director
Division of Rules and Records
Office of Administration

Enclosures: As stated

APPENDIX

FOIA REQUEST NUMBER 87-151

1.	Undated	Shift Foreman Shift Turnover Checklist	(1 Page)
2.	7/23/86	Log Sheet	(2 Pages)
3.	7/23/86	Nuclear Control Operators Report	(1 Page)
4.	7/23/86 (Date Initialed)	Beaver Valley Power Station, Unit 1 Operating Surveillance Test	(8 Pages)

Note: These are the originals of the records as requested

194 | 03, 001,013 | 1

	0800	1600	2400
1. On coming shift foremen conduct a walkdown of the control area observing evolutions in progress, unusual equipment or systems status and any unusual alarms or indications or status lights.	☐	☐	☐
Review previous shift logs.	☐	☐	☐
3. Discuss any identified trends that require special attention.	☐	☐	☐
4. Discuss any temporary logs, parameter trending or temporary procedures in effect.	☐	☐	☐
5. Review any MSP's, OST's, BVT's, PMP's (or other procedures effecting control room indications) in progress (By Offgoing).	☐	☐	☐
6. Discuss any gas or liquid discharges in progress.	☐	☐	☐
7. List on reverse any technical specification action items in effect (by Offgoing). Review list (oncoming).	☐ ☐	☐ ☐	☐ ☐
8. Verify normal shift complement and emergency squad requirements are met (Oncoming).	☐	☐	☐
9. Review the Station Equipment Status Board (Offgoing) (Oncoming).	☐ ☐	☐ ☐	☐ ☐
10. Review the ESF Status Panel for O.O.S. ESF Systems (Oncoming)*.	☐	☐	☐
11. Review the ESF Mimic Print for out of normal conditions (Oncoming)*.	☐	☐	☐
12. Discuss status of procedures in progress, being turned over.	☐	☐	☐
In modes 5 & 6, current OST 1.48.1 determines Train [A or B] (circle one) is the priority train.	☐	☐	☐
14. Key Inventory performed (L10-2).	☐	☐	☐
15. Check to assure all recorders are working properly.	☐	☐	☐
16. The Nuclear Control operators narrative log; S1-4, 5, 6; has been reviewed to assure all significant items have been listed with adequate description.	☐	☐	☐
17. Review OST's schedule for performance oncoming shift.	☐	☐	☐
18. On coming and offgoing foremen conduct a review of control boards and control panels looking for deviations from normal and the general status of equipment.	☐	☐	☐
19. Oncoming foreman, log 5 qualified fire brigade members on shift in foreman's log book. (Use the fire brigade qualification log supplied by OPS Support Group.) Notify each individual selected and his/her immediate supervisor of their status as a member.	☐	☐	☐
20. Identify "Off-Log Only" clearances that need reported to maintenance for sign off each morning.	☐	☐	☐

	(Offgoing)	(Oncoming)
0800	________ (Signature)	________ (Signature)
1600	________ (Signature)	________ (Signature)
2400	________ (Signature)	________ (Signature)

7/23/86

0841 MOL, 480 PPM B, Eq. Xe, stay at 100%
0852 Emergency Squad identified - 5 qualified members. Three members of shift complement identified for safe shutdown.
0909 Anomalous indications on recorders (spikes) Thermal barrier isolations
0915 Stopped loss of condenser by placing steady flow of seal water to vacuum breaker
0919 No entry
0931 Controlling pressurizer spray + level in manual. Investigating vital bus indications
0935 Requested I+C support on vital bus anomaly
0939 Requested AA to prepare MWR for investigation and repair of vacuum breaker seal water.
0946 I+C reports no obvious or de
0959

7/23/86

1125 Emergency squad identified (5 qualified members). Morabito, Sheib, Nendler responsible for safe shutdown.

1137 Requested I+C trouble shoot of Channel 4 feed flow indication for S/G B due to spike on recorder.

1142 Manual control on B S/G feedwater while I+C

1146 Failure of loop 3 Control ΔT + Tavg. Requested I+C to investigate output of control TH for loop 3

1137- BOP operator notices spike on S/G B level and FW. Informs me and I request I+C investigation because spiking similar to spiking in 1st scenario.

1142- BOP requests to take manual control of B S/G level due to continued increase. I gave permission.

1142- As indicated by incomplete sentence in my log entry, my attention is diverted to Tavg/ΔT failure. B S/G level is under control.

S1-4

NUCLEAR CONTROL OPERATORS REPORT

DATE 7/25/86
0000-0800

Plant Status - 0001

Operating Mode 1 Boron Concentration 480 PPM
Generator Gross MW 820 Reactor Power 100 (%, AMP, CPS)
Controlling Rod Bank Grp D Height 222 STEPS
RCS Pressure 2220 PSIG Tavg 576 F

TIME	
1305	Integrator BA 250717, PG water 174962
1307	Failure of NIS 44. Rod control is manual
1316	Removed 45 gpm orifice from serv. Press. lvl. falling
1318	LT-459 failed lo

Relieved by A.J. Morabito at 1305
Oncoming RO

_____________ RO (0000-0800)

Controlled keys transferred to and
Relieved by ____________ at ________
Oncoming PO

_____________ PO (0000-0800)

_____________ SOF Review (0000-0800)

CR CONT COPY

BEAVER VALLEY POWER STATION
UNIT NO. 1
OPERATING SURVEILLANCE TEST

SURVEILLANCE TEST NO: OST 1.49.2	TECHNICAL SPECIFICATION: 4.1.1.1.1.a, c, & e; 4.1.1.2.a & b	FREQUENCY: D
TITLE: SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6)		
OPERATING MODE REQUIRING TEST: 2*,3,4,5	PERFORMED BY:	DATE COMPLETED:

TEST RESULTS (To be completed by test performer)

☐ Test completed SATISFACTORY TOTAL MAN HOURS ______

☐ The following problems were encountered:

______

______

☐ Corrective action taken or initiated:

______

______

☐ MWR NO. (If Written) ______

TEST REVIEWED BY SHIFT SUPERVISOR ______ DATE ______

☐ Procedure properly completed and satisfactory

☐ Comments: ______

______

TEST REVIEWED BY OPERATING SUPERVISOR ______ DATE ______

☐ SATISFACTORY AND APPROVED

☐ Comments: ______

______

*When in Mode 2, # at least once during rod withdrawal and at least once per hour thereafter until the reactor is critical. # with Keff < 1.0.

SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6)

PURPOSE

To determine the minimum shutdown margin when the reactor is subcritical and either:

a. Within one hour after detection of an inoperable rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable; or

b. When in Mode 2 # at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical; or

c. Once per 24 hours when in Mode 3 or 4.

ACCEPTANCE CRITERIA

1. For Modes 2 #, 3 and 4, the shutdown margin shall be ≥ 1.77% ΔK/K by borating to a RCS boron concentration equal to or greater than the Minimum Shutdown Margin Boron Requirement.

2. For Mode 5, the shutdown margin shall be ≥ 1.0 ΔK/K by borating to a RCS boron concentration equal to or greater than the Minimum Shutdown Margin Boron Requirement.

3. For Mode 5, when the RCS is drained down, the shutdown margin shall be ≥ 5% ΔK/K by borating to an RCS boron concentration equal to or greater than 1416 ppm.

NOTE: If any control rods are declared inoperable due to being immovable or untrippable, the above required shutdown margin boron concentration shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rods. The shutdown margin shall be determined to be greater than this required margin within 1 hour after detection of the inoperable control rods and at least once per 12 hours thereafter while the rods are inoperable.

INITIAL CONDITIONS	Initial/Date
1. The reactor is subcritical.	7/23
2. The operator(s) performing this test have reviewed this procedure.	7/23
	N/A /

With Keff < 1.0

SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

PRECAUTIONS

1. With the boron concentration less than the minimum required to maintain the reactor shutdown margin of ≥ 1.77% ΔK/K or ≥ 1.0% ΔK/K, immediately initiate and continue boration at ≥ 30 gpm of 7,000 ppm boric acid solution or its equivalent until the required shutdown margin boron concentration is restored.

INSTRUCTIONS

Initial/Date

Date and time calculation applicable. ______/______

1. Minimum Boron Requirement.

NOTE: Figures 1 and 2 have been produced with the reactivity effects of RCS boron concentration, control rod position, RCS average temperature, fuel burnup, Xenon and Samarium concentrations taken into consideration as required by Technical Specifications 4.1.1.1 and 4.1.1.2.

a. Core age (Reactor Engineering Data Book) = 0 EFPD.

b. Present boron concentration = 2147 ppm.

c. Minimum Shutdown Margin Boron Requirement (No Xenon).

1) Mode 3 - Use Figure 1.

--Hot Zero Power, Tavg = 547 ± 2°F Use 547°F line.
--350°F ≤ Tavg < 545°F Use 350°F line.

2) Mode 4 - Use Figure 1.

--200°F < Tavg < 350°F Use 200°F line.

3) Mode 5 - Use Figure 2. RCS T= 94.8°

--140°F ≤ Tavg ≤ 200°F Use 140°F line.
--70°F ≤ Tavg < 140°F Use 70°F line.

4) Mode 5 RCS drained down RCS C_B ≥ 1416 ppm
= ~~2147~~ ppm ppm (1c)
assume 1416

SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

Initial/Date

2. Xenon Correction. ______/___

   a. Reactivity insertion due to Xenon from Figure 3 = - 0 pcm.

   b. Differential boron worth from Figure 4 = - -8.65 pcm/ppm.

3. Equivalent Boron Worth of Xenon. ______/___

   (2a) 0 pcm ÷ (2b) -8.65 pcm/ppm = 0 ppm.

4. Minimum Shutdown Margin Boron Requirement Corrected for Xenon. ______/___

   (1c) ~~2147 ppm~~ 1416 ppm - (3) 0 ppm = 1416 ppm.

5. Boron Concentration Comparison. ______/___

Present RCS Concentration	≥	Required Boron Concentration
(1b) 2147 ppm		(4) 1416 ppm

Acceptance Criteria: Present RCS boron concentration must be equal to or greater than the required boron concentration.

6. With the boron concentration less than the minimum required to maintain the reactor shutdown margin of ≥ 1.77% ΔK/K or ≥ 1.0% ΔK/K, immediately initiate and continue boration at ≥ 30 gpm of 7,000 ppm boric acid solution or its equivalent until the required shutdown margin boron concentration is restored. ______/___

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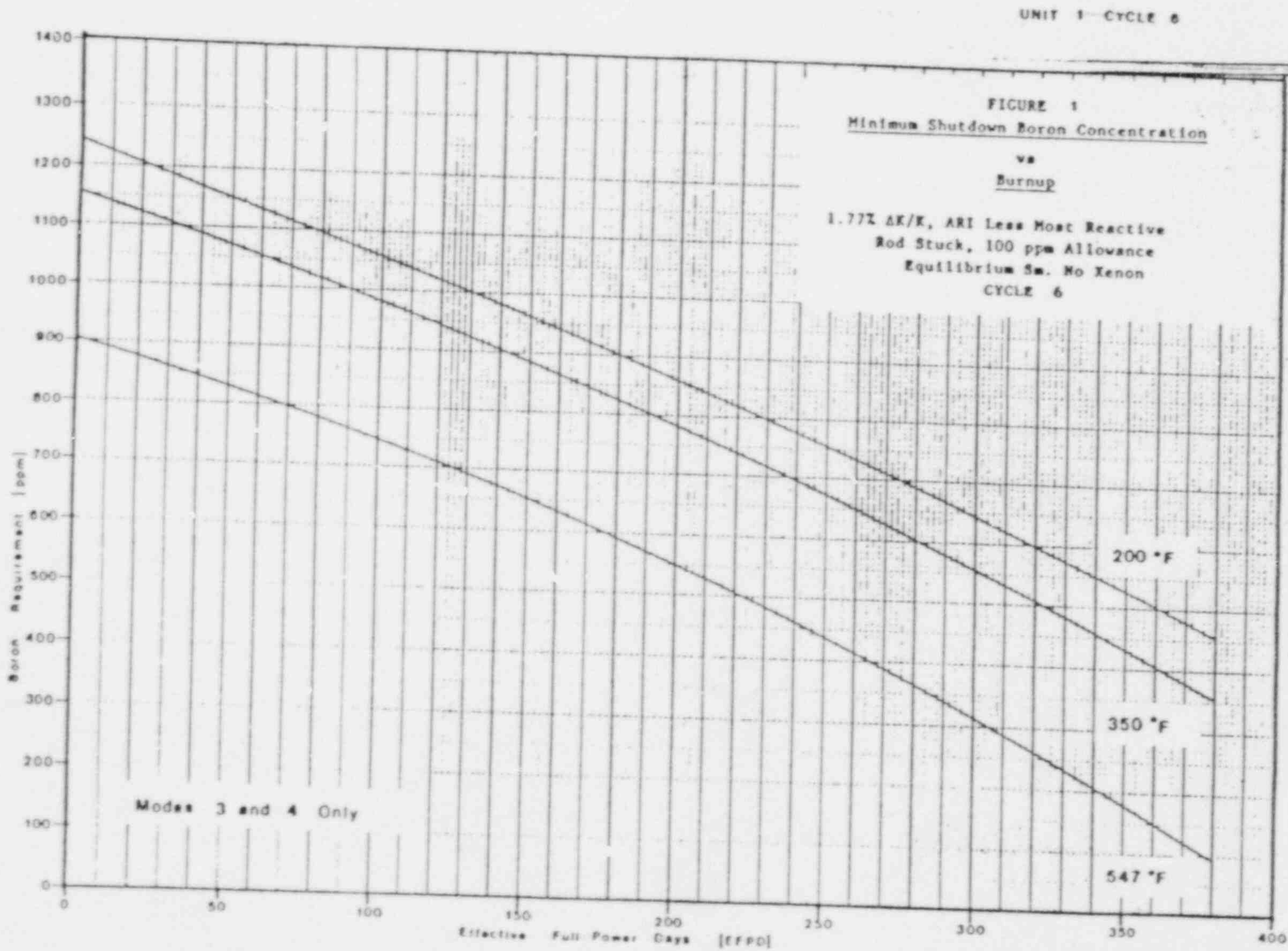
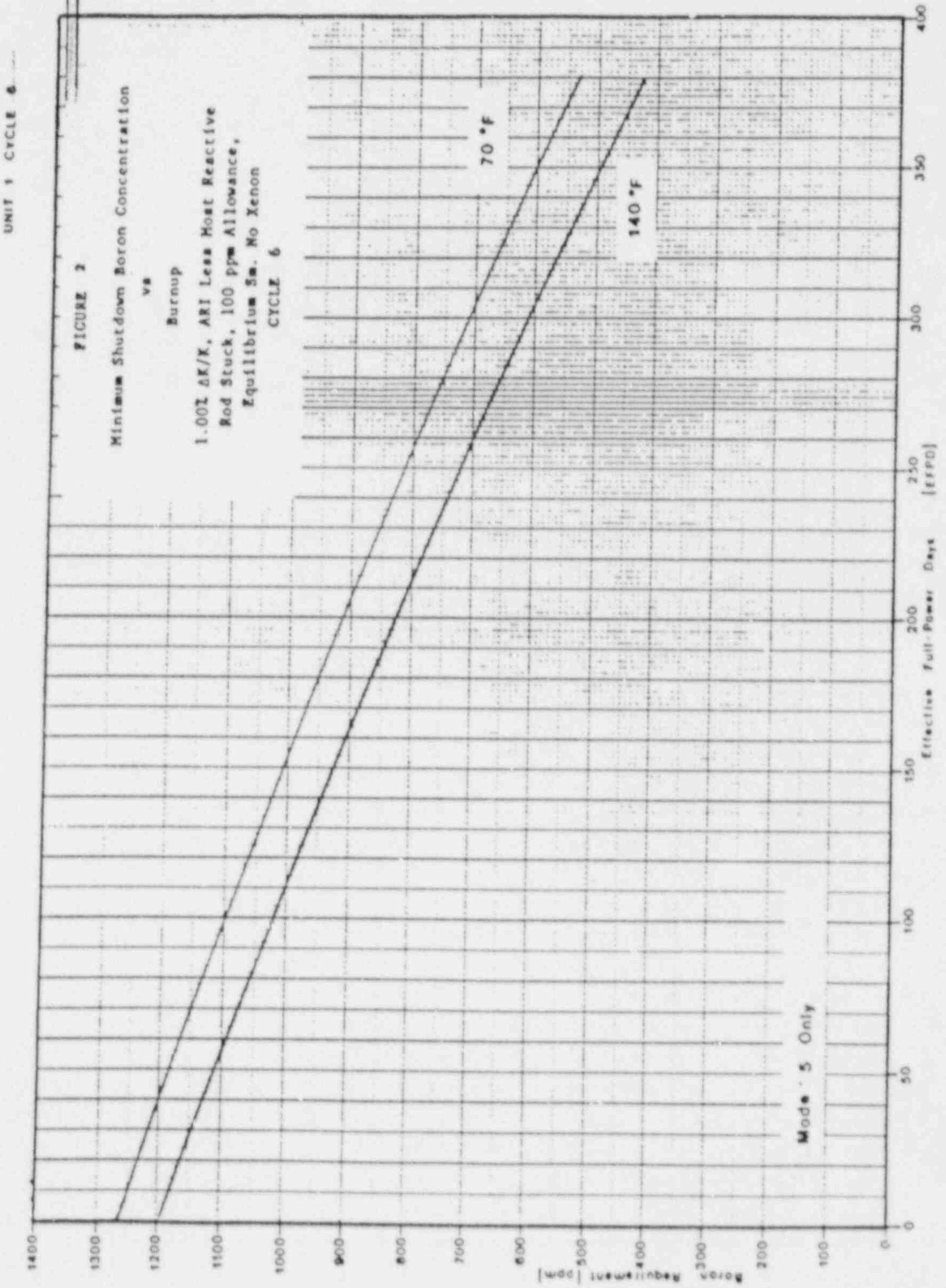
SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

FIGURE 1



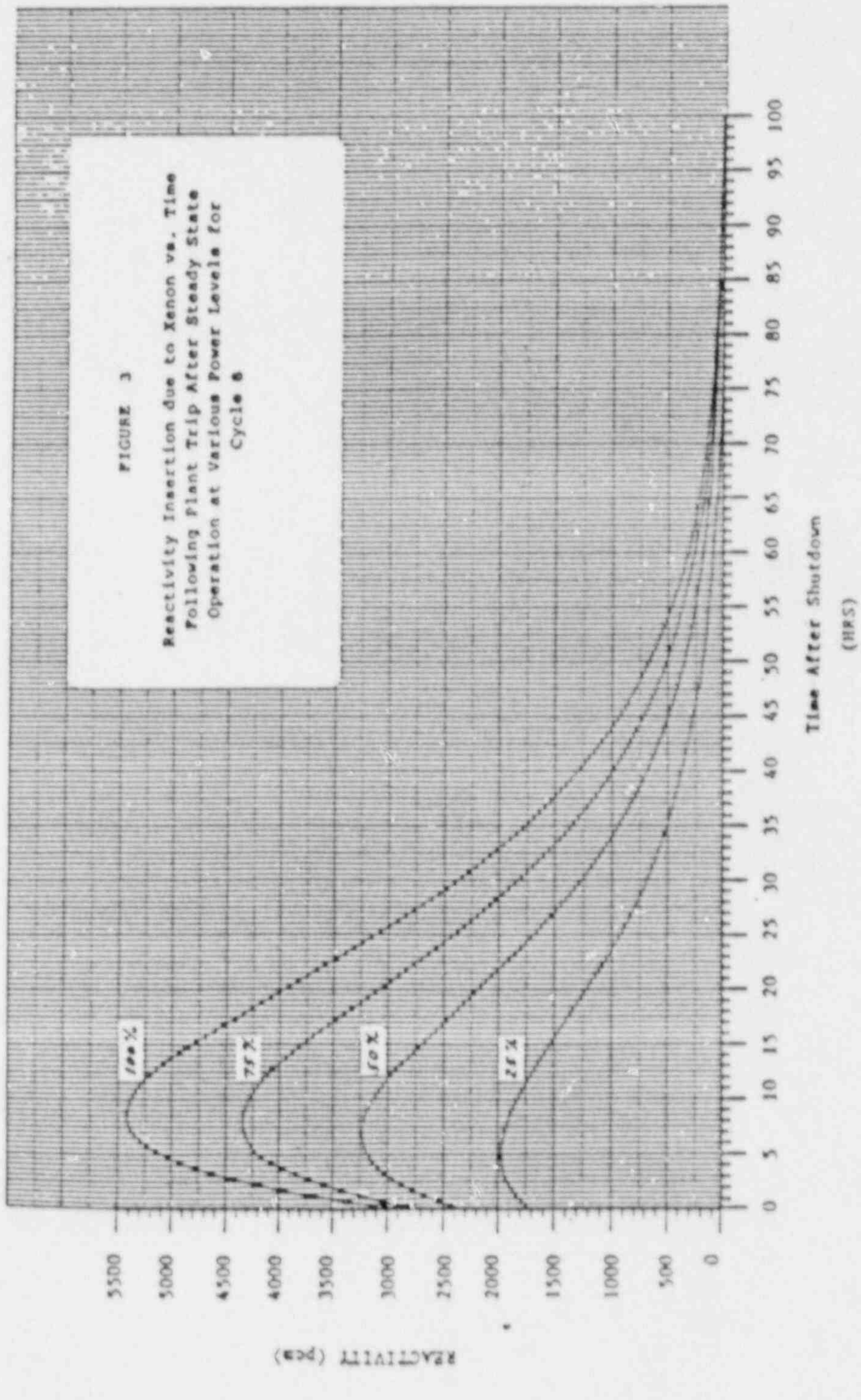
SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

FIGURE 2



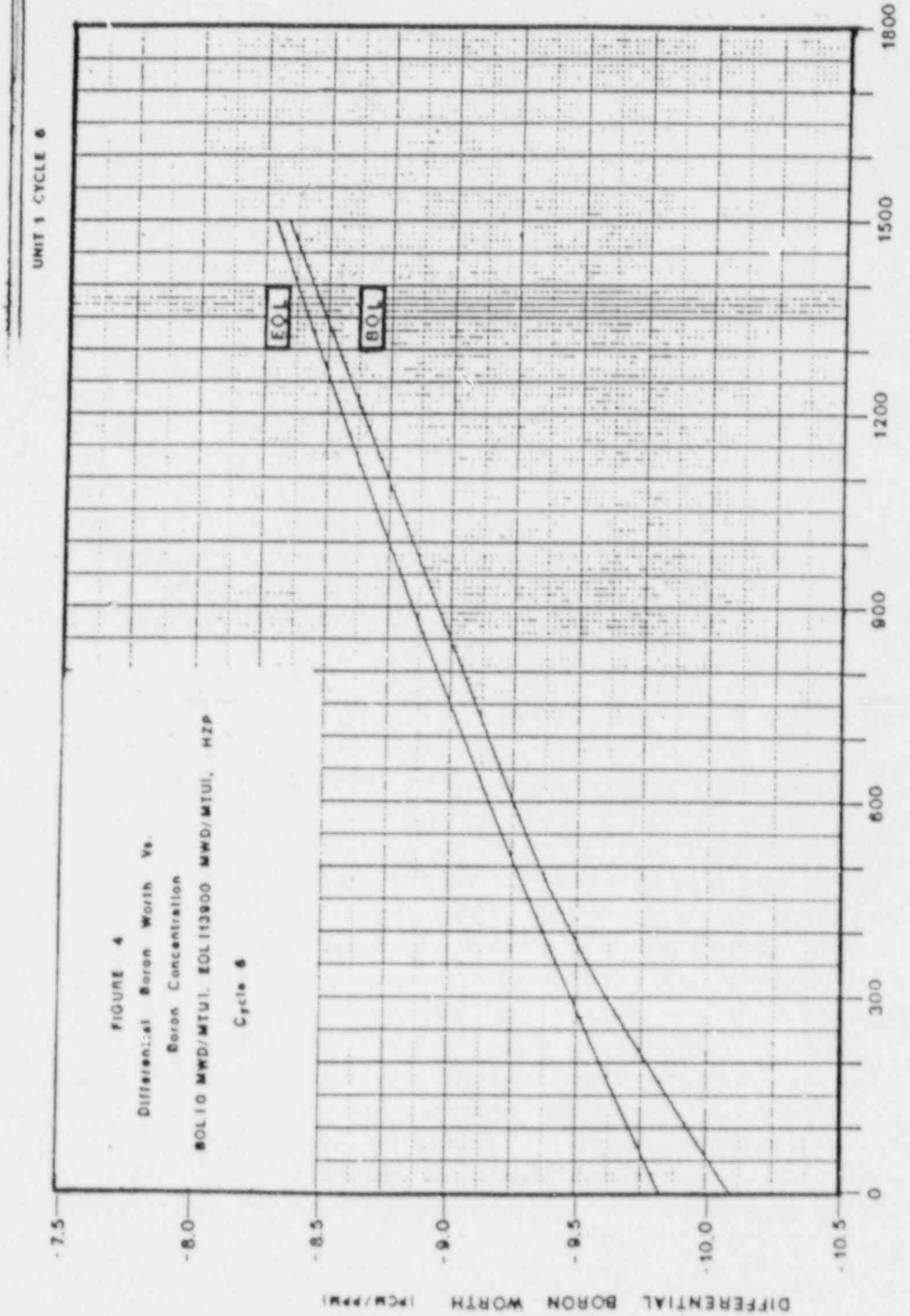
SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

FIGURE 3



SHUTDOWN MARGIN CALCULATION (PLANT SHUTDOWN) (UPDATED FOR CYCLE 6) (continued)

FIGURE 4



STEP DESCRIPTION TABLE FOR E-3 STEP 9

STEP

Isolate CNMT Vents And Drains System

PURPOSE

To instruct the operator to secure the containment vents and drains system prior to resetting SI.

BASIS

This precautionary step is intended to minimize the possibility of containment integrity being inadvertently compromised. When SI and CIA/CIB are reset in subsequent steps, no valve positions or equipment status should change; however to provide redundant protection, systems that could provide a direct path from containment are manually isolated.

ACTIONS

•Stop CNMT sump pumps [DA-P-4A, 4B]
•Pull-To-Lock primary drain tank transfer pumps No. 1 [DG-P-1A, 1B]
•Pull-To-Lock CNMT vacuum pumps [CV-P-1A, 1B]
•Close Non-areated vents header isolation valves [TV-DG-109A1, A2]

INSTRUMENTATION

•Pump status lights
•Valve control switches

CONTROL/EQUIPMENT

•Pump control switches
•Valve control switches

KNOWLEDGE

N/A

APPROVED, REV. 0

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

August 9, 1982
DLS:AJM:151

Fork Lift Truck Safety

TO: Distribution

It is my intention to react vigorously to the recent warning signals that have occurred involving fork lift trucks. Most of the defueling equipment being handled with fork lifts is a one-of-a-kind item. Damage to such items could delay the defueling effort. It is our responsibility to ensure that our handling equipment, training procedures, and operating techniques are of a quality consistent with the requirements of the LWBR project. I have talked to most of the maintenance personnel who are likely to be operating the fork lift trucks. I have asked that they make a greater effort to operate the equipment with care. In addition, I am directing that the following actions be taken:

1. The Safety Administrator is to obtain additional training material concerning the handling of material with fork lifts.

2. Maintenance supervision will provide the additional training to their personnel.

3. Maintenance supervision will have paths cut through the speed control humps in the parking lot to provide smooth operating surfaces for fork lift trucks.

4. Manchester maintenance personnel will be scheduled to inspect, and repair if necessary, the station's fork lift trucks.

5. Operators of fork lifts must tie down all loads to be transported from the warehouse to the station. If a load cannot be tied down, supervision must concur in moving the load.

6. The use of double fork lifts to transport long loads is prohibited. Any load which is too long for one fork lift will be handled by crane and truck.

7. In the interim, while additional training material is being developed, an Engineer will be assigned on a daily basis to aid fork lift operators in securing loads for transport. It is not expected that the Engineer will add any particular expertise to the loading of a fork lift but it is expected that with the operator's practical experience in handling fork lifts and the Engineer's theoretical knowledge of torques, bending moments and lever action that safe loading and handling techniques can be achieved.

8. The Joint Defueling Group (JDG) should resolve their existing open items concerning material handling.

A. J. Morabito

A. J. Morabito
Chief Engineer

cc: A. J. Morabito
V. F. Kraker
W. E. Strayhorn
R. G. Williams
J. M. Crum
C. K. Schultz
L. R. Freeland
D. R. Henderson
M. G. Haydin
F. R. Huey
T. McGourty (JDG)
Central File

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

November 1, 1982
DLS:AJM:154

Flush of Demineralizer Resin

Mr. . R. Huey, Manager
Shippingport Branch Office
U. S. Department of Energy
Shippingport, PA 15077

Dear Mr. Huey:

The flush of the AC Purification System demineralizer resin which was scheduled for today, November 1, 1982 has been placed on hold by Duquesne station management. There is a concern that during the next two months of Reactor Coolant System layup operation with borated water and decreasing pH conditions a potential exists for increasing the mobility of crud currently trapped in the Reactor Coolant System and in the LWBR core. If the resin in AC purification were discharged as scheduled, there would be no mechanism available to remove the mobilized crud. This would lead to increased radiation levels throughout the reactor plant. Future maintenance and defueling preparations in the reactor plant containers would become more cumbersome because of the increased radiation exposure controls needed for work in those areas.

Duquesne requests Naval Reactors and Bettis to assess the following questions.

1. Should the reactor plant be left with no resin in either purification demineralizer for the next two months?

2. Can the existing resin in the AC Purification System be placed in service in a borated water environment without eluting fission products from the resin?

3. With the extended service time already on the AC purification resin, is there sufficient filtering capacity available to enable that resin to clean up crud bursts?

4. Should the AC purification resin be discharged to support decay heat testing requirements but then schedule replacement of the resin with new resin immediately after completion of the test measurements?

   a) An option to the above would be to discharge the resin and be prepared to reinstall new resin only if crud bursts occur. Duquesne does not favor this option because of the possibility that personnel replacing the resin could be exposed to significantly increased levels of radiation from mobilized crud while replacing the resin.

The Shippingport Branch Office is requested to forward these concerns to Naval Reactors and Bettis for technical evaluation.

Very truly yours,

A.J. Morabito

A. J. Morabito
Chief Engineer

xc: Addressee (4)
J. F. Zagorski
F. X. Bayer (2)
L. R. Freeland
C. K. Schultz
M. G. Haydin
R. G. Williams
AJM File
Central File

DUQUESNE LIGHT COMPANY
Shippingport Atomic Power Station

December 9, 1982
DLS:AJM:159

Controlled Area Work Stoppage

Mr. J. F. Zagorski:

At 1410 hours today, I directed that all work in controlled areas, other than operations related activities, be stopped. This was done to give station management a chance to step back and look at a series of problems that have occurred during the last few days and take action to correct them before they develop into major problems. The problems have been identified as:

1. Use of a vacuum cleaner having an expired DOP certification for radioactive material control.

2. The occurrence of an incident involving an unlocked, unguarded high radiation area.

3. Improper release of radioactive material from the controlled area.

4. Personnel in the controlled area without proper monitoring devices.

5. Failure of personnel to properly control a high radiation area door key.

6. Loss of security control at the reactor pit.

7. Mismatched pins and yokes on load tested shackles.

8. Loss of defueling sequence control.

Problems 1 through 5 are related to radiation worker training and responsibilities. The corrective action is to require personnel to attend a lecture tailored to cover these noted deficiencies, take a written examination and be counseled on incorrect answers prior to restarting work in the controlled area.

Problem 6 will be corrected by installing a security lock on the entrance door to the reactor pit. A security guard, stationed in the canal, will unlock the door to permit authorized personnel to enter the area. A log will be kept of personnel entering and exiting the area. When personnel are in the area, the security lock will be left unlocked and the guard will remain on guard at the door. When no personnel are in the area, the guard will be permitted to lock the door and rove about the canal area.

Problem 7 will be corrected through improvements in the load test data sheet and education of personnel to be alert for incompatible identification numbers on pins and yokes.

Problem 8 has been addressed and controlled through the Joint Defueling group. It is included in this letter because it adds to the list of problems that have recently been encountered.

On the basis of the above actions being completed prior to resumption of work on each of the shifts, 4-12 on 12/9/82, 00-08 on 12/10/82 and 08-16 on 12/10/82, I am releasing the stop work directive.

A. J. Morabito

A. J. Morabito

cc: T. D. Jones
V. F. Kraker
W. E. Strayhorn
R. G. Williams
J. M. Crum
C. K. Schultz
L. R. Freeland
D. M. DiNuzzo
M. G. Haydin
G. T. Howard
R. J. Massimino
AJM File
Central File

EXHIBIT Y

To whom it may concern:

There is no doubt in my mind that the communications between Al Moribato and myself in reference to the radiation monitor reading during the second scinario were plainly understood.

Susan M. Neuder

Susan M. Neuder

Sworn and subscribed before me this 27th day of July 1987.

Debbie L. McCarthy

Notary public

DEBORAH L. McCARTHY, NOTARY PUBLIC
MOON TOWNSHIP, [illegible]
MY COMMISSION EXPIRES DEC. [illegible]
Member, Pennsylvania Association of Notaries

EXHIBIT Z

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

SEP 23 1986

Docket No. 50-334

Duquesne Light Company
ATTN: Mr. J. J. Carey
Vice President
Nuclear Group
Post Office Box 4
Shippingport, Pennsylvania 15077

Gentlemen:

SUBJECT: EXAMINATION REPORT NO. 50-334/86-16(OL)

During the weeks of July 21 and July 28, 1986, the NRC administered examinations to employees of your company who had applied for licenses to operate your Beaver Valley Power Station Unit 1. At the conclusion of the examination, the examination questions and preliminary findings were discussed with those members of your staff identified in the enclosed report.

In accordance with 10 CFR 2.790 of the Commissions regulations, a copy of this letter and the enclosures will be placed in the NRC Public Document Room.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Ebe McCabe, Acting for
Edward C. Wenzinger, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosure:
Examination Report No. 50-334/86-16(OL) w/Attachments 1, 2, 3, 4

cc: w/enclosure and Attachments 1, 2, 3, 4
W. S. Lacey, Plant Manager
T. Burns, Nuclear Training Director
W. Troskoski, Senior Resident Inspector
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Pennsylvania

cc: w/enclosure without Attachments 1, 2, 3, 4
H. M. Siegel, Manager, Nuclear Engineering Department
C. E. Ewing, QA Manager
R. Druga, Chief Engineer
R. Martin, Nuclear Engineer
J. Sieber, Manager, Nuclear Safety and Licensing
T. D. Jones, Manager, Nuclear Operations
N. R. Tonet, Manager, Nuclear Engineering

U. S. NUCLEAR REGULATORY COMMISSION REGION I
OPERATOR LICENSING EXAMINATION REPORT

EXAMINATION REPORT NO. 50-334/86-16(OL)

FACILITY DOCKET NO. 50-334

FACILITY LICENSE NO. DPR-66

LICENSEE: Duquesne Light Company
Post Office Box 4
Shippingport, Pennsylvania 15077

FACILITY: Beaver Valley Unit 1

EXAMINATION DATES: July 22-31, 1986

CHIEF EXAMINER: G. S. Barber, Reactor Engineer (Examiner) 9/3/86 Date

REVIEWED BY: R. M. Keller, Chief, Projects Section 1C 9/3/86 Date

APPROVED BY: Harry B. Kister, Chief, Projects Branch No. 1 9/15/86 Date

SUMMARY: Oral, written and simulator examinations were administered to twelve reactor operator and three senior reactor operator and one senior reactor operator retake candidate. In addition, just the oral and simulator portion was administered to one senior reactor operator retake candidate. Seven reactor operators passed all portions of their examinations and will be issued licenses. Both senior reactor operator retake candidates passed all required portions of their exams and will be issued licenses. Of the three remaining senior reactor operator candidates, one passed all portions of the exam and will be issued a license. The specific details of the candidates that failed all or portions of their exam can be found in the examination results table on the following page.

REPORT DETAILS

TYPE OF EXAMS: Initial ___ Replacement X Requalification ___

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	10 / 2	4 / 1
Oral Exam	11 / 1	5 / 0
Simulator Exam	9 / 3	3 / 2
Overall	7 / 5	3 / 2

I. CHIEF EXAMINER AT SITE: G. S. Barber, NRC

II. OTHER EXAMINERS: R. M. Keller, NRC
N. F. Dudley, NRC
B. S. Norris, NRC
D. M. Silk, NRC
B. Gruel, PNL
L. Defferding, PNL

III. Generic observations and weaknesses noted during the operating exams:

1. Candidates became preoccupied with hanging caution tags and OOS stickers while instrument malfunctions and casualties were still in progress. Attention to these administrative requirements is commendable, however, timing in the cases noted was improper.

2. Nonpermanent marker pen was used to list diesel trips, AFW start signals and other oral examination answers on panels, components and switchgear throughout the plant. If these operator aids are needed, they should be replaced with permanent tags or labels.

3. The control room key cabinet was improperly locked. It could be opened by turning the combination lock approximately one-quarter turn. Three instances were observed where Maintenance and Radiological Controls personnel took keys without an operations supervisor's approval. The follow-up on this issue was turned over to the senior resident inspector and will be discussed in Inspection Report 86-18.

4. Some ROs could not do a manual RCS subcooling calculation.

5. An examiner observed three plant personnel in the Spent Fuel Pool Area without proper anti-contamination clothing.

6. Some ROs could not explain the functions and principles of operation of the incore instrumentation system.

7. An SRO could not verify the accuracy of a manual calculation for an unplanned release. The unplanned release was due to a steam generator tube rupture with a stuck open atmospheric dump valve.

8. Candidates frequently reported proper SIS, CIA and CIB valve and pump alignment prior to checking or verifying the actual indication.

IV. Simulator Deficiencies noted during the Operating Examinations:

1. During a scenario, the RO identified that annunciators were alarming and resetting without a horn. When he attempted to acknowledge and test the alarm, the simulator froze. In a subsequent scenario, the manual tap changers failed to operate and the Building Services Panel Alarms could not be acknowledged. The simulator instructors attributed these deficiencies to an electrical storm that occurred the previous night and to the lack of surge suppressors for the simulator's power supply.

2. There were instances during scenarios where the turbine driven AFW pump did not start even with the proper valve alignment.

3. Candidates were distracted by erroneous electrical spikes in the megawatt, steam flow and feed flow recorders.

4. Instructors stated that the BOL snapshots were not as accurate as the MOL snapshots.

5. The operators normally have the source range fuses removed when operating in Mode 1. There is no administrative or procedural basis for this action.

6. Changing the lineup of the Boron Recovery System sometimes resulted in isolating CCR to the Reactor Coolant Pumps. The normal flow (48 gpm) is very close to the trip setpoint (50 gpm).

7. AFW pump flow was 350 gpm with the normal supply tank empty and suction valves from river water closed.

V. Generic weaknesses noted from grading of written exams:

A. RO candidates were unable to adequately explain the following:

1. The effect of starting a RCP or bypassing a string of feedwater heaters on actual critical rod position during reactor startup.

2. The effect of interstitial fission gas absorption on fuel centerline temperature.

3. What determines differential pressure across the #1 seal of a RCP.

4. The purpose of opening the turbine driven AFW pumps recirculation valve.

5. The automatic actions associated with the Containment Purge Exhaust Monitor during Refueling.

6. Immediate action substeps for E-0, Reactor Trip/SI. For example, the actions required to verify AFW flow.

7. FRGs may be entered from other than red path conditions. For example, FR-H.1 is required to be entered from E-0 when flow is less than 350 gpm.

8. Conditions that cause high activity in the RCS per AOP-43, High Reactor Coolant Activity.

B. SRO candidates were unable to adequately explain the following:

1. The design features that protect the CCR system in the event of a leak in the thermal barrier.

2. The advantages and disadvantages of using alternate dilute.

3. Reasons for closing the MSIVs on a steam line rupture.

4. System parameters checked that verify that inadequate core cooling no longer exists.

5. When pressurizer venting should take priority over containment hydrogen limits.

6. Given the RCS leakage T.S., determine if leakage limits are exceeded for a given set of plant conditions.

VI. Training/Reference Material:

1. The training material was improved from the previous examination. However, there were several instances where the material was incorrect or inadequate. (See Attachment 4)

VII. Personnel Present at the Exit Interview:

NRC Personnel

G. S. Barber, Reactor Engineer (Examiner)
B. S. Norris, Reactor Engineer (Examiner)
D. M. Silk, Reactor Engineer (Examiner)
W. Troskoski, Senior Resident Inspector
T. J. Kenny, Senior Resident Inspector
A. J. Lodewyk, Reactor Engineer

Facility Personnel

T. D. Jones, General Manager - Nuclear Operations
W. S. Lacey, Plant Manager, BV-1
J. D. Sieber, Senior Manager, BV-1
L. G. Schad, Coordinator, Simulator Training
A. J. Lindgren, Simulator Supervisor
T. E. Kuhar, Nuclear Operations Instructor
A. Nowinowski, Westinghouse Training
P. A. Russell, Nuclear Operations Instructor

VIII. Summary of NRC Comments made at exit interview:

The chief examiner reviewed the number and type of examinations administered during the previous two weeks and presented generic weaknesses observed during the simulator and oral examinations.

IX. Examination Review:

An examination review was conducted. Facility comments were discussed on a line item basis. All items were considered during grading but not all items resulted in a change to the master exam. Attachment 3 details the Facility's Comments on the written exam. Attachment 4 details the significant changes to the examinations.

Attachments:
1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)
3. Facility Comments on the Written Examination
4. NRC Response to Facility Comments

COMMENTS ON NRC SRO EXAM
7/22/86

Question	Comment
5.10.a	The key mentions power defect when it should say doppler defect.
5.10.b	There is a math error in the key. The answer should be 575.4°.
6.02.a	The reference for the answer was misinterpretted. The sample system also has delay coils to allow decay of N-16, therefore, this is not an advantage. There is no correct answer to this question and we request that it be deleted.
6.02.b	RM-DA-100 would indicate activity in the drains from the Aux. Feed pumps and should also be an acceptable answer.
6.03.a	The range of the CCR flow indicators in the Control Room are 0-1400, 0-2500 and 0-5000 gpm. It would be almost impossible to detect a 10 gpm leak on these meters. The best way to locate the leak from the control room would be by using component temperature alarms and sump alarms.
6.04.c	Another acceptable answer for when Alternate Dilute is used would be when a more rapid ρ change is desired.
6.05	The instrument failure presented in the question would not lead to a reactor trip. A high failure of PT446 would fail Tref to 577°. Rods will withdraw to raise Tavg to 577°. Reactor power will not go much higher than steam demand. Since the question presents an incorrect situation, we request that it be deleted.
6.06.a	The valve number on the key should be PCV-CH-145.
6.09.d	The answer key is incorrect. When containment pressure drops below the CIB setpoint, the CIB reset signal is cleared. If pressure goes back above the CIB setpoint, another automatic start of the quench spray pumps will result.
6.10.c	The key is incorrect. Answer 1, 2, and 3 occur simultaneously.
7.01.b	The second sentence in the key is not asked for the question and should not be required for full credit.
7.02.a	The Heat Sink red path should be < 5% in all SG's.
7.02.c	The question asked for parameters, not setpoints. Numbers should not be required for full credit.

Question	Comment
7.03.a	The key is not complete. "Run back the turbine" and "close the main steamline bypass valves" should be added to the key.
7.03.c	Same comment as 7.02.c.
7.06.a	This question asks that a step in a normal operating procedure be repeated from memory. This is not required knowledge and we ask that the question be deleted.
7.07	This question requires memorization of a procedure. This is not required knowledge and we ask that the question be deleted.
7.09.c	Time limits were not asked for in the question and are not required knowledge and should not be required for full credit.
7.10.c	Same comment as 4.07.c.
7.10.d	The referenced Tech Spec has been amended. The key should read "Operation may continue, provided the activity does not exceed the limit on the Tech Spec curve."
8.03	The question asks what action should be taken, not why. The key requires both for full credit. The first sentence should be deleted from the key.
8.10.b	Since five (5) minutes have passed already, a candidate could answer 10 minutes instead of 15 minutes for the time limitation.

ATTACHMENT 4

NRC RESPONSE TO FACILITY COMMENTS

1.04d	Accept comment
1.05a	Accept comment
1.06b	Accept comment
1.06c	Consider during grading
1.08b	Consider during grading
1.08d	Accept comment
2.01a	Accept comment - Reference material inaccuracy
2.02a	Consider during grading
2.06b	Accept comment
2.07a	Accept comment - Reference material inaccuracy
2.08b	Changed answer to CLOSE
2.09b	Accept comment
3.03a	Consider during grading
3.03b	Consider during grading
3.04b	Consider during grading
3.04d	Accept comment
3.05b	Consider during grading
3.08c	Consider during grading
3.09a	Consider during grading
3.09b	Accept comment
3.10	Accept comment
4.01a	Accept comment
4.02b	Question deleted
4.03b	Accept comment
4.04c	Consider during grading
4.07c	Accept comment
4.08c	Accept comment - Reference material inaccuracy
4.08e	Accept comment
4.09a	Accept comment
5.10a	Accept comment
5.10b	Accept comment - Math error
6.02a	Question deleted
6.02b	Consider during grading
6.03a	Consider during grading
6.04c	Consider during grading - Reference material inaccuracy
6.05	Question deleted
6.06a	Accept comment
6.09d	Accept comment
6.10c	Accept comment
7.01b	Accept comment - Redistributed points
7.02a	Accept comment
7.02c	Accept comment
7.03a	Accept comment - Four answers required
7.03c	Accept comment
7.06a	Question deleted

7.09c	Accept comment
7.10c	Accept comment
7.10d	Accept comment
7.07	Consider during grading
8.03	Accept comment - Redistribute points
8.10b	Accept comment

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

JUL 2 1987

Mr. Alfred J. Morabito
685 Tulip Drive
New Brighton, PA 15066

IN RESPONSE REFER
TO FOIA-87-A-26
(FOIA-87-202)

Dear Mr. Morabito:

This is in response to your letter dated May 15, 1987, in which you appealed Mr. Donnie H. Grimsley's form response dated May 5, 1987. Mr. Grimsley's form response denied one document in response to your Freedom of Information Act (FOIA) request for the releasable portions of NRC's letter to an individual regarding her reactor operator exam.

Acting on your appeal, I have carefully reviewed the record in this case and have determined that the previously withheld document will continue to be withheld from public disclosure pursuant to Exemption (6) of the FOIA (5 U.S.C. 552(b)(6)) and 10 CFR 9.5(a)(6) of the Commission's regulations. Your appeal is, therefore, denied.

However, I will confirm that the document contains a statement to the effect that emergency boration was not necessary.

This is a final agency action. As set forth in the FOIA (5 U.S.C. 552(a)(4)(B)), judicial review of this decision is available in a district court of the United States in the district in which you reside or have your principal place of business or in the District of Columbia.

Sincerely,

James M. Taylor for
Victor Stello, Jr.
Executive Director for Operations

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

June 16, 1987

ACKNOWLEDGMENT

This acknowledges receipt of your recent Freedom of Information Act (FOIA) request. The FOIA number which it has been assigned and the date it was received by this office are noted on the face of your request, a copy of which is enclosed.

The NRC will respond to your request as soon as possible; however, because of the existing backlog of cases and the time needed to search for records subject to your request, there may be some delay in our response.

Your request has been assigned for processing. If you have not heard from us within the next two weeks, or have any questions about the status of your request, you may telephone the staff member whose name and telephone number are identified below.

FOIA-87-352

Connie Pappas

492-8992

Sincerely,

Linda Robinson

Linda L. Robinson, Chief
Freedom of Information and Privacy
Acts Branch
Division of Rules and Records
Office of Administration

Enclosure: As stated

Docket No. 55-60755

June 5, 1987

FOIA REQUEST

Mr. D. H. Grimsley, Director
Division of Rules and Records
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

FREEDOM OF INFORMATION ACT REQUEST
FOIA-87-352
Rec'd 6-15-87

Dear Mr. Grimsley:

Pursuant to the Freedom Of Information Act, I request that copies of the following records be provided to me:

1. NUREG 1021, Rev. 2, 4/15/86
   a) ES-104
      Corrected summary sheets for each time my grade was changed during the appeal process.
   b) ES-105
      1) The training program provided for and given to Messrs. B. Norris and D. Silk: especially forms, ES-105-1 & -2.
      2) Additional training required for B. Norris and D. Silk from among items B.1.a,b,c(1)(2) and d.
      3) Number and types of examinations observed by B. Norris and D. Silk prior to their qualification as examiners.
      4) The oral report forms and the recommendations made by B. Norris and D. Silk per section 4.
      5) The written evaluation completed as a result of B. Norris and D. Silk completing the administration of their first written examination.
      6) Verification that all cited deficiencies for B. Norris and D.Silk were corrected prior to their certification as examiners.
      7) The certification forms qualifying B. Norris and D. Silk to conduct licensing examinations. These forms were to be issued by Region 1 to the Chief, Operator Licensing Branch.
      8) The annual evaluations of B. Norris conducted prior to July, 1986; especially NRC Form 308.
      9) Documentation of B. Norris's two year refresher training, including the reviewer's evaluation and suggestions.

10) Proceedings of the annual training meeting at OLB Headquarters or, if not conducted, documentation of the reasons why its usefulness was reevaluated.

c) ES-108

1) Documentation of the independent QA review of the grading of the July, 1986 licensing exam conducted at Beaver Valley Power Station.

2) The QA checkoff sheet, Attachment 1 of ES-108.

d) ES-107

1) Documentation of the independent QA review of the construction of the written exam conducted at Beaver Valley in July, 1986.

2) The Section Chief's certification that the QA review was completed.

3) Attachment 1 of ES-107.

e) ES-202

1) Documentation that the QA review ensured "only one correct answer for a question".

2) Documentation that the questions were read and reviewed by the author and at least one other examiner.

f) ES-302

1) The examiner's notes as required in 2.a.

2) All materials collected per 2.d.

3) Simulator Event Forms per F.2. Stage 1, and applicable competency checklists, Attachment 9.

I will be expecting to recieve this information within 10 working days of your reciept of this request or an explanation as to why more time is required by your agency. At any rate this request should be satisfied within 20 working days.

Please send the information to: A. J. Morabito
685 Tulip Drive
New Brighton, Pa. 15066

Sincerely,

A. J. Morabito

A. J. Morabito

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

Mr. A. J. Morabito
685 Tulip Drive
New Brighton, PA 15066

JUL 1 1987

IN RESPONSE REFER
TO FOIA-87-352

Dear Mr. Morabito:

This is in response to your letter dated June 5, 1987, in which you requested, pursuant to the Freedom of Information Act (FOIA), copies of six categories of records related to Reactor Operator/Senior Reactor Operator examinations

The NRC staff estimated that approximately 18 hours of professional search at \$12.00 per hour would be involved to search for the requested records. NRC regulations provide for the waiver of four hours of search. Therefore, if you are interested in having NRC undertake search for the requested records, please advise in writing your agreement to pay the estimated search fees of \$168.00 (14 hours x \$12.00 per hour). The breakdown of search hours according to the categories identified in your letter are as follows:

Category	1a.	1 hour of search
	b.	10 hours
	c.	1 hour
	d.	1 hour
	e.	1 hour
	f.	4 hours

Additionally, the charge for reproducing records not located in NRC's Public Document Room is five cents (\$0.05) per page, as specified in NRC's regulations. Our Division of Accounting and Finance bills the recipient directly.

Your request will not be further processed until you agree to bear the estimated costs or a determination is otherwise made with respect to the applicable fees.

If we do not receive a written response within 30 days from the receipt of this letter, we will assume that there is no further interest in subject NRC records and will close our file on your request.

We will be pleased to talk with you by telephone or meet with you to discuss your request or how the scope of your request might be modified or limited in order that it might be processed more expediously. If you have any questions in this matter, please telephone Mrs. Pappas of my staff at (301) 492-8992.

Sincerely,

Donnie H. Grimsley

Donnie H. Grimsley, Director
Division of Rules and Records
Office of Administration and
Resources Management

685 Tulip Drive
New Brighton, PA 15066
July 3, 1987

Mr. Donnie H. Grimsley, Director
Division of Rules and Records
Office of Administration and
Resources Management

SUBJECT: FOIA - 87-352

Dear Mr. Grimsley:

I received your notification dated July 1, 1987 that the estimated fees for searching for records requested in my letter of June 5, 1987 are $168.00. I agree to bear the costs of search and reproduction of those records. I ask that you proceed expediently with the processing of the request.

At this time, I also ask that the NRC waive or reduce the fees in accordance with 10 CFR 9.14a. for the following reasons:

1. The use of the documents in forthcoming legal proceedings will prove the inadequacy of the NRC in its ability to judge the competence of licensed operators in Region 1. In recognition of that inadequacy, I will press for reform of the license examination process. That reform when implemented, will greatly enhance the protection of the public health and safety.

2. As a result of the legal proceedings in this matter, an improved regulatory process will be achieved.

3. The requested records will contribute substantially to public debate on the important issue of the credentials that an NRC license examiner should have in order to examine candidates and grant or deny operator licenses.

4. Since these records will be used in the preparation of my appeal on the denial of my operating license, and since this is the first appeal on such an issue to reach the public hearing stage, the records will contribute substantially to what may become a precedent setting legal case with substantial historical significance.

Regardless of the outcome of this request for waiver or reduction of fees, please process my request for records, FOIA - 87-352.

Sincerely,

Alfred J. Morabito

Alfred J. Morabito

jlm

cc