Post-examination Comments

(Green Paper)

1. Licensee Submitted Post-examination Comments

MCGUIRE JUNE 2003 EXAM
50-369/2003-301 AND
50-370/2003-301

JUNE 16 - 30, 2003

Duke Power®
A Duke Energy Company

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July 7, 2003

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Subject: McGuire Nuclear Station
Initial Written Licensing Examination Comments
50-369/2003-301 and 50-370/2003-301

The enclosed initial written examination comments are provided in accordance with NUREG-1021, Section 402. These formal comments are associated with the McGuire Nuclear Station initial written licensing examination administered on June 30, 2003.

The following attachments are also provided:

1) McGuire Technical Specification 3.6.5, Containment Air Temperature
2) OP-MC-RAD-RP, "Radiation Protection Lesson Plan"
3) OMP 4-3, "Use of Abnormal and Emergency Procedures"
4) McGuire Nuclear Station Lesson Plan BNT-RT08, "Reactor Operational Physics"
5) OP/1/A/6450/001, "Containment Ventilation System "
6) OP/1/A/6400/006, "Nuclear Service Water System"

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SRO QUESTION # 3 ***Bank Question: 207.1*** ***Answer: C***

1 Pt. Unit 1 is preparing for a reactor start up following a refueling outage. Given the following conditions:

- T_{avg} = 515 °F
- Plant heatup in progress using NCPs

At 0200, a Station Engineer reports that a mistake had been made in analyzing the containment Appendix J Leak Rate Test results that were conducted prior to exceeding 200 °F in mode 5. Reanalysis indicated that the combined containment leak rate (Type A) had exceeded 1.0 L_a.

Which one of the following actions are required by Tech Specs in response to this situation?

REFERENCES PROVIDED
Tech Spec 3.6.1and Bases

A. Commence a plant cooldown to reach mode 5 within 36 hours.

B. Commence a plant cooldown to reach mode 5 within 37 hours.

C. Commence a plant cooldown to reach mode 5 within 42 hours.

D. Commence a plant cooldown to reach mode 5 within 43 hours.

Distracter Analysis:
A. Incorrect: Plausible.
B. Incorrect: Plausible.
C. Correct.
D. Incorrect: Plausible.

SOURCE: BANK

LEVEL: SRO

LEVEL OF KNOWLEDGE: ANALYSIS

KA: SYS 103 A2.01 (2.0*/2.6)

OBJECTIVES: OP-MC-ADM-TS Obj. 2

REFERENCES: Tech Spec 3.6.1 and Bases
OP-MC-ADM-TS pages 29, 31

Comment

The examination answer key erroneously lists “C” as the correct answer.

Proposed Resolution

The facility makes a recommendation that the answer key be revised to list “B” as the correct answer to this question.

<u>SRO Question # 24</u> *<u>Bank Question: 1072</u>* *Answer: A*

1 Pt(s) As an SRO working on a 'Complex Maintenance Plan' you are asked to evaluate four possible work teams who must repair filter housing in a 1500 mRem/hr radiation field.

Which one of the following work teams would maintain station ALARA?

A. A qualified male worker who has previously performed this task. He can complete this job in 20 minutes. This worker has exceeded his 'Alert' level for exposure and will require a dose extension.

B. Two male workers who are qualified to perform the task. Together they can perform the task in 15 minutes. Both workers have already accumulated 325 mRem this year.

C. A team of a female worker who is qualified to perform the task and a male worker who needs to qualify to this task. The female is a declared pregnant worker. The team will need 15 minutes to complete the task. The female has no dose and the male worker has 200 mRem for the year.

D. A team of a male and female both are qualified to the task but will take 20 minutes to complete the task. Each has less than 100 mRem this year.

Distracter Analysis:

A. **Correct**: 500 mR total.
B. **Incorrect**: 750 per mrem total
Plausible
C. **Incorrect: Declared pregnant worker.**
Plausible:
D. **Incorrect:** 1000 mrem total

Level: SRO

KA: G2.3.2 (2.5/2.9)

Lesson Plan Objective: RAD RP Obj. 135

Source: New

Level of knowledge: comprehension

References:
1. OP-MC-RAD-RP page 73

Comment

The question as written does not provide the candidate with enough information to adequately discern between either "A" or "B" as the more correct answer. The McGuire Nuclear Station ALARA program provides guidance that will allow either situation to be selected dependent upon a closer evaluation of the situation, including additional data not provided in the question.

Proposed Resolution

The facility recommends that both answers "A" and "B" be accepted as a correct answer to the question.

Justification

Answer "A" was originally selected because the total dose received during the evolution (500 mrem to one individual) was less than would be received by workers in the other distractors. Also, an extension to exceed an administrative dose limit is allowed with the permission of the station Radiation Protection manager. This answer supports the ALARA goal of maintaining a low collective dose for the station.

Answer "B" is equally correct because the dose received by each of the two individuals (375 mrem) is less than would be received by individuals in the other distractors. Also, an extension to exceed an administrative dose limit is not required for this situation. This answer supports the ALARA goal of maintaining a low individual dose. McGuire lesson plan OP-MC-RAD-RP Rev. 01, Radiation Protection, states the following on page 73 of 93:

> ALARA is a philosophy aimed at the minimizing exposure thru a management commitment. The goals and efforts of the McGuire Nuclear Station program are simple:
>
> - To maintain the annual dose to each individual ALARA
> - To maintain the collective dose (total person-rem) ALARA
> - Both points have to be considered simultaneously, as one without the other is not ALARA.

When the station Radiation Protection manager was asked to review this question following the exam, he replied that either answer "A" or "B" could be correct given the limited amount of information available in the question stem.

Training material associated on ALARA and this topic is supplied as part of this comment package.

- OP-MC-RAD-RP

RADIATION PROTECTION

LESSON PLAN

REFERENCES

1. Radiation Worker Training Lesson Plan (1995 Rev 01)
2. System Radiation Protection Manual (2/16/96)
3. 10CFR20 (1995)
4. MNS Technical Specifications. Rev. 3/21/96
5. PIP 1-M91-0428
7. PIP 2-M92-0356
8. PIP 2-M92-0358
9. PIP 2-M93-0685
10. PIP 2-M93-1019

COMMITMENTS

Review of ALARA goals and objectives

INSTRUCTOR AIDS

Transparencies

7.1, Alpha Radiation (07/08/96)

7.2, Beta Radiation (07/08/96)

7.3, Neutron Radiation (07/08/96)

7.4, Gamma Radiation (07/08/96)

7.5, Deep Dose Equivalent (07/08/96)

7.6, Committed Effective Dose Equivalent (07/08/96)

7.7, Committed Dose Equivalent (07/08/96)

7.8, Total Effective Dose Equivalent (07/08/96)

7.9, Lens Dose Equivalent and Shallow Dose Equivalent (07/08/96)

7.10, Administrative and Legal Dose Limits (07/08/96)

7.11, Definition of ALI and DAC (07/08/96)

7.12, Relationship Between DAC, ALI, TEDE, and MREM/HR (07/08/96)

7.13, Radiation Control Area Posting (07/08/96)

7.14, Radiation Area Posting (07/08/96)

7.15, High Radiation Area Posting (07/08/96)

7.16, Extra High Radiation Area Posting (07/08/96)

7.17, Very High Radiation Area Posting (07/08/96)

7.18, Airborne Radioactivity Area Posting (07/08/96)

7.19, Contaminated Area Posting (07/08/96)

7.20, Hot Spot Posting (07/08/96)

7.21, Low Exposure Waiting Area Posting (07/08/96)

7.22, Significant Dose Contributor Posting (07/08/96)

7.23, Four Primary Modes of Radioactive Decay (07/08/96)

7.24, Three Secondary Modes of Radioactive Decay (07/08/96)

7.25, Definitions of Roentgen, RAD, REM, Sievert (07/08/96)

7.26, Emergency Exposure Limits (07/08/96)

7.27, Daily Dose Report (07/08/96)

7.28, General Rule of Thumb for Spills (07/08/96)

OVERVIEW

This lesson will cover general radiation protection work practices, legal requirements and administrative controls. The text of the Radiation Worker Lesson Plan will be utilized however specific Operations Objectives have been written.

INSTRUCTOR ACTIVITY

The objectives listed in Section # 1 through Section # 15 are grouped to correspond to the sections in the Radiation Worker and Respiratory Training Student Guide. If these sections are to be covered as a classroom lecture, it is recommended that each section be handled as a mini-lesson plan. For example, only the Section #1 objectives would be reviewed prior to teaching Section #1. Upon completion of Section #1, the Section #1 objectives would be reviewed and then the instructor would review the Section #2 objectives prior to covering Section #2 and so forth. Upon completion of the entire lesson plan (all 16 sections) , only Section #16 objectives would be left for final review.

It is important for the instructor to stress that the test for the material in the Radiation Worker and Respiratory Training Student Guide will be based on the objectives in this lesson plan and not the ones listed in the Radiation Worker Training Guide.

Objectives which are marked with a " **G**" are not required to be covered as part of the class lecture. These objectives have been identified as not needing additional coverage since the coverage they receive in Radiation Worker and Respiratory Initial and Annual Training (**G**eneral Employee Training) is considered sufficient. The students are still responsible for this material on the test.

To aid the instructor in covering the Radiation Worker and Respiratory Training Student Guide (Sections 1 - 15), the objectives marked to be covered in class will be indicated on the Instructor Activity page with the handbook page number and sub-section number were the material is located. The instructor will need to refer to the Objective Page to determine if the objective to be covered in class is applicable to an ISS , HLP or Requal setting.

	OBJECTIVE	**RWT Page #**	**Section #**
3	List the sources of radiation in the plant including the following: • Reactor Coolant • Activation and Corrosion Products • Reactor Operations	8-9	1.2

CLASSROOM TIME (Hours)

NLO	NLOR	LPRO	LPSO	LOR
5.5	5	6	6	5

OBJECTIVES

The objectives listed below in Section # 1 through Section # 15 are grouped to correspond to the sections in the Radiation Worker Student Guide. This was done to aid in tying the objectives to the associated text material. Objectives listed under Section #16 are not supported in the Radiation Worker Respiratory Training Student Guide text to the level required by Operations. The text for this material is covered under Section 3.0 through 7.0 of this lesson plan.

Objectives which are marked with a "**G**" are not required to be covered as part of the class lecture. These objectives have been identified as not needing additional coverage since the coverage they receive in Radiation Worker Initial and Annual Training (**G**eneral Employee Training) is considered sufficient. Each student should review the objectives to determine if they need to self-study these items. The students are still responsible for this material on the test.

Section # 1 Sources of Radiation

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
1	State the basic structure of the atom including the three primary components.	G	G	G	G	G
2	Describe how radiation results from the nuclear process.	G	G	G	G	G
3	List the sources of radiation in the plant including the following: • Reactor Coolant • Activation and Corrosion Products • Reactor Operations	G	G	X	X	G
4	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
6	Characterize the four types of radiation by the follow: • Penetrating ability • Methods of shielding • Exposure Hazard • Where it is found	13 - 14	2.1
7	State the relationship between Total Effective Dose Equivalent (TEDE) and internal and external dose.	15	2.2

Section # 2 Types and Measurement of Radiation

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
5	List the four types of radiation normally encountered at a commercial nuclear station.	G	G	G	G	G
6	Characterize the four types of radiation by the following: • Penetrating ability • Methods of shielding • Exposure Hazard • Where it is found	X	X	X	X	X
7	State the relationship between Total Effective Dose Equivalent (TEDE) and internal and external dose.	G	G	X	X	G
8	Perform conversions from rem to millirem and from millirem to rem.	G	G	G	G	G
9	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	**RWT Page #**	**Section #**
15	State how the effects of receiving a radiation dose will be affected by the age of the person receiving the dose.	22	3.6
16	State the purposes of NRC Form-4.	22	3.7

Section # 3 Biological Effects

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
10	State four things which can happen to a cell after it has been exposed to ionizing radiation.	G	G	G	G	G
11	Define the following: • chronic radiation exposure" • *acute radiation exposure*	G	G	G	G	G
12	List the doses and risk of health effects resulting from acute whole body external exposures of radiation to man.	G	G	G	G	G
13	Define the following: • Genetic effects • Somatic effects • Teratogenic effects	G	G	G	G	G
14	Identify the possible effects of radiation on an unborn child due to prenatal exposure.	G	G	G	G	G
15	State how the effects of receiving a radiation dose will be affected by the age of the person receiving the dose.	X	X	X	X	X
16	State the purposes of NRC Form-4.	X	X	X	X	X
17	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
18	Define the following terms: • Deep Dose Equivalent (DDE) • Total Effective Dose Equivalent (TEDE) • Committed Dose Equivalent (CDE) • Lens Dose Equivalent (LDE) • Shallow Dose Equivalent (SDE)	27 - 28	4.1
19	List the Duke Power Company administrative and NRC 10CFR20 limits for: • Adult Total Effective Dose Equivalent • Adult Eye Dose Equivalent (Lens of the eye) • Adult Shallow dose equivalent to the skin and extremities • Individual organ or tissue • Minors • Declared "pregnant" woman • Planned Special Exposures	29	4.1
20	State four possible consequences if any NRC radiation dose limit is exceeded.	30	4.2
21	State two options a worker's supervisor has when he/she is notified that the worker is approaching an administrative exposure limit.	30	4.4
22	Define the "Alert" and "Exclude" exposure status for an individual.	31	4.4
23	State who (by title) controls annual dose targets below 2.0 rem for individual workers.	30	4.4

Note: For objective #23, the OSM or Day Staff Manager is the " Section Manager " counterpart for Operations personnel.

Section #4 Limits and Guidelines

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
18	Define the following terms: • Deep Dose Equivalent (DDE) • Total Effective Dose Equivalent (TEDE) • Committed Dose Equivalent (CDE) • Lens Dose Equivalent (LDE) • Shallow Dose Equivalent (SDE)	X	X	X	X	X
19	List the Duke Power Company administrative and NRC 10CFR20 limits for: • Adult Total Effective Dose Equivalent • Adult Eye Dose Equivalent (Lens of the eye) • Adult Shallow dose equivalent to the skin and extremities • Individual organ or tissue • Minors • Declared "pregnant" woman • Planned Special Exposures	X	X	X	X	X
20	State four possible consequences if any NRC radiation dose limit is exceeded.	X	X	X	X	X
21	State two options a worker's supervisor has when he/she is notified that the worker is approaching an administrative exposure limit.			X	X	X
22	Define the "Alert" and "Exclude" exposure status for an individual.	X	X	X	X	X
23	State who (by title) controls annual dose targets below 2.0 rem for individual workers.			X	X	X

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
24	State the rights of a declared pregnant woman.	32	4.5
25	Define Planned Special Exposure (PSE).	33	4.6
27	State the purpose of ALARA (As Low As Reasonably Achievable).	37	5.1
28	Describe three methods of controlling external exposure.	37 - 38	5.3
29	Calculate stay time given a dose rate, current exposure, and an exposure limit.	39 - 40	5.5
30	State the primary root cause for most overexposures in US utilities.	42	Note:

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
24	State the rights of a declared pregnant woman.	X	G	X	X	G
25	Define Planned Special Exposure (PSE).	X	X	X	X	X
26	Reserved					

Section #5 ALARA

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
27	State the purpose of ALARA (As Low As Reasonably Achievable).	X	X	X	X	X
28	Describe three methods of controlling external exposure.	X	G	X	X	G
29	Calculate stay time given a dose rate, current exposure, and an exposure limit.	X	G	X	X	G
30	State the primary root cause for most overexposures in US utilities.	X	G	X	X	G
31	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
33	List the types of radiation detected by the following devices: • Thermoluminescent Dosimeter (TLDs) • Self- reading Dosimeters (SRDs) • Electronic Alarming Dosimeters (EADs)	53	6.1
35	State the two modes of operation for the Merlin Gerlin self-reading dosimeter.	55	6.2
36	State the action(s) to be taken if dosimetry is lost.	57	6.4
37	State the action(s) to be taken if your self-reading pocket dosimeter goes off-scale while you are working in a radiation area.	57	6.4
38	State the action to be taken if your Merlin Gerlin dosimetry starts alarming while you are working in a radiation area.	57	6.4

Section #6 Radiation Dosimetry

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
32	State the purpose of dosimetry.	G	G	G	G	G
33	List the types of radiation detected by the following devices: • Thermoluminescent Dosimeter (TLDs) • Self- reading Dosimeters (SRDs) • Electronic Alarming Dosimeters (EADs)	X	X	X	X	X
34	Identify how to wear dosimetry devices including placement and orientation.	G	G	G	G	G
35	State the two modes of operation for the Merlin Gerlin self-reading dosimeter.	X	G	X	X	G
36	State the action(s) to be taken if dosimetry is lost.	X	G	X	X	G
37	State the action(s) to be taken if your self-reading pocket dosimeter goes off-scale while you are working in a radiation area.	X	G	X	X	G
38	State the action to be taken if your Merlin Gerlin dosimetry starts alarming while you are working in a radiation area.	X	G	X	X	G
39	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
40	Define the following terms: • Contamination • Radiation	 63 64	 7.1 Note
41	Define the following types of contamination: • Fixed • Loose • Discrete (Hot) Particle	63	7.1
42	State the station limits above which the following would be considered contaminated: • Personnel • Personal items • Areas, tools and equipment	64	7.2
44	Identify three sources and three indications of contamination.	64 - 65	7.2
46	State how the following would be marked/labeled: • A tool in the Hot Tool Room which is available for checkout • A piece of non-contaminated equipment needing to be removed from the RCA • A piece of equipment needing to be removed from the RCA which is not cleared for unconditional release	66 - 67	7.3 - 7.4

Section #7 Types of Contamination

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
40	Define the following terms: • Contamination • Radiation	X	X	X	X	X
41	Define the following types of contamination: • Fixed • Loose • Discrete (Hot) Particle	X	X	X	X	X
42	State the station limits above which the following would be considered contaminated: • Personnel • Personal items • Areas, tools and equipment	X	X	X	X	X
43	Explain why contamination is controlled.	G	G	G	G	G
44	Identify three sources and three indications of contamination.	X	X	X	X	X
45	State five methods that are effective in preventing the spread of contamination to personnel and areas.	G	G	G	G	G
46	State how the following would be marked/labeled: • A tool in the Hot Tool Room which is available for checkout • A piece of non-contaminated equipment needing to be removed from the RCA • A piece of equipment needing to be removed from the RCA which is not cleared for unconditional release	X	X	X	X	X

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
47	State the contamination release limits for unconditional release for loose and total contamination.	66	7.4
52	Regarding discrete (hot) particles, be able to state: • The hazard • Methods to identify a discrete particle • Sources of discrete particles • Work activities that may result in discrete particle contamination • Special precautions to be used in an area that may contain discrete particles	72	7.8

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
47	State the contamination release limits for unconditional release for loose and total contamination.	X	X	X	X	X
48	Explain how to monitor personnel and personnel items for contamination including the use of: • RM-14 Frisker • Personnel Contamination Monitors (PCMs) • Small Article Monitors (SAMs) • Portal Monitors • Hand and Foot Monitors	G	G	G	G	G
49	State the actions to be taken upon indications of becoming contaminated.	G	G	G	G	G
50	State the methods for control of contaminated tools, equipment, and materials including: • Minimizing materials contaminated • Tool issue from Auxiliary Building/Hot Tool Room • Bagging/surveillance requirements	G	G	G	G	G
51	State two methods used to designate Contaminated Areas.	G	G	G	G	G
52	Regarding discrete (hot) particles, be able to state: • The hazard • Methods to identify a discrete particle • Sources of discrete particles • Work activities that may result in discrete particle contamination • Special precautions to be used in an area that may contain discrete particles	X	X	X	X	X
53	Identify three situations that require immediate exit from a Contaminated Area.	G	G	G	G	G
54	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
59	Define the following terms: • Committed Effective Dose Equivalent (CEDE) • Annual Limit on Intake (ALI) • Derived Air Concentration (DAC)	79	8.5
60	State the relationship between: • DAC-hour and ALI • ALI and TEDE • DAC-hour and mrem	80	8.6

Section #8 Internal Exposure

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
55	State four pathways for radioactive material to enter the body.	G	G	G	G	G
56	State five methods used to limit the internal deposition of radioactive materials into the body.	G	G	G	G	G
57	State the two primary processes by which radioactive material is eliminated from the body.	G	G	G	G	G
58	Identify two primary methods used to measure the amount of radioactivity in the body for internal dose determination.	G	G	G	G	G
59	Define the following terms: • Committed Effective Dose Equivalent (CEDE) • Annual Limit on Intake (ALI) • Derived Air Concentration (DAC)	X	X	X	X	X
60	State the relationship between: • DAC-hour and ALI • ALI and TEDE • DAC-hour and mrem	X	X	X	X	X
61	State five plant conditions/activities that can increase the potential for airborne radioactivity.	G	G	G	G	G
62	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
63	State three major functions of a RWP.	85	9.1
64	State two types of RWPs and the function of each.	85 - 86	9.1
65	State five different types of information that can be found on a RWP.	88	RWP Form
66	State the responsibility for complying with RWP requirements.	86	9.3
67	State five different types of information that can be found on a survey map.	90	9.4
68	State the required action(s) to be taken if the work scope or radiological conditions change so that they are not within the scope of an RWP.	92	9.5

Section #9 Radiological Work Permit

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
63	State three major functions of a RWP.	X	X	X	X	X
64	State two types of RWPs and the function of each.	X	X	X	X	X
65	State five different types of information that can be found on a RWP.	X	X	X	X	X
66	State the responsibility for complying with RWP requirements.	X	X	X	X	X
67	State five different types of information that can be found on a survey map.	X	G	X	X	G
68	State the required action(s) to be taken if the work scope or radiological conditions change so that they are not within the scope of an RWP.	X	X	X	X	X
69	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
70	Define the following radiological areas: • Restricted Area • Radiologically Controlled Area (RCA) • Radiation Control Zone (RCZ) • Radioactive Materials Area	95, 98	10.1
71	Define the following radiological areas and describe the associated postings: • Radiation Area • High Radiation Area (HRA) • Extra High Radiation Area (EHRA) • Very High Radiation Area (VHRA) • Airborne Radioactivity Area • Contaminated Area • Hot Spot • Low Exposure Waiting Area (LEWA) • Significant Dose Contributor	96 - 100	10.1

Section #10 Posting

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
70	Define the following radiological areas: • Restricted Area • Radiologically Controlled Area (RCA) • Radiation Control Zone (RCZ) • Radioactive Materials Area	X	X	X	X	X
71	Define the following radiological areas and describe the associated postings: • Radiation Area • High Radiation Area (HRA) • Extra High Radiation Area (EHRA) • Very High Radiation Area (VHRA) • Airborne Radioactivity Area • Contaminated Area • Hot Spot • Low Exposure Waiting Area (LEWA) • Significant Dose Contributor	X	X	X	X	X
72	State the potential consequences of violating, moving, or altering a radiological posting.	G	G	G	G	G
73	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
74	State the alarm indications for the following instruments when their setpoints are exceeded: • Continuous Air Monitor • Area Radiation Monitor • Process Radiation Monitor • Merlin Gerlin in dose rate mode • Merlin Gerlin in dose mode	103 - 104	11.1
75	State the proper action(s) to be taken when the following alarms are activated: • Continuous Air Monitor • Area Radiation Monitor • Process Radiation Monitor • Containment Evacuation • Merlin Gerlin in dose rate mode • Merlin Gerlin in dose mode	103 - 104	11.1

Section #11 Radiological Alarms

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
74	State the alarm indications for the following instruments when their setpoints are exceeded: • Continuous Air Monitor • Area Radiation Monitor • Process Radiation Monitor • Merlin Gerlin in dose rate mode • Merlin Gerlin in dose mode	X	X	X	X	X
75	State the proper action(s) to be taken when the following alarms are activated: • Continuous Air Monitor • Area Radiation Monitor • Process Radiation Monitor • Containment Evacuation • Merlin Gerlin in dose rate mode • Merlin Gerlin in dose mode	X	X	X	X	X
76	State the potential consequences of ignoring a radiological alarm.	G	G	G	G	G
77	Reserved					

INSTRUCTOR ACTIVITY

Section #12 Radioactive Waste

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
78	Define " radioactive waste ".	G	G	G	G	G
79	Contrast the disposal cost of radioactive waste versus non-radioactive waste.	G	G	G	G	G
80	State four effective methods for minimizing the generation of radioactive waste.	G	G	G	G	G
81	Explain why it is important to keep contaminated and non-contaminated waste separate.	G	G	G	G	G
82	Explain why it is important to keep wet and dry contaminated materials separate.	G	G	G	G	G
83	Explain why it is important to keep contaminated and hazardous waste separate.	G	G	G	G	G
84	Reserved					

INSTRUCTOR ACTIVITY

Section #13 Rights and Responsibilities

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
85	State individual rights and responsibilities regarding: • Keeping dose ALARA • Adhering to instructions provided by RP personnel (including stop work authority) , written policies and procedures, RWPs, and posted warning signs. • Maintaining awareness of current personal dose. • Remaining within federal and plant administrative dose limits and guide lines • Identifying the actions and reporting responsibilities when abnormal radiological conditions and/or violations of radiological requirements are encountered. • Knowing the rights of the individual and the process to be followed in obtaining personal radiation dose.	G	G	G	G	G
86	Reserved					

INSTRUCTOR ACTIVITY

	OBJECTIVE	RWT Page #	Section #
87	Select the correct RWP for the job to be performed.	86	9.1
88	For a given RWP, determine the following information: • Protective clothing requirements • Dosimetery requirements • Respiratory protection requirements • Any special conditions defined by the RWP • Dose rate and contamination levels	121 88	14.1 RWP form
89	Provide the correct sequence of donning protective clothing to include the hood, coveralls, gloves liners, gloves, shoe liners, and shoe covers.	122	14.2
90	Provide the correct sequence of removing protective clothing to include the hood, coveralls, gloves liners, gloves, shoe liners, shoe covers and at what point the "step off" pad is used.	123 - 129	14.3

Section #14 Practical Exercise

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
87	Select the correct RWP for the job to be performed.	X	X	X	X	X
88	For a given RWP, determine the following information: • Protective clothing requirements • Dosimetery requirements • Respiratory protection requirements • Any special conditions defined by the RWP • Dose rate and contamination levels	X	X	X	X	X
89	Provide the correct sequence of donning protective clothing to include the hood, coveralls, gloves liners, gloves, shoe liners, and shoe covers.	G	G	X	X	G
90	Provide the correct sequence of removing protective clothing to include the hood, coveralls, gloves liners, gloves, shoe liners, shoe covers and at what point the "step off" pad is used.	G	G	X	X	G
91	Enter and exit a simulated RCZ by performing the following: • Obtain access on the RWP. • Don protective clothing to include hood, coverals, glove liners, gloves, shoe liners, and shoe covers. • Meet the requirements on signs and postings. • Read a self-reading dosimeter while wearing protective clothing. • Minimize dose and spread of contamination. • Properly remove protective clothing when exiting contaminated area. • Perform required monitoring. • Ensure radiation dose is properly recorded.	G		G	G	

INSTRUCTOR ACTIVITY

Section #15 Respiratory Protection

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
92	Identify the purpose of the respiratory protection program.	G	G	G	G	G
93	Identify the four classifications of respiratory hazards.	G	G	G	G	G
94	List three types of engineering controls used to eliminate and/or reduce exposure to respiratory hazards.	G	G	G	G	G
95	Identify the qualification requirements that must be met prior to using a respirator or bubble hood.	G	G	G	G	G
96	Describe the basic operating principles of an air-purifying respirator.	G	G	G	G	G
97	Define the general application of an air-purifying respirator, and state three limitations on its use.	G	G	G	G	G
98	Define the general application of an air-supplying respirator, and state three limitations on its use.	G	G	G	G	G
99	Describe three limitations associated with bubble hood use.	G	G	G	G	G
100	Identify the purpose of a fit test.	G	G	G	G	G
101	Identify four factors that can affect the fit of a full-face respirator.	G	G	G	G	G
102	Identify when a negative and positive pressure test is required.	G	G	G	G	G
103	Define "respirator protection factor".	G	G	G	G	G

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
104	List the following respirators in order based on the protection factor they provide: • Full-face air-purifying • Air-supplied hood • Air-supplied respirator with full face mask	G	G	G	G	G
105	Describe the procedure for issue of respirators, including bubble hoods.	G	G	G	G	G
106	Describe how to perform pre-operational checks and don, test, operate, and remove required respiratory protection equipment for each type of respirator the student will be qualified to use, including the bubble hood.	G	G	G	G	G
107	Describe the procedure for returning respirators, including bubble hoods.	G	G	G	G	G
108	Identify the actions to be taken if a respirator malfunctions or physical or psychological distress occurs while wearing a respirator or bubblehood.	G	G	G	G	G
109	List two early warning signs of loss of air supply to a bubble hood.	G	G	G	G	G
110	Describe the actions to take if you lose air supply to a bubble hood.	G	G	G	G	G
111	Perform pre-operational checks. Don, test, operate and remove required respiratory equipment.	G	G	G	G	G
112	Reserved					

INSTRUCTOR ACTIVITY

Section #16 Supplemental

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
113	Explain what causes an atom to be unstable.	X		X	X	
114	Define the following terms: • radioactivity • curie	X	X	X	X	X
115	List four primary modes of radioactive decay.	X	X	X	X	X
116	List the three secondary modes of radioactive decay.	X	X	X	X	X
117	Define the following terms: • Radioactive Half-life • Biological half life • Effective half life	X	X	X	X	X
118	Define the following terms: • Internal exposure • external exposure	X	X	X	X	X
119	Define the following terms: • Roentgen • RAD • REM • Sievert	X	X	X	X	X
120	Define Quality Factor and state the Quality factors for alpha, beta, gamma and neutron radiation.	X	X	X	X	X
121	Define the following terms: • Stochastic effect • Nonstochastic effect	X	X	X	X	X
122	Define the term "Tenth Value Layer".	X	X	X	X	X

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
123	List the Tenth Value Layers (TVL) for gamma radiation for the following: • concrete • lead • steel • water	X	X	X	X	X
124	Describe three problems associated with an internal deposition of radioactive material.	X	X	X	X	X
125	Briefly explain how a TLD measures radiation.	X	X	X	X	X
126	Briefly explain how a self-reading pocket dosimeter measures radiation.	X	X	X	X	X
127	State the following information for the RM14 portable survey instrument: • Type of Detector • Radiation Measured and Detected • Approximate Range • General Application	X	X	X	X	X
128	Define technical overexposure.	X	X	X	X	X
129	State the purpose of the System Radiation Protection Manual.	X	X	X	X	X
130	Provide a basic description of the information contained in the System Radiation Protection Manual.	X	X	X	X	X
131	List the TEDE dose limits for Emergency Exposure.	X	X	X	X	X
132	State the approval required prior to the use of the Emergency Exposure dose limits.	X	X	X	X	X

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
133	Identify the type of exposures (i.e., Emergency Exposure, Planned Special Exposure, etc.) that are included as part of a radiation workers total allowable exposure.	X	X	X	X	X
134	Discuss the way in which exposure to neutron radiation is considered for the following: • Duke's administrative policy requirements for neutron dosimetry for entering the RCA • How neutron dose accumulation is tracked between the times that the individual's TLD is read	X	X	X	X	X
135	State the goals and efforts of the ALARA Program.	X	X	X	X	X
136	For the following areas: • Define and describe MNS Unrestricted Area • Describe MNS Restricted Area • Describe MNS RCA	X	X	X	X	X
137	State the dose limit associated with an unrestricted area.	X	X	X	X	X
138	List the requirements for wearing dosimetry devices inside and outside the RCA.	X	X	X	X	X
139	State the additional controls placed on entry/access to: • High Radiation Area • Very High Radiation Area • Extra High Radiation Area	X	X	X	X	X
140	Explain when a RWP is required.	X	X	X	X	X
141	Discuss how long a RWP or SRWP is valid.	X	X	X	X	X
142	Correctly interpret the information on the Daily Dose Report.	X		X	X	
143	State the purpose of protective clothing.	X	X	X	X	X

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
144	Describe the general rule of thumb for radioactive spills.	X	X	X	X	X
145	Describe three radiological problems that may result from a fire in areas where radioactive materials are stored and/or used.	X	X	X	X	X
146	State why vacuum cleaners used within the RCA are controlled.	X	X	X	X	X
147	State the immediate notification requirements listed in 10CFR20.2202 Notification of incidents.		X	X	X	X

INSTRUCTOR ACTIVITY

OUTLINE

1.0 INTRODUCTION

2.0 RADIATION WORKER AND RESPIRATORY PRESENTATION

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INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
113	Explain what causes an atom to be unstable.	X		X	X	

1.0 INTRODUCTION

This lesson plan will address fundamentals of radiation protection, administrative controls and the legal requirements associated with radiation protection to ensure personnel safety and the control of radioactive equipment/materials as applicable to operations personnel. The first portion of this lesson plan will use the Radiation Worker and Respiratory Training Student Guide as the text however the objectives to be used are the objectives listed in this lesson plan, not the objectives listed in the Radiation Worker and Respiratory Training Student Guide.

2.0 RADIATION WORKER AND RESPIRATORY PRESENTATION

Refer to the Radiation Worker and Respiratory Training Student Guide in section 8.0 below for objectives listed in Objectives Section 1 through Section 15 above.

3.0 SUPPLEMENTAL

3.1. Fundamentals

Objective #113

All matter is composed of atoms. Atoms in turn are composed of three primary components:

- **Protons** are particles in the nucleus of the atom which have a positive electrical charge. The nucleus of a hydrogen atom is a single proton. The atomic number of an atom is the number of protons that the atom contains.
- **Neutrons** are also located in the nucleus and have a neutral electrical charge. The mass of a neutron is approximately equal to the mass of a proton.
- **Electrons** are particles which form a cloud around the nucleus of the atom. The electrons have very little mass and carry a negative electrical charge.

The protons and neutrons contain a nuclear force which acts over a short range. This nuclear force is an attractive force and provides the cohesive force of the nucleus. The positive electrical charge on the protons produces a repulsive force (like charges repel each other). In order for the nucleus to be stable, the electrical (coulomb) force should be equal to the nuclear force. Since the forces are not equal in strength, as the atoms become larger the number of neutrons must increase more rapidly than the number of protons in order for the nucleus to be stable. If the ratio of neutrons to protons is too high or too low the nucleus could be unstable. Some nuclear interactions can cause an imbalance in the proton-neutron ratio resulting in an unstable atom.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
114	Define the following terms: • radioactivity • curie	X	X	X	X	X
115	List four primary modes of radioactive decay.	X	X	X	X	X

Objective #114

When an atom becomes unstable, it emits packets or waves of energy or particles. The emission of this energy is called radiation. 10CFR20 defines radiation as " Alpha particles, beta particles, gamma rays, X-rays, neutrons, high speed electrons, high speed protons and other particles capable of producing ions. Radiation , as used in this part, does not include non-ionizing radiation, such as radio- or microwaves, or visible, infrared or ultra violet light".

The decay of an atom from one nuclear ground state to another is called disintegration. **Radioactivity** is define as the spontaneous disintegration of a radioactive atom with the emission of radiation.

The **curie** is defined as the amount of radioactive material that decays at a rate of 3.7E10 dps. The unit of the curie is disintegration's per second (dps) . As stated earlier, a disintegration is the decay of an atom from one nuclear ground state to another. When going from one ground state to another, an atom may give off multiple particles. Therefore the 3.7E10 dps is not the number of particles being emitted.

Objective #115

The four primary modes of decay rearrange the structure of the nucleus towards a more stable atom. The **four primary modes of decay** and their decay equations are :

1. **Alpha decay** (α) : $_{Z}X^{A} \rightarrow {}_{Z-2}Y^{A-4} + {}_{2}He^{4}$ (A Helium nucleus is ejected from the parent atom.)

2. **Beta decay** (β^-) : $_{Z}X^{A} \rightarrow {}_{Z+1}Y^{A} + \beta^-$ (A neutron decays into a proton by ejecting an energized electron from the nucleus. The parent atom gains a proton and loses a neutron.)

3. **Positron decay** ($\beta+$): $_{Z}X^{A} \rightarrow {}_{Z-1}Y^{A} + \beta+$ (A proton decays into a neutron by ejecting an energized positively charged electron from the nucleus. The parent gains a neutron and loses a proton.)

4. **Electron capture** (EC): $_{Z}X^{A} + e^- \rightarrow {}_{Z-1}Y^{A}$ (The nucleus actually captures one of the inner orbital electrons and therefore loses a proton and gains a neutron.)

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
116	List the three secondary modes of radioactive decay.	X	X	X	X	X
117	Define the following terms: • Radioactive Half-life • Biological half life • Effective half life	X	X	X	X	X
118	Define the following terms: • Internal exposure • external exposure	X	X	X	X	X

Objective #116

The secondary modes of decay do not result in a change in the structure of the nucleus but removes the excitation from the daughter products. The **three secondary modes of radioactive decay** are:

1. **Gamma rays** (γ) are electromagnetic energy emitted from the nucleus following the primary mode of decay.

2. **Internal conversion** - The excited nucleus interacts with a tightly bound electron. The electron absorbs the excitation energy and is ejected from the atom. The vacant inner electron shell is then reoccupied by another electron resulting in characteristic X-rays.

3. **Isometric Transition** - This decay mode is the same as the gamma except there is a measurable time delay between the primary and secondary decay. Example: Cs-137 decay to BA-137M with a half-life of 2.55 minutes. The M in BA-137M stands for metastable.

3.2. Biological Effects

Objective #117

The **radioactive half-life** is the time it takes for a radioisotope to lose half of its activity.

The **biological half-life** is the time required for the body to eliminate half of any substance by normal body processes. This time period would be the same for radioactive or stable isotopes of the same element.

The **effective half-life** is the time required for a radioactive element fixed in tissue to be reduced to half as a result of a combination of radioactive and biological half-life.

Objective #118

The source of radiation can come from within the body or externally to the body. **Internal exposure** is defined as that portion of the dose equivalent received from radioactive material taken into the body. **External exposure** is defined as that portion of the dose equivalent received from sources outside the body.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
119	Define the following terms: • Roentgen • RAD • REM • Sievert	X	X	X	X	X
120	Define Quality Factor and state the Quality factors for alpha, beta, gamma and neutron radiation.	X	X	X	X	X

Objective #119

The following terms are use to quantify the energy associated with radiation:

Roentgen (R) is a measure of X or gamma radiation exposure to air. One roentgen is equal to 83 ergs of energy deposited into a gram of air. This measure does not address the biological effects of radiation.

RAD is the special unit of absorbed dose. One rad is requal to an absorbed dose of 100 ergs/gram or 0.01 joule/kilogram (0.01 gray).

Rem is the special unit of any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor (*1 rem = 0.01 sievert*) .

1 Rem = (1 RAD) X (QF of specific radiation)

Sievert (Sv) is the SI unit of any of the quantities expressed as dose equivalent. The dose equivalent in sieverts is equal to the absorbed dose in grays multiplied by the *quality factor. (1 Sv = 100 rems).*

Gray (Gy) is the SI unit of absorbed dose. One gray is equal to an absorbed dose of 1 joule/ kilogram (100 rads).

Objective #120

Quality factor (QF) *is a modifying factor that is used to normalize each radiation's* relative biological effect.

The quality factors for the four radiation's are:

- Alpha = 20
- Beta = 1
- Gamma = 1
- Neutron (Fast) = 10 , (Thermal) = 2

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
121	Define the following terms: • Stochastic effect • Nonstochastic effect	X	X	X	X	X
122	Define the term "Tenth Value Layer"	X	X	X	X	X
123	List the Tenth Value Layers (TVL) for gamma radiation for the following: • concrete • lead • steel • water	X	X	X	X	X

Objective #121

There are two opinions on how the amount of radiation received is related to biological effects. The threshold theory believes that there is certain amount of radiation which must be reached before any significant damage occurs. The linear theory believes that any amount of radiation exposure is harmful. Along this line, two terms further classify health effects which can occur:

Stochastic effects: - Health effects that occur by chance and occur in unexposed individuals as well as exposed individuals. A certain minimum dose is not required for the effects to be seen and an increase in the dose does not increase the magnitude of the effect, instead the probability increases. Lung cancer due to smoking is an example.

Nonstochastic effects: - Health effects for which the severity of the effects varies with the dose received. It is believed that a particular threshold dose exist. Radiation induced cataracts would be an example.

Objective #122

Typical shielding materials utilized in the plant are water, concrete, steel and lead. Each shield material has its own attenuation thickness. Two terms are used to define attenuation factors are:

Tenth value layer (TVL) is the thickness of the material that will reduce the exposure rate by a factor of ten.

Half value layer (HVL) is the thickness of material that will reduce the exposure rate by a factor of two.

Objective #123

Typical values for TVL and HVL for gamma radiation are:

Shield material	TVL	HVL
Water	24"	8"
Concrete	12"	4"
Steel	4"	1"
Lead	2"	0.5"

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
124	Describe three problems associated with an internal deposition of radioactive material.	X	X	X	X	X
125	Briefly explain how a TLD measures radiation.	X	X	X	X	X
126	Briefly explain how a self-reading pocket dosimeter measures radiation.	X	X	X	X	X

Objective #124

A dose can be received from radioactive material which is taken into the body. Trying to utilize time, distance and shielding in this situation is of little help. The problems with internal exposure are:

- Exposure continues 24 hrs per day
- It is difficult to measure
- Some isotopes seek and concentrate in one or more organs

3.3. Dosimetry

Objective #125

The **TLD (Thermoluminescent Dosimeter**) is used for the permanent occupational dose record. It can measure gamma, neutron and beta. The TLD contains lithium fluoride crystals. Electrons in these crystals are elevated to a metastable energy level when exposed to radiation. The electrons prefer the ground state but the metastable state "traps" the electrons. A special TLD reader is used once per quarter to " read " the exposure. It does this by heating the crystals which causes the electrons in the metastable state to go to a higher unstable state. When the electrons drop to ground state they give off light which is converted to an electrical signal by the TLD reader. The light emitted is proportional to the number of electrons which have been raised to the metastable state which is also proportional to the amount of radiation which has interacted with the TLD.

Objective #126

There are two self reading dosimeters used at McGuire which are the pocket dosimeter (PD) and the Merlin Gerlin (MG). The MG is the primary instrument used for self-reading Dosimetry. The PDs are used if the Electronic Dose Capture (EDC) System is down and the MGs are out of service and for some specialized jobs. The PD is zeroed by placing an electrical charge in the chamber with a calibration device. When fully charged a small hair like fiber which is used as an indicator is aligned with zero on the scale. As the dosimeter is exposed to radiation, the chamber will lose its charge causing the fiber to move up scale. Most PDs have a range from 0 mrem to 500 mrem and are re-zeroed at 60% of full scale or 300 mrem. The reading on the PD can also be affected by leakage within the detector, high humidity and mechanical shock.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
127	State the following information for the RM14 portable survey instrument: • Type of Detector • Radiation Measured and Detected • Approximate Range • General Application	X	X	X	X	X

Objective #127

For personnel decontamination, the RM-14 instrument is available. This instrument detects beta and gamma radiation. The instrument has three scales which cover the following ranges:

Scale	Range
• X1	0 - 500 cpm
• X2	0 - 5000 cpm
• X3	0 - 50,000 cpm

For personnel contamination, the instrument must be on the X1 scale with background reading less than 240 cpm. The alarm is set for 100 cpm above background. Frisking is performed with the probe within 1/2 inch from the skin for about 40 seconds for a partial frisk and three minutes for a whole body frisk.

The RM-14 detector is a Geiger-Mueller (GM) tube. The GM tube is designed such that a single ionizing event produces an avalanche of secondary ionization's. At some point the ability of the tube to continue the secondary ionization's stops. The tube is then ready to respond to another event. The avalanche produces a pulse which is counted by the detector which corresponds to the detector readout in counts per minute (cpm). The avalanche operation of the detector has the following operating characteristics:

- Each pulse produced by the detector is independent of the energy of the incident radiation. Thus the detector is not able to differentiate different energy levels of radiation.
- If another particle enters the GM tube during avalanche it will not produce a pulse large enough to be detected. This period is referred to as dead time. A quench gas such as halogen is designed into the GM tube to reduce the time it takes the tube to recover.
- For very high radiation fields, it is possible for the GM tube to go into a continuous discharge mode which doesn't allow the detector to recover. In this situation. a " fold over" condition occurs which causes the instrument readout to go low giving the appearance that the person has passed through the high radiation field. Operation in radiation areas of this level require a RP escort who is trained to use the appropriate monitoring instrumentation.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
128	Define technical overexposure.	X	X	X	X	X
129	State the purpose of the System Radiation Protection Manual.	X	X	X	X	X
130	Provide a basic description of the information contained in the System Radiation Protection Manual.	X	X	X	X	X

3.4. 10CFR20 and System Radiation Protection Manual

Objective #128

Exceeding the 10CFR20 permissible exposure limit(s) is called a **technical overexposure.**

Objective #129

Per 10CFR20.1101 (Radiation Protection Programs): "Each licensee shall develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of this part." The purpose of the System Radiation Protection Manual is to describe the Radiation Protection Program which has been designed specifically for Duke Power facilities to ensure that their operation complies with 10CFR20 regulations and, insofar as possible, with applicable Regulatory Guides.

Objective #130

The following is a listing of the type of information found in the System Radiation Protection Manual:

Volume I

- Section I : Radiation Protection Program and General Information: Includes: **Occupational Dose Limits**, Surveys and Monitoring, Contamination Control and Decontamination, Respiratory Program, Storage and Control of Radioactive Materials, Records, Reports and Required Notifications, **Radioactive Stop Work Authority**, Implementation of Manual Directives.
- Section II : Personnel Monitoring : Includes: Radiation Access and Monitoring Devices, Radiation Exposure Records, Beta Program, **Planned Special Exposures**, TLD to EAD/PD correlation, **Declared Pregnant Worker.**
- Section III Exposure Control : Includes: **Use of RWPs**, Taking and Counting Surveys, Posting of RCZs, Airborne Radioactive Control and Accountability, Personnel Contamination Monitoring, Tool, Equipment and Area Decontamination, Removal of items from RCA/RCZs, S/G Work on primary side, Diving Operations, Access Controls for High, Extra High and Very High Radiation Areas, **Exposure Extensions and Reduction**, Radiological Respiratory Program, Use of Vacuum Cleaners, Neutron Exposure Monitoring, Use of Portable Ventilation Systems in RCAs, Hot Particle Program.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
133	Identify the type of exposures (i.e., Emergency Exposure, Planned Special Exposure, etc.) that are included as part of a radiation worker's total allowable exposure.	X	X	X	X	X

- Section IV Radioactive Materials Control and Waste Disposal : Includes: Labeling and Marking of Containers, Radioactive Waste Handling, Effluents, Radiological Environmental Monitoring, Repose or Disposition of Bulk Material or Clean Trash from RCAs.
- Section V Radioanalysis and Instrumentation: Includes: Quality control, testing and calibration of various counters, dosimetry and analysis instruments, Procurement of new RP equipment.
- Section VI Emergency Response: Includes: Internal Dose Assessment Area, Investigation of Unusual Dosimetry Incidents or Possible Overexposures. Investigation of Skin and Clothing Contamination's, Radiation Accident and Emergency Procedures, **Emergency Dose Limits for Occupational Exposed personnel.**
- Section VII Radiation Protection Program Control: Includes: Audits and Assessments, Records, Logs and Reports.
- Section VIII Agreement State Activities: Includes: Control of Radiography.

Volume II

- Appendix A Glossary of Terms
- Appendix B Regulatory Cross-reference This section provides a cross reference to show where the 10CFR 20 requirements are addressed in the System Radiation Protection Manual. It also lists the associated INPO and Regulatory Guide references.
- Appendix C Technical Positions
- Appendix D References

Objective #133

Each radiation worker will have their own dose/exposure history record. This record will be used to track the accumulation of exposure to determine lifetime exposure and to determine if the individual should be limited in obtaining additional dose due to annual or lifetime limits. *All dose received from occupational exposure ("normal" , planned* special exposure and emergency exposure) will be recorded in the individual's dose history. Within this exposure history record there are two accounts. One account is for exposure received during regular work activities and emergencies. The other account is for planned special exposures. In addition, if a person exceeds the normal 10CFR20.1201 limits for exposure, the amount in excess of the limit will be charged against their planned special exposure account. Each account has it's own limits for

INSTRUCTOR ACTIVITY

exposure and is handled separately. The planned special exposure account has an annual limit which has the same numerical value as the "regular" limits however, the lifetime limit for planned special exposures is five times the annual limits. Once a person receives five times the regular annual limit in their planned special exposure account , they will no longer be able to receive any more planned special exposure.

Example: A worker could receive up to 5 rem from their "regular" account and an additional 5 rem from their planned special exposure account during one year. They could do this for a total of five years, but at the end of the fifth year they will have exhausted their planned special exposure account and would no longer be able to receive any more planned special exposure.

The planned special exposure was primarily a provision provided in 10CFR20.1206 to accommodate vendors who have workers with highly specialized skills. Apparently the vendors felt that an annual 5 rem limit was too restrictive for them to conduct their business. Although this provision is also available to Duke Power, it is expected that the use of this provision will be rare.

There is the possibility that an **emergency situation** can arise where immediate action is required to protect valuable property, save lives, or protect large populations. The conditions governing the use of the Emergency Exposure directive are:

- Emergency situations where it is necessary to protect valuable property.
- Emergency situations where it is necessary to save lives or to protect large populations.
- The Emergency Exposure directive is not to be used for Planned Special Exposures.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
131	List the TEDE dose limits for Emergency Exposure.	X	X	X	X	X
132	State the approval required prior to the use of the Emergency Exposure dose limits.	X	X	X	X	X

Objective #131

The dose limits for Emergency Exposure are provided in the table below.

Emergency Worker Exposure Limits (a)

Activity	**Total Effective Dose Equivalent (TEDE)**	Lens of the Eye	Other Organs (b)
All	**5 rem**	15 rem	50 rem
Protect valuable property	**10 rem**	30 rem	100 rem
Lifesaving or protection of large populations	**25 rem**	75 rem	250 rem
Life saving or protection of large population (c)	**> 25 rem**	> 75 rem	> 250 rem

(a) Excludes declared pregnant workers

(b) Includes skin and body extremities

(c) Only on a volunteer basis to persons fully aware of the risks involved

Objective #132

Prior to the use of the **Emergency Exposure** dose limits, the following approvals (written or verbal) are required:

- Radiation Protection Manager or designee
- Emergency Coordinator or EOF Director

The person(s) who is/are to receive the dose must sign that they have been informed of the potential dose they will receive, have been fully briefed on the task to be accomplished and the risks of this exposure.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
134	Discuss the way in which exposure to neutron radiation is considered for the following: • Duke's administrative policy requirements for neutron dosimetry for entering the RCA. • How neutron dose accumulation is tracked between the times that the individual's TLD is read.	X	X	X	X	X
135	State the goals and efforts of the ALARA Program.	X	X	X	X	X
136	For the following areas: • Define and describe MNS unrestricted area • Describe MNS Restricted Area • Describe MNS RCA	X	X	X	X	X

Objective #134

Regulatory Guide 8.14 requires a personnel **neutron** dosimeter if the neutron dose equivalent is likely to exceed 100 mrem in a quarter. Duke Power has an administrative requirement which requires all personnel entering the RCA to wear a TLD (which measures neutron dose equivalent). Estimation of neutron exposure is a method used to temporarily track exposure until the TLD is processed. Estimated neutron exposure tracking for personnel is required if the neutron dose equivalent is likely to exceed 10 mrem per entry or per job if consecutive multiple entries are required. There are two methods used to estimate neutron exposure:

- One method is to measure the neutron dose rate and then calculate the exposure based on stay time.
- The second method is to determine the gamma exposure dose and neutron exposure dose for the given area. If it is determined that the neutron to gamma ratio is essentially constant during the period of personnel exposure, then a gamma/neutron ratio can be utilized. The gamma dose received can be ratioed to find the neutron dose received.

Objective #135

ALARA is a philosophy aimed at the minimizing exposure thru a management commitment. The goals and efforts of the McGuire Nuclear station Program are simple:

- To maintain the annual dose to each individual ALARA
- To maintain the collective dose (total person-Rem) ALARA
- Both points have to be considered simultaneously, as one without the other is not ALARA.

3.5. Radiation Areas and Access Controls

Objective #136

Unrestricted Area - an area, access to which is neither limited nor controlled by the licensee. At McGuire this is the area out side the protected area fence.

Restricted Area - an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. The restricted area at McGuire is enclosed by the protected area fence.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
137	State the dose limit associated with an unrestricted area.	X	X	X	X	X
138	List the requirements for wearing dosimetry devices inside and outside the RCA.	X	X	X	X	X
139	State the additional controls placed on entry/access to: • High Radiation Area • Very High Radiation Area • Extra High Radiation Area	X	X	X	X	X
140	Explain when a RWP is required.	X	X	X	X	X

Radiation Control Area (RCA) - that portion of the restricted area in which additional controls are imposed for radiation protection purposes. The RCA at McGuire is the Auxiliary Building , the Fuel Building, the Reactor Building and the Solidification Pad.

Objective #137

Each licensee shall conduct operations so that the dose in any **unrestricted area** from external sources does not exceed 2 mrem (0.02 mSv) in any one hour.

Objective #138

All personnel who will enter the Radiation Control Area (RCA) or a Radiation Control Zone (RCZ) in the restricted area of the station shall be issued and required to wear TLDs and Self reading dosimeters (SRDs) or an Electronic Dosimeter (EDs) when in these areas.

Objective #139

The High Radiation Area, Extra High Radiation Area and Very High Radiation Area have addition controls as follows:

- High Radiation Area: - This area must be accessed with an alarming dosimeter or RP escort. Individuals must leave the area immediately if dosimeter alarms continuously and contact RP.
- Extra High Radiation Area - This area must be locked or guarded 24 hours/day. In areas that can not be locked, a flashing yellow light is used as a warning. Personnel entering this area must have Dosimetry and be accompanied by a qualified RP technician. If access to the area is secured with a chain and padlock, the access must remain unlocked while personnel are inside and someone must guard the access to prevent unauthorized entry.
- Very High Radiation Area - The posting will have the words Grave Danger. This area must be locked at all times. The fuel transfer tube and the incore sump below the reactor vessel during incore detector thimble guide tube withdrawal are examples of very high radiation areas.

Objective #140

An RWP or SRWP is required for any work or activities performed inside the RCA or an RCZ. From a radiological stand point, the RWP/SRWP provides the person with the authorization to enter a radiological controlled area.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
141	Discuss how long a RWP or SRWP is valid.	X	X	X	X	X
142	Correctly interpret the information on the Daily Dose Report.	X		X	X	

Objective #141

The RWP is issued for a particular job, in a particular area and on a specific piece of equipment or on a specific component/system. The length of the time a RWP is valid is:

- For the duration of that job.
- If radiological conditions require a change in Radiation Protection requirements for longer than one shift, a new RWP shall be issued with the updated information.
- A change that exists for one shift or less may be communicated to the workers and noted in the RP Shift Log book in lieu of reissuing the RWP.
- Active RWPs are revised at the end of the year.

A SRWP is valid for 6 months or when conditions change to such an extent to warrant a change of the permit.

Objective #142

In the event that the Electronic Dose Capture System becomes inoperable, manual entry of information on individual dose cards will be required. A Daily Dose Report printout, which contains the current dose information for each individual, will be available at the Control Point. Drawing 7.1 provides an example of how a Daily Dose Report looks. The report is self explanatory with the codes for Exemption Class (EXP CLS) and the Status located on the front page of the report. The codes are listed below :

Exposure class:
- Class 1 - Limited Exposure
- Class 2 - Reduced exposure (Pregnancy)
- Class 3 - Active Radiation Worker
- Class 7 - Limits restricted by employer (no Planned Special Exposure)
- Class 9 - Current year record only. (no Planned Special Exposure)

Status codes:
- A - Alert - indicates an individual has exceeded 80% of the administrative limit
- E - Exclude - indicates an individual has exceeded 90% of allowable limit and worker is "excluded" from further work in the RCA, unless an exposure extension is granted.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
143	State the purpose of protective clothing.	X	X	X	X	X
144	Describe the general rule of thumb for radioactive spills.	X	X	X	X	X
145	Describe the three radiological problems that may result from a fire in areas where radioactive materials are stored and/or used.	X	X	X	X	X

3.6. Operational Aspects

Objective #143

The purpose of protective clothing is to protect personnel from getting contamination on their skin or clothing.

Objective #144

The general rule of thumb for spills is the SWIMS concept:

1. **S**top the spill - by righting the container; isolating the system if possible.
2. **W**arn others - Notify other personnel in the area or who may be affected by the spill and Radiation Protection.
3. **I**solate the Area - through the use of barricades, signs, etc.
4. **M**inimize exposure and contamination spread - Using only personnel necessary for cleanup and utilizing absorbent materials at the outer edges.
5. **S**ecure unfiltered exhaust and all supply ventilation in cases where airborne contamination may be a problem.

Objective #145

Fires within areas where radioactive materials are stored and/or used can pose the following radiological problems:

- Inhalation of airborne radioactive materials: The burning of materials containing radioactive materials can release radioactive materials into the air.
- Exposure to high levels of external radiation: Fires in high radiation areas can result in the fire fighters having to spent a significant amount of time in the area ensuring the fire is extinguished.
- Contamination of skin, clothing , and wounds and of equipment used in combating the fire.

INSTRUCTOR ACTIVITY

	OBJECTIVE	NLO	NLOR	LPRO	LPSO	LOR
146	State why vacuum cleaners used within the RCA are controlled.	X	X	X	X	X
147	State the immediate notification requirements listed in 10CFR20.2202 Notification of incidents.		X	X	X	X

Objective #146

Vacuum cleaners can create a significant radioactive airborne problem in RCAs or RCZs if they are not properly sealed at fittings, if the exhaust is not properly filtered or if proper controls/procedures are not employed when removing the bag for disposal. In addition, the concentrating of radioactive materials in the vacuum cleaner bag could result in high dose rates for personnel using the vacuum cleaner or performing maintenance on it. For these reasons, vacuum cleaners to be use in RCAs or RCZs are controlled to ensure the vacuum cleaners are properly maintained and personnel using them have received training on their use. Vacuum cleaners must be logged out to an accountable individual who has completed training. The accountable individual is responsible to ensure that only trained personnel use the vacuum cleaner.

Objective #147

10CFR20.2202 requires **immediate notification** by the licensee for any event involving byproducts, source, or special nuclear material possessed by the licensee that may have caused or threatens to cause any of the following conditions:

1. An individual to receive:
   - A total effective dose equivalent of 25 rem (0.25 Sv) or more,

   or
   - An eye dose equivalent of 75 rems (0.75 Sv) or more,

   or
   - A shallow-dose equivalent to the skin or extremities of 250 rads (2.5 Grays) or more

   or

2. The release of radioactive materials, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake of five times the annual limits on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

INSTRUCTOR ACTIVITY

Twenty-four hour notification: Each licensee shall, within 24 hours of discovery of the event, report any event involving loss of control of licensed material possessed by the licensee that may have caused, or threatens to cause, any of the following conditions:

1. An individual to receive in a period of 24 hours:
   - A total effective dose equivalent exceeding 5 rems (0.05 Sv)

   or
   - An eye dose equivalent exceeding 15 rems (0.15 Sv)

   or
   - A shallow-dose equivalent to the skin or extremities exceeding 50 rems (0.5 Grays)

   or
2. The release of radioactive materials, inside or outside of a restricted area, so that , had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (the provisions of this paragraph do not apply to locations where personnel are not normally stationed during routine operations, such as hot-cells or process enclosures).

4.0 TECHNICAL SPECIFICATIONS

Tech Spec Section 6.0 Administrative Controls requires the development and implementation of and adherence to the Radiation Protection Program. This also includes records and reports required to document adherence and/or deviations. From a day to day Operations standpoint the following are of particular interest:

Tech. Spec. 6.2.2c requires: "A Radiation Protection Technician shall be on site when fuel is in either reactor." Any deviation from the above guideline shall be authorized by the Station Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Routine deviations from the above guideline is not authorized.

Tech. Spec. 6.11 Radiation Protection Program: "Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR20 and shall be approved, maintained and **adhered to** for all operations involving personnel radiation exposure."

INSTRUCTOR ACTIVITY

Tech. Spec. 6.12 High Radiation Area: This Tech Spec. provides alternatives to the control and alarm devices required by 10CFR20 for High Radiation Areas. It also provides exemption requirements for issuance of RWPs prior to entry into High Radiation Areas.

5.0 INDUSTRY EVENTS

The following events occurred at McGuire Nuclear Station which resulted in contamination of personnel or equipment, added personnel dose and/or excessive man-hours to decon personnel or equipment.

PIP 1-M91-0428 Horseplay resulting in contamination of individuals:
While deconning the lower containment pipe chase, one worker threw a mophead at another and the water hit her in the face. The worker did not stop to have her face checked. Later she was found to have a contaminated face.

Corrective action: The worker was deconned and counseled on the responsibility to have potential contamination checked early. Also they were counseled on the proper use of PCM's.

PIP 2-M92-0356 Failure to follow SRWP results in contamination of individual
Two I&E Technicians were assigned the PM on ZMP5180. SRWP92-20 required the workers to notify RP prior to beginning work in the RCA, however they failed to do this. In addition. one worker failed to wear glove as required by the SRWP. Unexpected condition, i.e., the presence of boron was not brought to the attention of RP resulting in a personnel contamination and an extremity skin dose of 225 mrem.

Corrective action: The area was deconned to prevent further contamination's. Use of gloves and notification of RP prior to work was stressed with the individuals.

PIP 2-M92-0358 Torn protective clothing results in contamination of individual
Worker tore glove while working in highly contaminated area and did not stop work or notify RP. They eventually spread the contamination to other parts of their body.

Corrective action: The Worker was deconned and counseled by RP that they should stop work, leave the contaminated area and contact RP if they discover that they have torn/damaged personnel protective clothing.

INSTRUCTOR ACTIVITY

PIP 2-M93-0685 Work beyond scope of RWP results in personnel contamination. Operator "A" was working with the fuel handling crew as they were trying to remedy a problem with the fuel handling mast. At one point he was asked by the fuel handling crew to shake the cables attached to the mast in an attempt to fix the problem. This evolution was not addressed in Operator "A's" RWP #2804. This was also the probable source of the Hot Particle contamination that was found on Operator "A's" cheek after exiting containment.

Corrective action: Both Unit #1&2 Refueling Operation RWPs were amended to require wearing an extra set of gloves and a face shield when handling or touching the cables on the fuel handling bridges.

PIP 2-M93-1019 Excessive pressure in the NCDT results in contamination of floor. Unit #2 was coming out of an outage to plug S/G tubes. During the fill and vent of the NCS, overpressurization of the NCDT caused "A" and "C" NCPs to leak water to the containment floor through #3 seal. This resulted in contamination of the floor below the pumps which in turn caused excessive exposure and man-hours to clean up the water and boron.

Corrective action: A Limit and Precaution was added to OP/1&2/A/6500/01 to warn operators that excessive pressure in the NCDT can result in water spilling out of #3 seal into containment.

6.0 SUMMARY

Review Operations objectives

INSTRUCTOR ACTIVITY

7.0 DRAWINGS

7.1. Daily Dose Report (07/08/96)

DAILY DOSE REPORT PAGE : 48

MCGUIRE NUCLEAR STATION DATE : 07/05/96

WORK GROUP NO : 20

WORK GROUP : ESS- GENERAL SERV TIME : 00:12

NAME	RP NO	EXP CLS	STA	TEDE (MREM)			STATUS	DAC HOURS	STATUS	SHALLOW (MREM)	STATUS	LDE (MREM)	STATUS	CEDE (MREM)	STATUS
				MAE	USED	LEFT		7 DAYS				Y- T -D		Y -T-D	
STEWARD, CHRIS R.	403450	1	MC	2000	284	1716		.00		308		284		0	
STEWARD, HARVEY S.	400387	1	MC	2000	14	1986		.00		24		29		0	
SURRATT, STEVE L.	400888	1	MC	2000	32	1968		.00		36		35		0	
TILLMAN, CHRIS B.	300876	3	MC	2000	4	1996		.00		4		4		0	
TOWERS, BILL E.	150087	3	MC	2000	1838	162	E	.00		722		647		0	
TRAVIS, JOE N.	160987	1	MC	2000	49	1951		.00		53		52		0	
TRULL, ROBERT, G.	143870	3	MC	2000	15	1985		.00		17		16		0	
TULLEY, JOHN R.	153243	3	MC	2000	23	1977		.00		26		23		0	
TWINKLE, BOB K.	023524	3	MC	2000	12	1988		.00		13		13		0	
UPJOHN, PHIL H.	007832	3	MC	2000	470	1530		.00		515		472		0	
UPRIGHT, TOM K.	098730	3	MC	2000	171	1829		.00		197		176		0	
URICH, KEN, C.	283945	3	MC	2000	154	1846		.00		172		154		0	
WALL, GREG M.	167432	3	MC	2000	136	1864		.00		145		136		0	
WALTON JOHN B.	134865	3	MC	2000	49	1951		.00		54		049		0	
WALTON, SUSAN A.	005673	2	MC	500	39	461		.00		39		0		0	
WARDEN, JOE S.	000739	1	MC	2000	9	1991		.00		9		9		0	
WATERS, DOLLIE P.	439092	3	MC	2000	25	1975		.00		27		25		0	
WELLER, WESS W.	356424	3	MC	2000	1742	258	A	.00		688		665		0	
WHITE, LARRY R.	333562	3	MC	2000	162	1838		.00		174		162		0	
WILKINSON, CHUCK K.	267847	3	MC	2000	15	1985		.00		15		15		0	
WILLIAMS, HOMER T.	176904	3	MC	2000	115	1885		.00		128		119		0	
WILSON, STEVE K.	187384	3	MC	2000	66	1934		.00		74		66		0	
WONG, CHU C.	173824	3	MC	2000	69	1931		.00		69		69		0	
YELSON, BORIS K.	168363	3	MC	2000	91	1909		.00		98		91		0	

TOTAL EMPLOYEES FOR WORK GROUP 198

TOTAL EMPLOYEES FOR STATION 1579

INSTRUCTOR ACTIVITY

8.0 RADIATION WORKER AND RESPIRATORY TRAINING STUDENT GUIDE

Following this page is the General Employee Training Radiation Worker and Respiratory Protection Student Guide.

APPROVAL PROCESS RECORD

REV. __________ DATE __________

INSTRUCTOR ______________________________ DATE __________

ELECTRONIC COPY

[1]REVIEWED BY SIGNATURE COPY ON FILE DATE __________

APPROVED BY ______________________________ DATE __________

NOTE 1: If no review is performed, place N/A in the REVIEWED BY section.

SRO Question # 15 Bank Question: 1053 *Answer: A*

1 Pt(s) Given the following conditions on Unit 1:

- Mode 3
- NC System is at 1700 psig and 450 degrees
- In process of cooling down and depressurizing the NC System
- Safety Injection has occurred
- NC Pressure going down in an uncontrolled manner
- Containment pressure going up in uncontrolled manner

Which one of the following describes the proper procedures to mitigate the above?

A. Enter AP/35 (ECCS Actuation During Plant Shutdown) and then go to E-0 (Reactor Trip or Safety Injection).

B. Enter E-0 and then go to AP/35

C. Enter AP/35 and then go to AP/34 (Shutdown LOCA)

D. Enter E-0 and then go to E-1 (Loss of Reactor or Secondary Coolant).

Distracter Analysis:

A. Correct.
B. Incorrect: Plausible.
C. Incorrect: Plausible.
D. Incorrect: Plausible.

LEVEL: SRO

KA: 006 G 2.4.4 (4.0/4.3)

SOURCE: NEW

LEVEL OF KNOWLEDGE: ANALYSIS

AUTHOR: CWS

LESSON: - AP/1/5500/35 Background Document

OBJECTIVES: OP-MC-AP-35 Obj. 1
MC-AP-34 Obj. 1

REFERENCES: AP/1/5500/35 Background Document pages 2-4
AP/1/A/5500/35 page 3
AP/34 Background Document page 2

Comment

The plant conditions stated in the question do not provide the candidate enough information to accurately determine the initial plant conditions at the time of the safety injection (SI) actuation. The plant conditions given in the question are possible whether SI occurred at 2000 psig or at 1700 psig. This information is important because the proper operator response is dependent upon the plant conditions at the time SI occurred. In addition, the different grammatical tense of the verbs (both present and past tense) used in the description of plant conditions contributed to the different interpretations of the question by the candidates.

Proposed Resolution

The facility recommends that answers "A" and "D" both be accepted as a correct answer to the question.

Justification

Answer "A" is the correct answer if the candidate interpreted the given plant conditions to mean that SI occurred at 1700 psig, which is below the P-11 permissive setpoint (1945 psig). With this interpretation of plant conditions, entering AP/35 and transitioning to E-0 is the proper procedure flow path.

Answer "D" is the correct answer if the candidate interpreted the plant conditions to mean that SI occurred above the P-11 permissive setpoint and that plant conditions have degraded to those stated in the question. With this interpretation of plant conditions, entering E-0 and subsequently transitioning to E-1 is the proper procedure flow path.

The station procedure reference for this question and comment is provided in this comment package:

- OMP 4-3, Revision 21, Use of Abnormal and Emergency Operating Procedures

Duke Power Company McGuire Nuclear Station **Use of Abnormal And Emergency Procedures** **Information Use**	Document No. **OMP 4-3** Revision No. **021** Electronic Reference No. **MP0070PK**

Revision History (significant issues, limited to one page)	
Rev 021	(02/25/02) Converted to standard template.
Rev 020	08/23/00 Added description of how to "validate" subcriticality status tree using WR neutron flux indication.
Rev 019	Section 7.9 (Abnormal Operation Not Covered by Procedure) was moved to Section 9 (Deviation from Approved Procedures). Clarifications were made in this section so operators know what type of procedure deviations require implementing 10CFR50.54x and 10CFR50.72. Minor human factor enhancements made to Section 7.4, 7.5, 7.18, and 9.1. Clarification made in Section 7.15.1.6 on completion of orange and red path EPs.

Use of Abnormal and Emergency Procedures

1. Purpose

To provide guidance to the operations group in the use of abnormal and emergency procedures.

2. References

2.1 Nuclear System Directive NSD-703 (Administrative Instructions for Station Procedures)

2.2 Nuclear Policy Manual

2.3 McGuire Technical Specifications

3. Description

3.1 This procedure provides instruction in the following areas:

3.1.1 Control of approved procedures

3.1.2 Use of approved procedures.

3.1.3 Completion of a procedure.

3.1.4 Deviation from approved procedures.

4. Responsibilities

4.1 The Superintendent of Operations, or designee, shall ensure:

4.1.1 All shift crews are consistently implementing emergency and abnormal procedures per this procedure.

4.1.2 This procedure is revised to reflect current NRC/Industry standards for shift crews implementing emergency and abnormal procedures.

4.1.3 The McGuire Training Center Operations' instructors consistently teach and critique the licensed operators per the guidance in this procedure.

4.2 Operations Shift Manager or Shift Supervisor

4.2.1 Ensures personnel follow procedures.

4.2.2 Ensures operators are qualified to perform the required procedure.

4.2.3 Reviews and approves completed procedures.

4.3 All Personnel who use procedures.

4.3.1 Must follow procedures in compliance with NSD-703 (Administrative Instructions for Station Procedures) and this OMP.

4.3.2 Contact the EP/AP procedure staff to correct procedural inadequacies as they are discovered.

4.4 All licensed operators shall memorize immediate actions of emergency and abnormal procedures.

4.5 Shift Support Assistant

4.5.1 Replenishes Working Copy file.

4.5.2 Ensures completed AP's and EP's are routed to the Operations Work Process Manager (or designee) for routing to Master File.

4.5.3 Makes new Control copies if a Control Copy has been used.

5. Reporting Requirements

None

6. Control Of Approved Procedures

6.1 Control Copy Procedures

6.1.1 A set of approved Control Copy Procedures will be kept in the Control Room. Control Copy Procedures may also be stored at local stations as authorized by the Operations Support Manager.

6.1.2 Control Copy procedures should **NOT** be removed from their authorized area except to make working copies.

6.1.3 EPs and APs are used during emergency situations and require immediate access. The Control Copy of the procedure will be used to perform the steps. The procedure group should be contacted after the use of any EP or AP. The Control Copy of any EP or any AP should be replaced by the Shift Support Assistant.

6.2 Working Copy Procedures.

6.2.1 Working Copies of EP's and AP's are maintained in a file near the Control Room. Use of these Working Copies shall be in accordance with NSD-703 (Administrative Instructions for Station Procedures) and the instruction provided in this procedure. Colored folders contain Working Copies that are used to make additional Working Copies when the files reach a minimum. The Working Copy used to make additional copies shall be verified against the Control Copy prior to copying. **WHEN** the last Working Copy (located in front of the colored folder) is used, the colored folder should be pulled and placed in the "Folders to be Copied" basket. Working Copies are normally used for training or making new Control Copies only.

6.3 Periodic Review of Approved Procedures.

6.3.1 All approved EP's and AP's shall be reviewed at intervals **NOT** to exceed six (6) years.

6.3.2 The review shall ensure the adequacy of the procedures as well as determining that valve numbers are current.

6.3.3 This review shall be done by a cognizant and responsible individual (who is a Qualified Reviewer) as designated by the Superintendent of Operations.

6.3.4 These reviews will be scheduled and documented as a PT.

7. Use Of Approved Procedure

7.1 EP/AP Usage

Generally, entry into the emergency procedure set is limited to two conditions:

- **IF** a safety injection or reactor trip occurs or is required with initial conditions above P-11, the operator will enter EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection). (During a normal plant heatup, selected rods may be withdrawn as available source of negative reactivity insertion. **IF** these rods are dropped with initial conditions below P-11, most of the EP steps do **NOT** apply, so implementing EPs is **NOT** required. Abnormal procedures dealing with reason rods were dropped should provide adequate guidance to address this situation.)

NOTE: The following reactor trips do **NOT** require entry into E-0:

- Control rod drop tests performed at power levels below 5% full power.
- Trip was initiated and specifically called for in an in progress test procedure or was part of a planned shutdown.
- Trip with initial conditions less than P-11 as discussed above.

- **IF** a complete loss of power on both emergency busses takes place, the operator will enter EP/1,2/A/5000/ECA-0.0 (Loss of All AC Power). This includes any time during the performance of any other EP.

During periods when EPs are **NOT** being implemented, the SPDS and critical safety function status trees may be used to determine or identify abnormal conditions. Emergency procedures referenced by them may be used to correct the alarming condition.

7.2 Use of Control Copies:

Since EPs and APs are used during emergency situations and require immediate access, the Control Copy of the procedure will be used to perform the steps. The Control Copy of any EP or AP should be replaced by the Shift Support Assistant. The procedure group should be contacted after the use of any EP or AP.

7.3 Abnormal Procedure Usage:

Entry into an abnormal procedure is based on an evaluation of symptoms referenced in the respective procedure. There are times when symptoms exist that are **NOT** at the level that the applicable abnormal procedure would be implemented. Those situations would be addressed by existing operating and/or alarm response procedures. **WHEN** more than one AP is applicable at the same time, the Control Room SRO will determine how many procedures can be implemented at a time and their priority based upon manpower available and ability to maintain control of the situation running more than one procedure at a time.

7.4 Immediate and Subsequent Actions

Immediate actions are to be performed immediately upon the recognition of the event. Immediate actions should be performed prior to the SRO reading them. **WHEN** the SRO does read the immediate action, the RO will ensure that all the requirements of the step have been fully implemented.

Subsequent action steps should be performed as the SRO reads the steps. ROs should **NOT** perform steps prior to the SRO reading the step. This practice can mask "root cause" symptoms of the event in progress, especially when performing EP/1(2)/A/5000/E-0 (Reactor Trip or Safety Injection).

The only subsequent actions that can be taken prior to procedure direction are the following:

1. CA may be reset and controlled to a S/G when that S/G reaches its normal level setpoint.

2. Known leaks may be isolated. Note that if procedure covers isolation of leak, it is usually preferable to wait until procedure step is reached. Licensed operator may isolate leak prior to reaching step only if plant conditions require it to be done.

3. NC Pumps may be tripped on an Sp signal due to the loss of support functions unless the procedure has directed otherwise.

7.5 Manual Initiation of Safeguards Actions

ROs and SROs are expected to manually initiate safeguards actions if an automatic action setpoint is being approached, to avoid challenging the automatic safeguards function. An example of this is to manually initiate safety injection if pressure is decreasing in an uncontrolled manner to 1845 psig.

However, during an ATWS, it is undesirable to initiate S/I in "anticipation" of an S/I signal if the reactor will **NOT** trip, since this will cause a loss of CF flow to the S/Gs. This exception is stated in the APs that manually initiate S/I in "anticipation" of an S/I signal.

The operator is expected to manually initiate any action which should have automatically occurred if the automatic function fails, such as the Safety Injection fails to initiate during an uncontrolled Reactor Coolant depressurization at 1845 psig (even during an ATWS) or an ECCS pump fails to start on a Safety Injection signal.

IF directed to initiate a signal, initiate both trains unless otherwise specified.

7.6 Resetting Safety Systems

IF directed to reset a signal, reset both trains unless otherwise specified.

IF a procedure directs resetting a signal that has **NOT** been received or that has been previously reset, the reset pushbuttons do **NOT** have to be depressed since the intent of the step has been met. Likewise, if a procedure directs the operator to stop, start or reposition a component which is already in the desired position; the component's control switch does **NOT** have to be depressed.

7.7 Blocking of Automatic Safety Actuations

It is the policy of Operations to **NOT** block any automatic safety actuation from performing its intended function. The RO or SRO has the option to violate this policy if, in his/her judgment, plant conditions require it to be done. Blocking an automatic safety actuation should only be done to better protect the health and safety of the public or to protect the lives of plant personnel. Non-procedural blocking of automatic safety actuations must be approved prior to taking the action by two licensed personnel, one of whom is a supervisor who holds an SRO license.

WHEN an automatic safety actuation is blocked outside of procedure, the action must be reported to the NRC under the provisions of 10CFR50.54x.

In the event that an RO/SRO blocks an automatic actuation of a safety system, it is his/her responsibility to be fully aware of all associated plant parameters, and to manually actuate safety equipment as necessary to assure safe plant conditions.

The following is a list of automatic safety signals that can be "blocked" from the main control board:

- Feedwater Isolation on Reactor Trip and Lo Tave
- Pressurizer Lo press. SI (Below P-11)
- Low Pressure Steamline Isolation (Below P-11).
- CA Auto Start on Lo Lo S/G Level or both CF pumps tripped (Below P-11)
- S/R Hi Flux Reactor Trip
- I/R Hi Flux Lo Setpoint Reactor Trip
- P/R Hi Flux Lo Setpoint Reactor Trip
- St Interlock Bypass on NC Sample valves: NM-22, NM-25, NM-26

These signals should **NOT** be "blocked" except under the direction of an approved station procedure or to better protect the health and safety of the public or to protect the lives of plant personnel.

7.8 A.T.W.S.

An A.T.W.S. (Anticipated Transient Without Scram) is defined in 10CFR50.62 as an anticipated operational occurrence followed by the failure of the reactor trip portion of the protective system. An anticipated operational occurrence is defined in 10CFR50, Appendix A, as those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are **NOT** limited to loss of power to all NC pumps, tripping of the turbine generator, isolation of the main condenser and loss of all offsite power. Clearly, to have an A.T.W.S. there must be transient followed by a failure of the reactor trip breakers.

Instrument failures, by themselves, are **NOT** necessarily transients. For example, if one channel of Pressurizer Pressure was out of service for preventive maintenance (bistable in tripped condition) and another Pressurizer Pressure channel failed (**NOT** the controlling channel), a reactor trip signal would be generated. **IF** the reactor failed to trip, this would be a failure of the reactor trip breakers and the automatic trip features of the reactor protection system and **NOT** an A.T.W.S. event. Obviously, the control operators would have to recognize and check that the channel failure was indeed a channel failure by checking the other two Pressurizer Pressure channels in this example. This would, however, force Operations to shutdown the affected unit to at least Hot Standby per Tech Specs.

7.9 Adverse Containment Setpoints

Many setpoints in the EP's are presented in a dual format with a second setpoint enclosed in parentheses. This second setpoint is used to account for the additional error in the setpoint due to the containment environment following a high-energy line break. The setpoint in parentheses will be used whenever containment pressure has exceeded 3 psig.

7.10 Two-Column Format

The pages for presentation of operator action steps will use a two-column format. The left-hand column is designated for expected operator action and response, and the right-hand column is designated for contingency actions when the expected response is **NOT** obtained.

7.10.1 Left Hand Column

Each step in the left-hand column contains a high-level statement which describes the task to be performed. **IF** the high-level step requires multiple actions, then substeps are specified. Following each step or substep, the expected response or result is given in capital letters, separated from the step by a dash. (Expected responses are **NOT** supplied for simple control manipulations or actions.)

IF a sequence of performance is important, then the substeps are designated by letters (or numbers if finer detail is provided). **IF** sequence of performance is **NOT** important, the substeps are designated by bullets (•). In either case, there is an understood "and" between each substep such that all substeps must be true to proceed in the left-hand column.

7.10.2 Action Steps

Action steps are written so that you proceed directly down the left-hand column only. **IF** however, the expected result or response is **NOT** obtained or the action **CANNOT** be performed, move to the right-hand column for contingency instructions. This column is titled "Response Not Obtained". **IF** one contingency action is appropriate for any of a series of left-hand column substeps, it is simply stated once as a high-level contingency. **IF** a contingency is **NOT** provided or is **NOT** successful, proceed to the next step or substep in the left-hand column.

Example:

Check if main steamlines intact: ____• All S/G pressures - STABLE OR GOING UP ____• All S/Gs - PRESSURIZED	**IF** any S/G pressure going down in an uncontrolled manner OR depressurized, THEN: ____a. Check if all faulted S/Gs have been previously isolated by EP/1/A/5000/E-2 (Faulted Steam Generator Isolation ____b. **IF** faulted S/G(s) **NOT** isolated, THEN GO TO EP/1/A/5000/E-2 (Faulted Steam Generator Isolation).

The two-column format thus equates to logical terms which would otherwise be specifically stated: **IF** the conditions required in the left-hand column are **NOT** achieved, **THEN GO TO** the right-hand column for contingency instructions.

After taking the contingency action in the right-hand column, proceed to the next step or substep in the left-hand column. **IF** the contingency action **CANNOT** be performed or is **NOT** successful, and further contingency instruction is **NOT** provided, return to the next step or substep in the left-hand column.

Unless otherwise specified, a required task need **NOT** be fully completed before proceeding to the next step; it is sufficient to begin a task and have assurance that it is progressing satisfactorily. This ensures efficient implementation where steps are very time consuming.

It may be appropriate to return to a step if conditions change or if condition is subsequently diagnosed. Usually procedure loops through the procedures, the foldout page, or other continuous actions will cover subsequent completion of any required actions, but operators may back up in EP or AP based on operator judgment. This flexibility may be particularly important if action must be completed in a timely manner, and a continuous action is **NOT** in effect that would immediately perform action. **WHEN** backing up in an EP or AP, caution must be taken to ensure steps completed that are being backed over do **NOT** adversely affect procedure performance and that you are meeting the intent of the procedure.

EP/1,2/A/5000/ES-0.0 (Rediagnosis), is used as an operator aid and is entered based purely on operator judgment. It is intended to be used after departure from E-0, but only if SI has been actuated. It provides reassurance to the operator that he is in the correct Optimal Recovery Procedure, or provides the necessary transition instruction to get to the correct Optimal Recovery Procedure for the existing symptoms.

7.10.3 Continuous Action Steps

"Continuous action steps are identified by using **WHEN THEN, IF AT ANY TIME THEN**, control, monitor, and maintain. These steps, unless otherwise stated, shall apply until subsequent steps or procedures provide alternate guidance. Note that many **IF** AT ANY TIME steps that only apply to procedure in effect will state this limit in step. Foldout page also identifies continuous action steps; foldout page only applies to EP in effect."

7.10.4 Transitions

Transitions to other procedures or to different steps in the same procedure may be made from either column. Such transitions should be made realizing that preceding notes or cautions are applicable. Any tasks still in progress need **NOT** be completed prior to making a transition; however, the requirement to complete the tasks is still present and must **NOT** be neglected. Unless specified otherwise, a transition to an enclosure or to another procedure shall be made to the first step of the cited enclosure or procedure.

7.10.5 Use of Enclosures

The decision on whether to read or hand-off an enclosure will be based on SRO judgment depending on the event. The following are some general guidelines to help the SRO make this decision.

- It is usually preferable for SRO to read enclosure if:
  - The crew must wait for the enclosure to be completed in order to continue in the EP/AP.
  - No more ROs are available to continue in the EP/AP, unless RO can perform enclosure concurrent with performing other steps.
  - There are no more time critical actions to be performed.
- It is usually preferable for SRO to hand-off the enclosure if:
  - It is critical for the crew to continue in the body of the procedure in a timely manner.
  - It is a valve checklist.
  - Actions are outside the horseshoe.

Additionally, an enclosure will be handed off if procedure specifies to hand--off the enclosure or if it is the foldout page.

7.11 Place Keeping Aids

EPs and APs contain a single line to the left and adjacent to the step number. The line is provided as a placekeeping aid. Check step after it is READ (whether it applies or **NOT**), unless it is a step that must be followed up on later.

IF the step is a diagnostic step that requires transition to RNO, place right arrow (→) next to step in lieu of, or in addition to check mark.

Note that if you read an **IF/THEN** step that does **NOT** require performing its substeps ("**IF**"condition **NOT** met), don't check the substeps. The substeps will **NOT** be read or performed, and you only to check steps if you READ them. Check next to **IF/THEN** step if it is all that is read, whether it has a placekeeping line or not.

IF the step must be followed up on later (ex: **WHEN** or **IF AT ANY TIME** steps), perform the following:

7.11.1 Leave place keeping line blank until step is completed.

7.11.2 Circle step.

7.11.3 Mark the page using sticky paper, to ensure page is returned to as needed. (Pad of sticky paper will be in each EP and AP drawer).

7.12 Foldout Page

Many EPs and a few APs have been provided with a foldout page. This is located on the left-hand side of the Control Copy folder, and is intended to be visible when the procedure steps are being performed. The foldout page contains several pieces of information or actions which are applicable at any step in the procedure. Copies of the foldout page will be distributed to R0s (unless otherwise stated) when procedure calls for foldout page to be monitored.

7.13 Level of Detail

To allow efficient execution of the action steps in a procedure, all unnecessary detail has been removed. Much information which an operator is required to know (based on his training and experience) will **NOT** be included. **WHEN** an operator is given a task without all the details on how to accomplish that task (i.e. dump steam), the operator is expected to use knowledge to complete that particular task (i.e. block low pressure steamline isolation, reset main steam isolation and dump steam utilizing either the main steamline power operated relief valves or the main steam condenser dump valves).

7.14 Usage of a Configuration Control Card

7.14.1 A configuration control card shall be used in conjunction with EPs and APs as follows. Any time a component is positioned in the plant and the person doing the positioning does **NOT** have a procedure or R&R in his possession, a configuration control card shall be filled out.

7.14.2 Any component that is positioned on the control board to work around equipment failures or isolate leaks that are **NOT** specifically covered in the EP/AP shall be documented on the configuration control card. Example: AP-10 directs actions to isolate a leak if the source of the leak is known. This is a general statement which could lead to isolating one train of ND if it was a check valve leak. The closing of the valves and removal of power shall be recorded on a configuration control card.

7.14.3 The configuration control cards filled out in Steps 7.14.1 and 7.14.2 shall be handled per the following two situations:

- Without operational support center activation

  The configuration control card will be handled by Ops shift per OMP 7-1 (Removal and Restoration (R&R) Requirements).

- With operational support center activation

  WHEN the OSC is activated, Operations will report to the OSC and shall bring with them all configuration control cards that have been filled out. The cards taken to the OSC shall be given to the OPS SRO in the OSC. For handling cards in the OSC, refer to RP/0/A/5700/020 (Activation of the Operations Support Center).

7.15 Usage of Status Trees

There are six different trees, each one evaluating a separate Critical Safety Function of the plant. Color-coding of the status tree end points will be either red, orange, yellow, or green, with green representing a "satisfied" safety status. Each nongreen color represents an action level that should be addressed according to the Rules of Priority as discussed below.

The six Status Trees are always evaluated in the sequence:
- Subcriticality
- Core Cooling
- Heat Sink
- Integrity
- Containment
- Inventory

IF identical color priorities are found on different trees during monitoring, the required action priority is determined by this sequence.

Initial monitoring of the status trees should begin on either of the following conditions:

- As directed by an action step in EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection).

- **WHEN** a transfer is made out of the Safety Injection procedure to another EP.

An exception to this is that no status tree monitoring is required during the Loss of All AC Power EP since none of the electrically powered safeguards equipment can be used. **WHEN** power is subsequently restored, EP/1,2/A/5000/ECA-0.1 or 0.2 (Loss of All AC Power Recovery procedures) will direct the operator when monitoring of status trees is required.

7.15.1 Implementing CSF Path Procedures

7.15.1.1 CSF procedures are **NOT** to be implemented prior to transition from EP/1,2/A/5000/E-0 (Reactor Trip or Safety Injection). **IF** a CSF path is red or orange while the operating crew is in EP/1,2/A/5000/E-0, but has turned to green upon transition from E-0, the CSF procedure which was in alarm shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. **IF** there are any valid red or orange path CSF's on transition from E-0 (unless transition is to EP/1,2/A/5000/ECA-0 (Loss of All AC Power), the associated CSF procedure shall be implemented.

7.15.1.2 **IF** a valid red or orange path flickers into alarm on SPDS but returns to green prior to the crew validating the condition and implementing the procedure (implementation of procedure being that the SRO either hands out fold-out pages or starts reading from the procedure), the CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7. Likewise, if a valid red path or orange path goes into alarm during performance of a higher priority CSF procedure, but returns to green prior to transition from the higher priority CSF path procedure to the lower priority CSF procedure, the associated CSF procedure shall **NOT** be implemented. **IF** the CSF path is yellow, it shall be handled as any other yellow path procedure per Section 7.15.1.7.

7.15.1.3 **IF** a CSF procedure directs the operator to return to the procedure and step in effect, AND the corresponding status tree continues to display the offnormal conditions, the corresponding CSF procedure doesn't have to be implemented again, since all recovery actions have been completed. However, if the same status tree subsequently changes to a valid higher priority condition, (OR if it changes to lower condition and returns to higher priority condition again), the corresponding CSF procedure shall be implemented as required by its priority.

7.15.1.4 Red Path

IF any valid red path is encountered during monitoring, the operator is required to immediately implement the corresponding EP. Any recovery EP previously in progress shall be discontinued. **IF** during the performance of any red path procedure, a valid red condition of higher priority arises, the higher priority condition should be addressed first, and the lower priority red path procedure suspended.

7.15.1.5 Orange Path

IF any valid orange path is encountered, the operator is expected to scan all of the remaining trees, and then, if no valid red is encountered, promptly implement the corresponding EP. **IF** during the performance of an orange path procedure, any valid red condition or higher priority valid orange condition arises, the red or higher priority orange condition is to be addressed first, and the original orange path procedure suspended.

7.15.1.6 Completion of red or orange path procedure

Once procedure is entered due to a red or orange condition, that procedure should be performed to completion, unless preempted by some higher priority condition. It is expected that the actions in the procedure will clear the red or orange condition before all the operator actions are complete. However, these procedures should be performed to the point of the defined transition to a specific procedure or to the "procedure and step in effect" to ensure the condition remains clear. At this point any lower priority red or orange paths currently indicating or previously started but **NOT** completed shall be addressed.

FR-S.1, P.1 and Z.1 can be entered from either an orange or red path status. **IF** the color changes from orange to red while you are in one of these EPs, the crew should continue and complete the EP from where they are. Crew does **NOT** have to backup and restart the EP. **IF** you exit the orange path, and it subsequently turns red, the EP must be reentered at Step 1.

Upon continuation of recovery actions in Optimal Recovery procedure, some judgment may be required by the operator to avoid inadvertent reinstatement of a Red or Orange condition by undoing some critical step in the Function Recovery procedure. The Optimal Recovery procedures are optimal assuming that safety equipment is available. The appearance of a Red or Orange condition in most cases implies that some equipment or function required for safety is **NOT** available, and by implication some adjustment may be required in the Optimal Recovery procedure.

7.15.1.7 Yellow Path

A yellow path does **NOT** require immediate operator attention. Frequently, it is indicative of an off-normal and/or temporary condition which will be restored to normal status by actions already in progress. In other cases, the yellow status might provide an early indication of a developing red or orange condition. The operator is allowed to decide whether or **NOT** to implement any yellow path procedure.
Implementation of a yellow path function restoration guideline is based on operator judgment when it is determined that adequate time exists to implement it. In other words, the operator does **NOT** have to implement a yellow path guideline if a judgment has been made that it is inappropriate based on available time or current plant state; and if an event of higher priority is in progress, the operator should attend to the more important matters prior to implementing a yellow path function restoration guideline. In the prioritization scheme in the EPs, the optimal recovery procedures (including applicable foldout pages) have priority over the yellow path function restoration procedures. The yellow path procedure can be considered as a supplementary set of actions that were provided to address one parameter being in an off-normal state. The controlling guideline in effect is the optimal recovery procedure that the operator is in when he decides that he has enough time to perform the yellow path procedure actions. **While performing the actions of the yellow path, continuous actions or foldout page items of the optimal recovery procedure in effect are still applicable and should be monitored by the operator.** This concurrent procedure usage should **NOT** cause the operator any difficulties since yellow path procedures are only performed when adequate time exists.

For example, if the operator is in ES-1.1 (SI Termination) and decides to implement FR-H.5 because of low SG level and NC subcooling is lost while in FR-H.5, the operator should terminate FR-H.5 and implement the action of the ES-1.1 foldout page to reinitiate SI flow.

7.15.1.8 SPDS

Normally, the condition of the status trees is continuously monitored and displayed by the OAC. The OAC can be used to check any off normal alarm and to determine which EP to implement. The entire control room crew is responsible for monitoring the SPDS.

SPDS indication must be validated using reliable control board indicators prior to implementing CSF procedure.

7.15.1.9 How Long to Monitor Status Trees

Monitoring of status trees may be stopped when any of the following are met:

- Cold shutdown and TSC concurrence
  OR
- Transition to normal recovery procedure (OP).
  OR
- Transition to a Severe Accident Mitigation Guideline (SAMG).

7.16 Selected Definitions

Some words used in the emergency procedures have unique meanings. These unique meanings should be understood based upon training and experience or by the specific use of the word in the context of the step being performed. Some words with unique meanings are listed below:

Check	-	to determine present status. (no action)
Ensure	-	to take necessary actions to guarantee that the component or reading is as specified. (Local actions in EPs and APs are only required if step specifies to dispatch personnel though).
Faulted	-	refers to a steam generator that has a secondary break.
Ruptured	-	refers to a steam generator that has a primary to secondary leak (SGTR).
Implement	-	begin a required program or series of procedures.
Intact	-	refers to a steam generator that is **NOT** faulted or ruptured and is available as a heat sink.

GO TO	-	discontinue use of present procedure and stay in the referenced procedure. The referenced procedure is always entered at the first step unless otherwise specified.
REFER TO, PER		user is directed to a supplemental procedure/enclosure for actions but will remain in the controlling procedure.
Stable	-	Maintained steady. **IF** a parameter is being controlled within a desired range, or if a slight trend in either direction is occurring, operator judgment may be used to determine if parameter is considered stable.
Evaluate	-	Appraise the situation. Includes taking action based on evaluation.

7.17 Tolerances

Ranges or tolerances are provided if it is important to maintain a parameter within a given band. **WHEN** a range or tolerance (e.g.: 5-15%) is provided, it is understood to mean extra attention should be paid to maintain the parameter within this range.

WHEN a single value is given, it is assumed the value is an ideal value. **WHEN** an ideal value (e.g.: at no load or 350 psig) is provided, it is understood to mean attention should be paid to maintain the parameter at the ideal value but **NOT** be overly concerned if the exact value is **NOT** achieved.

7.18 Multiple Use of EPs and APs.

The Control Room SRO will determine how many procedures can be implemented at a time and their priority based on manpower availability and the particular event in progress. More than one EP shall **NOT** be run concurrently unless directed by the procedure. Generally the use of AP's in conjunction with EP's should be avoided. In some instances it would be proper to use an AP concurrently during a major accident which is being addressed by the EP's. An example of this is upon loss of all Nuclear Service Water in the middle of an accident, the operators would need to utilize the AP for Loss of RN also. **IF** an AP is used during an SI event, USE CAUTION. AP's are generally written assuming an SI has **NOT** occurred (exception - AP/35, ECCS Actuation During Plant Shutdown). Evaluate any AP steps in post SI events to ensure the steps do **NOT** conflict with any EP in effect. **NOT** all AP actions would be appropriate if an SI occurred. (Enclosures in EP/G-1 (Generic Enclosures) may be used when reference by EPs or APs).

Use of most APs that have foldouts will likely be terminated when a reactor trip or SI occurs. There are a few APs with foldouts that could potentially be implemented concurrently with an EP though (Loss of VI or Loss of KC for example). Rules of foldout page use as specified in Section 7.13 should be applied in this situation also. Although unlikely, it is possible that the crew may have one EP foldout in effect at the same time as one of the AP foldouts. Implementation and priority of the AP foldouts will be evaluated as discussed in paragraph above.

ROs may be given procedure responsibilities when AP's and EP's or multiple AP's are in effect at the same time.

7.19 The STA/Operations Shift Manager Interface

The STA monitors the Critical Safety Functions (CSF) and otherwise ensures Core Safety through monitoring of activities and parameters. **IF** any one CSF is other than green, the STA will check whether the CSF non-green status is valid or being caused by an invalid input. **IF** the non-green CSF is invalid, the STA will notify the operating crew of the invalidity. For Red or Orange path procedures, the STA will immediately notify the operating crew that the condition exists and give the associated functional restoration procedure to the crew to implement as the controlling procedure. For Yellow path procedures, the STA will pull the functional restoration procedure and evaluate whether to implement the procedure, with the Operations Shift Managers concurrence, as time allows. This evaluation should consider whether the optimal recovery procedure is properly addressing the current plant conditions in as timely a manner as the functional restoration procedure.

Once status tree monitoring is initiated, the STA should monitor status trees continuously if an orange or red condition is found to exist. **IF** no condition more serious than yellow is found, monitoring frequency may be reduced to 10-20 minutes, unless some significant change in plant status occurs. Status tree monitoring may be performed using OAC SPDS display or EP/1(2)/A/5000/F-0 (Critical Safety Function Status Trees). **IF** the OAC SPDS display is being used, the STA will validate the OAC SPDS status every 10-20 minutes using control board indications.

IF the STA is **NOT** available, the Operations Shift Manager shall assume the STA responsibilities or delegate the STA responsibilities to another licensed operator.

7.20 Control Room Team Responsibilities During the Use of EP/APs

7.20.1 Operations Shift Manager - Responsibilities

7.20.1.1 Assume role of Emergency Coordinator upon activation of the Emergency Plan until properly relieved by the Station Manager.

7.20.1.2 Assess plant conditions to maintain a broad picture of the emergency to ensure:

A. EP/APs are implemented properly and in a timely manner to place the plant in a safe and stable condition.

B. Emergency Classification is done properly and in a timely manner.

C. Monitor performance of entire Control Room team and *intervene and assist as necessary.*

7.20.1.3 Coordinate information exchanges to ensure the Control Room Team, the plant and management are aware of current plant status and progress in the recovery effort.

7.20.1.4 Ensure direction to the board operators is only coming from one person at a time, normally the Control Room SRO.

7.20.1.5 Conduct tailgate sessions at appropriate transitions or pauses discussing plant status, Emergency Plan classifications, etc.

7.20.1.6 **IF** the Operations Shift Manager is initially the Control Room SRO, then the Operations Shift Manager shall turn Control Room SRO duties over to a Shift Supervisor as soon as possible.

7.20.2 Control Room SRO Responsibilities

7.20.2.1 Assess plant condition based on Control Room indication and prior plant status.

7.20.2.2 Assume Control Room command and control function until relieved by Ops Shift Manager (OSM).

7.20.2.3 Assume role of Procedure Reader

A. Select appropriate EP/AP based on recognized procedure symptom(s) and announce to the crew the procedure being entered.

B. Direct all operator actions through the procedure.

C. Delegate use of procedure enclosures to control board operators per procedure guidance or his direction when appropriate with the Operations Shift Manager's concurrence.

D. May check plant parameters personally, as long as the procedure step command and response are verbalized to the crew.

E. Obtain concurrence from the Operations Shift Manager (if available) when transitioning to each procedure.

F. Controls the pace of the crew by reading the procedure at an appropriate pace, **NOT** too fast or too slow.

7.20.3 STA Responsibilities

7.20.3.1 Monitor critical safety functions and advise the Operations Shift Manager if a critical safety function is in jeopardy.

7.20.3.2 Access the severity of any critical safety function challenge and recommend appropriate procedural action to the Operations Shift Manager.

7.20.3.3 Maintain an independent, overall view of the abnormal or emergency situation.

7.20.3.4 Support the Control Room Team in diagnosing and mitigating abnormal plant conditions to ensure a safe overall plant status.

7.20.3.5 Maintain communications with the Operations Shift Manager to ensure awareness of all concerns and precautions involved in restoring the plant to a safe condition.

7.20.3.6 Assist the Operations Shift Manager in classifying an emergency by independently classifying the event in accordance with the Emergency Plan Implementing Procedures.

7.20.3.7 Check Control Room Team implementing appropriate procedures.

7.20.4 RO and BOP Responsibilities

7.20.4.1 Perform immediate actions of appropriate EPs and APs from memory.

7.20.4.2 Provide pertinent information to other team members as necessary.

7.20.4.3 Perform required actions upon command and respond to the procedure reader to check actions taken or results obtained.

7.20.4.4 Provide input to assist the SROs in the decision making process.

7.20.4.5 Utilize the Annunciator Response Procedures as required.

7.20.4.6 Communicate plant status with other team members.

7.20.4.7 Utilize all available valid indication/information to assess plant conditions.

7.20.4.8 Perform all control board manipulations. An SRO may perform control board manipulations in extreme circumstances with the Operations Shift Manager's approval.

7.20.5 Offsite Communicator Responsibilities

Offsite Communicator shall ensure accurate, timely notifications are made to the County, State, and Federal authorities as required by the Response Procedures (RPs).

7.20.6 Command and Control

7.20.6.1 The Control Room Team should avoid a false sense of urgency or security during emergency mitigation. The pace of the crew will be controlled by the Control Room SRO Procedure Reader and monitored by the Operations Shift Manager.

7.20.6.2 Commands to the Control Board Operators should normally come from the Control Room SRO Procedure Reader. The Operations Shift Manager may give commands to a Control Board Operator if the Operations Shift Manager stops the Control Room SRO Procedure Reader, the instructions are needed to safely implement the mitigation strategy and the command or instructions are short and simple. For long complicated instructions, the Operations Shift Manager should stop the whole team and have a tailgate type session to discuss the situation.

7.20.6.3 Communications from the Control Room Board Operators should normally be directed at the Control Room SRO Procedure Reader.

7.20.6.4 Communications from the STA should normally be directed to the Operations Shift Manager who will share them with the Control Room SRO or Control Room Team as appropriate.

7.21 Use of the Balance of Plant Alarm Normal/Silence (N/S) Switch

7.21.1 **WHEN** a Safety Injection has occurred, an RO may place the N/S switch to "Silence" the first time an alarm is acknowledged or subsequently during the event.

7.21.2 The N/S switch will be returned to "Normal" when directed by the station start-up procedures or when directed by the OSM or Emergency Coordinator.

8. Completion Of A Procedure

8.1 Each completed EP and AP should have a Completed Procedure Process Record form attached to the procedure and the person completing the procedure should fill out the required parts and sign the "Verified By" blank.

8.2 An Operations Supervisor will review the completed procedure and will be responsible for approval by signing "Procedure Completion Approved" on the form.

8.3 Completed EPs and APs should be routed to Master File.

9. Deviation From Approved Procedures

9.1 Abnormal/Emergency Operation Not Covered by Procedure:
Actions within bounds of Technical Specifications and Licensing Conditions

This guidance is intended for use only in abnormal operating situations when immediate plant conditions or Technical Specification requirements require the SRO to move quickly. **IF** sufficient time exists, a procedure change is always the preferred route.

WHEN situations occur where established procedures do **NOT** apply, or plant operation **CANNOT** be performed in accordance with approved procedures, use of the procedure shall be suspended and action shall be taken to place the plant in a safe condition or mitigate the event. This should be performed as follows:

- Based on knowledge and resources needed (including the partially applicable AP's or EP's), the SRO shall develop a planned course of action. The SRO is responsible for ensuring these actions are within the bounds of Technical Specifications and licensing condition. **IF** action is outside bounds of Technical Specifications or licensing conditions, see Section 9.2 below.
- The planned course of action should be reviewed and approved by a second SRO, preferably the Operations Shift Manager. The SRO should obtain other reviews as necessary based on the complexity of the situation and the time allowed.
- **IF** time allows, the plan shall be entered in the SRO logbook. **IF** time does **NOT** allow, this may be done after the actions have been taken. However, document components positioned on configuration control card.
- The SRO shall carefully monitor the situation to ensure the course of action is appropriate. As soon as approved procedures can be re-entered, the SRO shall return to the applicable procedures.

9.2 Abnormal/Emergency operation **NOT** covered by procedure: Actions outside License Condition or Technical Specifications

Deviations from Technical Specifications or License Condition during normal plant operation is **NOT** allowed. However, in the event of an emergency, the Operations Shift Manager has the authority and responsibility to take action necessary to protect the health and safety of the public as stated in 10CFR50.54(X) and 10CFR50.72 which reads:

A licensee may take reasonable action that departs form license condition or a technical specification in an emergency 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.

Licensee action permitted by the previous paragraph shall be approved at a minimum, by a licensed Senior Reactor Operator (SRO) prior to taking the action.

The licensee shall notify the NRC Operations Center by ENS telephone of emergency circumstances requiring it to take any protective action that departs from a license condition or technical specification as permitted by the preceding paragraphs. **WHEN** time permits, the notification must be made before the protective action is taken; otherwise, the notification must be made as soon as possible thereafter. The Commission may require written statements from a licensee concerning its action.

9.3 In the event of a national security emergency, the Operations Shift Manager has the authority to take action necessary to implement national security objectives as stated in 10CFR50.54(dd) which reads:

9.3.1 A licensee may take reasonable action that departs from license condition or a technical specification in a national security emergency when this action is immediately needed to implement national security objectives as designated by the national command authority through the Commission, and no action consistent with license conditions and technical specifications that can meet national security objectives is immediately apparent.

A national security emergency is an occurrence, including nuclear attack, a national disaster, or other emergency, which seriously degrades or seriously threatens the national security of the United States or has been declared by the Congress. A national security emergency is established by a law enacted by the Congress or by an order or directive issued by the President pursuant to statutes or the Constitution of the United States.

End of Body

RO QUESTION # 45 ***Bank Question: 1029*** ***Answer: D***

1 Pt. Given the following Unit 1 initial conditions

- 100% power with Tave = Tref
- NC System Boron Concentration 953 ppm
- Control Bank 'D' rods are at 217 steps
- Control Bank 'D' Rod H-8 drops fully into the core
- AP/1A/5500/14 *Rod Control Malfunction* is entered and immediate actions are completed

Thirty minutes after the rod drops

- Load has been reduced to 95% power with Tave = Tref
- NC System Boron Concentration 953 ppm
- Control Bank 'D' rods are at 217 steps
- Rod H-8 has not been retrieved

Which one of the following describes the effect of the event on Rod Insertion Limits and Shutdown Margin?

A. Rod insertion limit is unchanged and shutdown margin is increased.
B. Rod insertion limit is decreased and shutdown margin is unchanged.
C. Rod insertion limit is unchanged and shutdown margin is decreased.
D. Rod insertion limit is decreased and shutdown margin is decreased.

Distracter Analysis:
A. Incorrect: Plausible.
B. Incorrect: Plausible.
C. Incorrect: Plausible.
D. Correct.

Level: RO

KA: 000003.AK1.07 (3.1/3.9)

Lesson Plan Objective:

Source: New

Author: CWS

Level of knowledge: comprehensive

References:

1. McGuire 1 Cycle 16 COLR pages 9,11 & 12
2. REACT (Reactor Engineering Analysis & Computer Tools) Shutdown Margin-Unit at Power, Modes 1 & 2 Calculations

Comment:

Answer "D" is based on the Reactor Engineering Analysis & Computer Tools (REACT) software application, which reflects a reactivity penalty for the dropped rod and a reduction (decrease) in shutdown margin (SDM). This is a conservative adjustment to the calculation of shutdown margin that is specific to McGuire Nuclear Station. The question did not specifically state that the applicants were to determine the effect of a dropped rod on SDM as defined by the REACT software.

Proposed Resolution

The facility recommends that the answer to this question be changed from "D" to "B."

Justification

According to the classical definition, shutdown margin is unaffected by a dropped control rod. Interviews with all the RO applicants confirmed that they interpreted the question to be based upon the classical definition of SDM.

Training material on shutdown margin contained in the Generic Fundamentals training program lesson plan is supplied as part of this comment package:

- MNS Lesson Plan BNT- RT08

REACTOR OPERATIONAL PHYSICS

REVISION HISTORY

REVISION NUMBER	DATE	COMMENTS
0	11/26/01	Converted to BNT format and objectives combined with other lesson plans as appropriate.
0	08/23/02	Corrected typos on page 34
1	10/23/02	Inserted color graphics, highlighted definitions in red, and re-sequenced objectives. Combined duplicate objectives and reworded objectives for clarity.

APPROVAL PROCESS RECORD

PROGRAM: Basic Nuclear Training

TOPIC: Reactor Operational Physics

REV. 1 **DATE** 10/23/02

ORIGINATOR Wiley Killette **DATE** 10/23/02

REVIEWED BY (1) Richard Michael **DATE** 11/05/02

APPROVED BY Ed Roberts **DATE** 11/06/02

NOTE:

(1) If no review is performed, place N/A in the signature section.

SEQ	REACTOR OPERATIONAL PHYSICS ENABLING OBJECTIVES	BOT	GFES	ENG
1	Define shutdown margin (SDM). RT08029	X	X	X
2	Calculate shutdown margin and changes in shutdown margin for given plant parameters. RT08030, RT08031	X	X	X
3	Relate the concept of subcritical multiplication to predict count rate response for control rod withdrawal during the approach to criticality. RT08004	X	X	
4	Calculate ECP using a 1/M plot. RT08006	X	X	X
5	List the reactivity control mechanisms that exist during approach to criticality. RT08002	X	X	
6	Describe the purpose of performing the following: A. Estimated critical Boron Concentration (ECB) B. Estimated Critical Rod Position (ECP) RT08032			X
7	Using materials provided, calculate an ECP. RT08007	X	X	
8	Describe count rate and instrument response that should be observed for rod withdrawal during the approach to criticality. RT08003	X	X	
9	List the parameters that should be monitored and controlled during the approach to criticality. RT08001	X	X	
10	Explain the parameters to be observed when the reactor is very close to criticality. RT08005	X	X	
11	List the parameters that should be monitored and controlled upon reaching criticality. RT08008	X	X	
12	Define criticality as related to a reactor startup. RT08009	X	X	

SEQ	REACTOR OPERATIONAL PHYSICS ENABLING OBJECTIVES	BOT	GFES	ENG
13	Describe reactor power once criticality is obtained. RT08010	X	X	
14	Describe how to determine if a reactor is critical. RT08011	X	X	X
15	List the parameters that should be monitored and controlled during the intermediate phase of a reactor startup (from criticality to POAH). RT08012	X	X	
16	Describe reactor power response prior to reaching the POAH. RT08014	X	X	
17	Discuss the concept of the point of adding heat (POAH) and its relationship to reactor power. RT08013	X	X	
18	Explain the characteristics to look for when the POAH is reached. RT08015	X	X	X
19	Describe reactor power response after reaching the POAH. RT08016, RT08017	X	X	X
20	Explain the effects of power mismatch (PMM) on core reactivity. RT08020			X
21	Describe the means by which reactor power will be increased to rated power. RT08021	X	X	
22	Describe the monitoring and control of reactor power and primary temperature from 0% to 15% (B&W). RT08018	X	X	
23	Describe the monitoring and control of Tave, Tref, and power during power operation. RT08019	X	X	
24	Explain the effects of control rod motion, boration, and dilution on reactor power. RT08022	X	X	
25	Explain the relationship between steam flow and reactor power given specific conditions. RT08023	X	X	
26	Explain reactor response to a control rod insertion. RT08026	X	X	

SEQ	REACTOR OPERATIONAL PHYSICS ENABLING OBJECTIVES	BOT	GFES	ENG
27	Explain how boron concentration affects core life. RT08024	X	X	
28	Explain the shape of the curve of reactor power versus time after a reactor trip. RT08025	X	X	
29	Define decay heat. RT08027	X	X	X
30	Explain the relationship between decay heat generation and: A. Power Level History B. Power Production C. Time since shutdown/reactor trip RT08028	X	X	X
31	Describe the significance and causes of events in SOER 88-2. RT08033			X

INTRODUCTION

In previous chapters, we have studied the various reactivity effects and methods of reactor control associated with a pressurized water reactor (PWR). In this chapter we will study these effects as they interrelate during actual operation of a PWR. The plant operations to be covered are:

- Heatup of the Reactor Coolant System to normal operating temperature.
- Reactor startup and power increase to full power.
- Plant external load changes at power (load follow operations).
- Reactor shutdown and RCS cooldown.
- Reactor response following a reactor trip (power vs. time).

It is not the intent of this chapter to cover all actions required to complete the evolutions described above. This chapter contains a general discussion of the major steps which involve reactivity changes.

RCS HEATUP TO NORMAL OPERATING TEMPERATURE

One of the first steps in preparing for the unit for startup is the heatup of the RCS to normal operating temperature (~557°F). Prior to any boron dilution, it is necessary to fully withdraw all the shutdown rod banks from the reactor. This provides an adequate negative reactivity reserve which can be rapidly inserted into the reactor if a problem occurs. Pulling the shutdown rods is a positive reactivity addition. The reactor operator must assume that the reactor may achieve criticality during this evolution. The reactor must be maintained sufficiently subcritical at all times unless a startup is in progress. Two different terms measure this: the Shutdown Value and the Shutdown Margin.

SHUTDOWN VALUE

The shutdown value (SDV) is that amount of reactivity that the core is, or can be made subcritical by, from its present condition. The shutdown value may be calculated by using the following equation.

$$SDV = \frac{1 - K_{eff}}{K_{eff}}$$

Equation 1

Note that this equation is different from the reactivity equation in that the terms in the numerator are reversed. Any parameter that varies core reactivity will cause the shutdown value to change (e.g. control rod density changes, moderator density changes, poison concentration changes, etc.) If the core reactivity becomes less negative the shutdown value will decrease.

SHUTDOWN MARGIN

Objective 1

Shutdown margin (SDM) is the instantaneous amount of reactivity that the core is, **OR** can be made, subcritical from its present condition:

- *with all control rods fully inserted* with the exception of the most reactive control rod which is fully withdrawn from the core
- at any time during the core cycle

SDM by definition applies to two different operating conditions.

"Instantaneous amount of reactivity that the core is" applies to a shutdown reactor (if the reactor is currently shut down with all control rods inserted, then the highest worth rod were to eject, the amount of reactivity that the reactor is shut down by would be the SDM).

"Instantaneous amount of reactivity that the core can be made" applies to conditions other than fully shut down. "Can be made" implies that, by tripping the reactor, all control rods (excluding the highest worth rod) fully insert. The instantaneous term in this portion of the definition is to include the time it takes for the control rods to reach the bottom of the core and the time for the power defect (including moderator temperature) to reach its reference value if operating above the POAH at the time of the trip.

Technical Specifications require a minimum shutdown margin to ensure an unanticipated criticality event does not occur. A typical value required is a shutdown margin of –1.3% ΔK/K. These values change depending on the operational mode. Core design and existing conditions will determine the amount of reactivity by which a reactor is actually shutdown.

Objective 2

Some common conditions (parameters) will effect the SDM whether the plant is operating at power or shut down.

- Xenon concentration – An increase would add negative reactivity, increasing the shutdown margin.
- Samarium concentration – An increase would add negative reactivity, increasing the *shutdown margin.*
- Boron concentration – An increase would add negative reactivity, increasing the shutdown margin.
- Stuck Control Rod – A stuck control rod (not the highest worth rod) will reduce the amount of reactivity that the reactor can be shutdown by, therefore reducing the shutdown margin.

Conditions that have an effect on SDM while shutdown are the common conditions listed above as well as:

- Moderator Temperature – An increase would insert negative reactivity, increasing the shutdown margin.
- Number of fuel assemblies in the core – An increase would add positive reactivity, decreasing the shutdown margin.
- Fuel temperature – An increase (caused by a decrease in heat removal rate or an increase in moderator temperature) would insert negative reactivity, increasing the shutdown margin.

Conditions that have an effect on SDM while operating at power are the common conditions listed above as well as:

- Exposure/burnup of fuel assemblies in the core – An increase would add negative reactivity, increasing the shutdown margin

Changing moderator temperature using control rods while in the power range will not change the SDM. If control rods were withdrawn adding +10 pcm, the net effect (assuming an undermoderated core) would be an increase in moderator temperature adding –10 pcm. Upon a reactor trip, the control rods would add –10 pcm more negative reactivity while the moderator returning to its reference temperature would add +10 pcm more positive reactivity so the two conditions effectively cancel each other out. The shutdown margin is calculated using the same equation as used for shutdown value.

Objective 2

A reactor is shutdown with a SDM of –1.8 %ΔK/K. What is the new SDM if boron concentration is reduced by 25 ppm? Assume differential boron worth is –10 pcm/ppm.

Change in boron concentration = (-25 ppm)

Change in reactivity = (-25 ppm) x (-10 pcm/ppm) = +250 pcm

+250 pcm = .25 %ΔK/K

New SDM = $SDM_{initial}$ + .25 %ΔK/K = -1.8 %ΔK/K + .25 %ΔK/K

-1.55 %ΔK/K

Example 1

SUBCRITICAL MULTIPLICATION

Objective 3

A useful parameter for observing the effects of subcritical multiplication is the neutron count rate (CR) measured by the nuclear instruments. Recall from previous chapters that the neutron count rate can be expressed as:

$$CR = S_o \frac{1}{1 - K_{eff}} \eta$$

Equation 2

Where: CR = the neutron count rate, S_o = the source strength, η = the detector efficiency

The indicated count rate in the subcritical range, by itself, is often quite useless. However, count rate comparisons are very useful, especially during reactor startups. The count rate ratio is the comparison of two count rates and can be expressed as:

$$\frac{CR_2}{CR_1} = \frac{1 - K_{eff1}}{1 - K_{eff2}}$$

Equation 3

Where: 1 represents a reference time, and 2 represents some time later

There are some other variations of the count rate comparison shown above. When using the values for reactivity instead of K_{eff} the above equation becomes:

$$\frac{CR_2}{CR_1} = \frac{\rho_1 (1 - \rho_2)}{\rho_2 (1 - \rho_1)}$$

Equation 4

The reactor is subcritical with a stable count rate of 200 counts per second (cps) on the Source Range instruments. K_{eff} at this time is 0.99. Control rods are then withdrawn from the core. After several minutes the operator observes a stable count rate of 400 cps on the Source Range instrument. What is the new K_{eff}? CR_1 = 200 cps, CR_2 = 400 cps and K_{eff-1} = 0.99

$$\frac{CR_2}{CR_1} = \frac{1 - K_{eff1}}{1 - K_{eff2}} \qquad K_{eff-2} = 1 - \left(\frac{(1 - K_{eff1})CR_1}{CR_2}\right)$$

$$K_{eff-2} = 1 - \left(\frac{(1 - 0.99)\,200}{400}\right)$$

$$K_{eff-2} = 0.995$$

Example 2

If the operator determines that the source range count rate has doubled, he notes the position of the shutdown rods. Using an integral rod worth versus steps withdrawn curve (Figure 1 shows a typical curve), the operator can determine how much reactivity remains in those shutdown rods not yet fully withdrawn. If the amount of reactivity remaining is greater than that which he has already added, he should not proceed. Criticality will occur if he completes withdrawal of the shutdown rods.

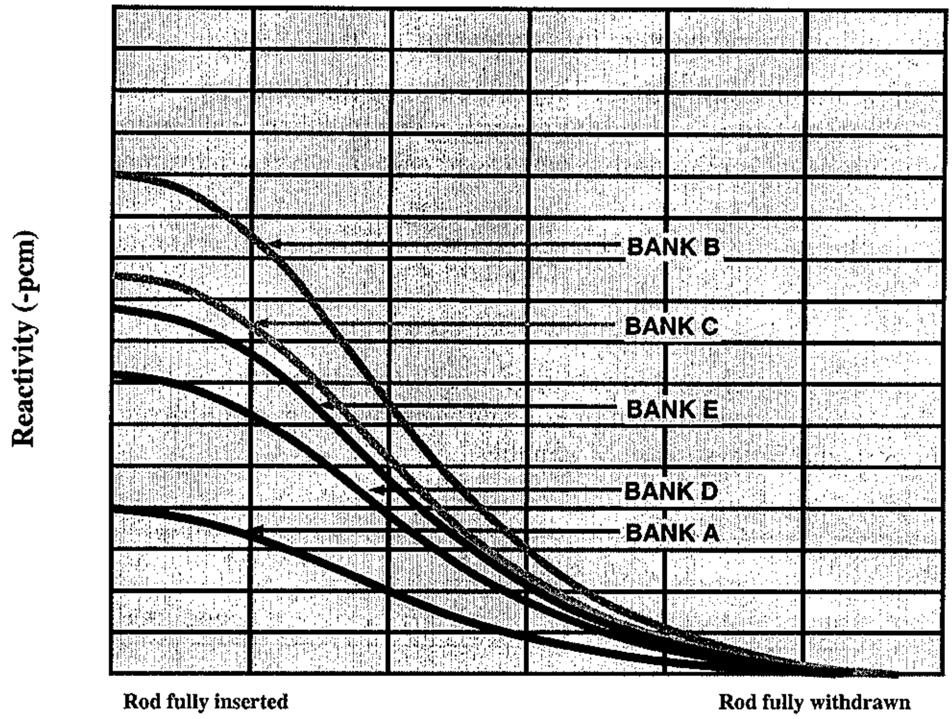


Figure 1 Integral Rod Worth vs. Steps Withdrawn

The reactor coolant pumps, pressurizer heaters, and reactor decay heat are the sources of heat during heatup to normal operating temperature. Heatup procedures typically limit the heatup rate of the RCS to 50°F per hour.

Under certain conditions the possibility exists that the moderator temperature coefficient is positive. This may cause the plant to lose some of its required shutdown margin due to adding positive reactivity during the heatup. Typical plant technical specifications require the shutdown margin be maintained greater than one percent ΔK/K when RCS temperature is less than 200°F and greater than 1.3 percent ΔK/K when RCS temperature is above 200°F. It is the operator's responsibility to ensure that the minimum shutdown margin requirements are met at all times.

Once the RCS has been increased to startup temperature and normal operating pressure, the reactor startup may begin.

1/M PLOTS

Objective 4

The count rate comparison process discussed above yields an ever-increasing value as criticality is approached. To accurately plot an expected point of criticality would be difficult, because the count rate ratio could increase from approximately 1 to several million prior to criticality. To plot the effects of subcritical multiplication, the inverse of the multiplication is used. In other words, the ratio of CR_1 to CR_2, or the inverse count rate ratio, is plotted following each reactivity addition (Figure 2). As K_{eff} approaches one, the inverse count rate ratio approaches zero. Plotting a function that approaches zero is a lot easier than plotting a function that approaches infinity. A conservative estimate of the critical rod position can be obtained by extrapolating the

slope between the two latest points to predict criticality. This approach may also be used during fuel loading to monitor for and prevent inadvertent criticality.

M, the subcritical multiplication **"factor"** is defined to be the factor by which the neutron population has increased from some initial condition due to subcritical multiplication. Recalling the subcritical multiplication relationship developed earlier, we can determine M as:

$$M = \frac{CR_{final}}{CR_{initial}} = \frac{1 - k_{initial}}{1 - k_{final}}$$

Equation 6

Since we are measuring the change in subcritical multiplication from an initial state, the value of CR_0 remains the same for each change in K_{eff} that we make. Let's start from an initial state and make some changes and observe how M changes:

Control Bank Position	Count Rate	M
0% Initial Condition	100 cps	Undefined
12% withdrawn	200 cps	2
20% withdrawn	300 cps	3
28% withdrawn	580 cps	5.8

As subcritical multiplication increases, the value of M increases as well. Mathematically, the limit of M as K_{eff} approaches 1 is infinity. This is not much use in predicting criticality.

The value of 1/M, however, as discussed above, approaches 0 as the rods are withdrawn and K_{eff} approaches 1.

Control Bank Position	Count Rate	M	1/M
0% Initial Condition	100 cps	Undefined	1
12% withdrawn	200 cps	2	0.5
20% withdrawn	300 cps	3	0.33
28% withdrawn	580 cps	5.8	0.17

Since 1/M is the inverse of M (subcritical multiplication factor), we can derive an expression for 1/M as:

$$1/M = \frac{CR_{initial}}{CR_{final}} = \frac{1 - k_{final}}{1 - k_{initial}}$$

Equation 7

Initially, $K_{eff} = 0.8$ and the stable count rate is 50 cps. Complete this table:

K_{eff}	Count Rate	M	1/M
.8	50	---	1.0
.9			
.95			
.975			
.9875			

Example 2

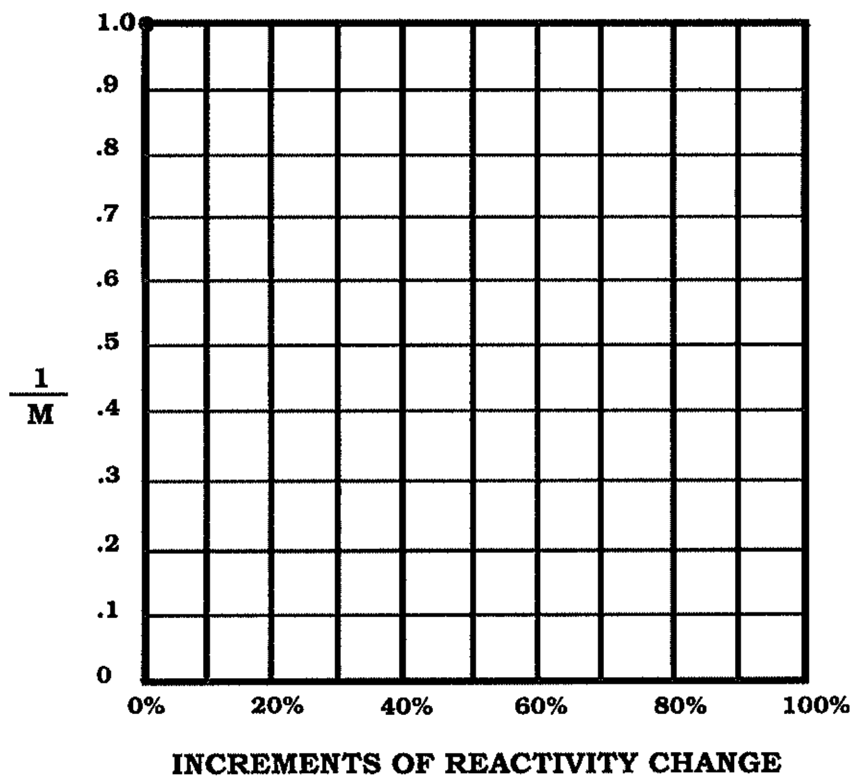


Figure 2 1/M Graph

During refueling and reactor startups, a plot of 1/M versus the changing reactivity parameter is kept to predict the value of the parameter at criticality. For fuel loading, this would be the number of fuel assemblies loaded, for a startup, the position of the control rods would be plotted.

A 1/M plot is performed in the following manner. Conditions are "initialized" and the 1/M plot labeled with these initial conditions. For example, let's assume an initial count rate of 20 cps with the control banks fully inserted (0% withdrawn). Our initial 1/M plot would then look like Figure 2. We then make a change from this initial condition, allow the neutron population to re-establish its subcritical equilibrium and then take data. Assume in our case that we withdraw the control rods to the 10% withdrawn position and the indicated count rate stabilizes at 27 cps. We would first of all determine the value of 1/M at the new condition. Since $1/M = CR_0/CR_1$, the

value of 1/M at 10% withdrawn is 0.74 (20 cps/27 cps). We then plot that data point and connect it with a straight line to the immediately preceding point. See Figure 3 below.

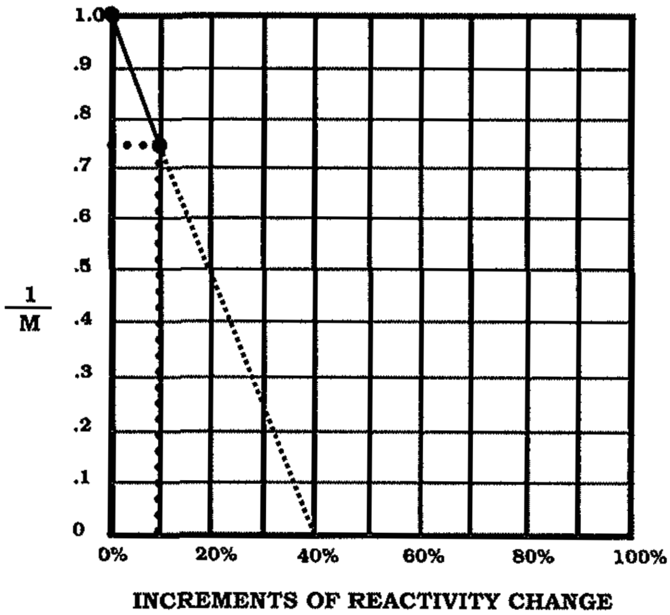


Figure 3 1/M Graph for control rods 10% withdrawn

We continue along the straight line that we just drew, differentiating between actual (straight line) and prediction (dashed line), until we cross the x-axis. We are predicting the value of the rod position when the reactor becomes critical, the ECP (1/M goes to zero). This process is then repeated for each subsequent reactivity addition.

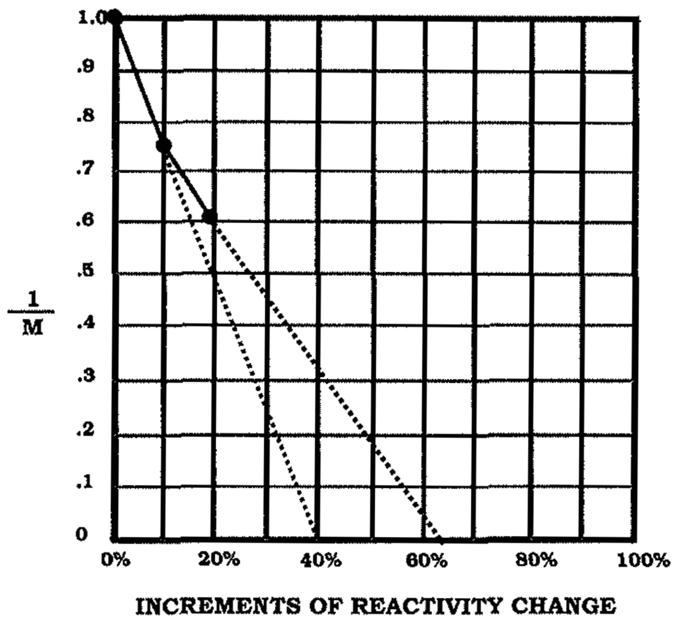


Figure 4 1/M Graph for control rods 20% withdrawn

Assume that we withdrew the rods to 20%. Count rate stabilizes at 33 cps.

First calculate the new 1/M value:

20 cps / 33 cps = 0.61

Next, plot the results. Start at the previous 1/M value (0.74) and draw a straight line between the two points. The new calculated critical rod position is 62% withdrawn. See Figure 4 above.

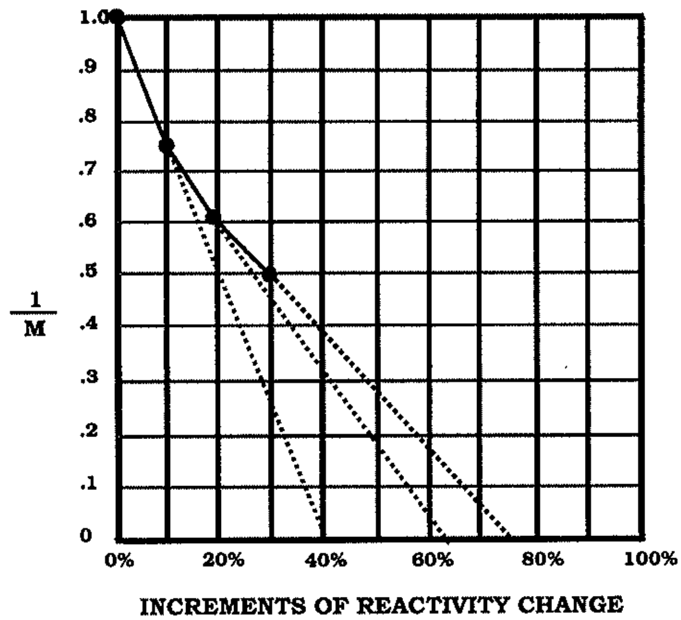


Figure 5 1/M Graph for 30% withdrawn

We withdraw control rods to 30%. After counts have stabilize, the new count rate is 40 cps. Calculating for 1/M (20 cps / 40 cps) results in 0.5. Plotting the new point and drawing a straight line between the previous and new point leaves us with a predicted criticality of 75% withdrawn. See Figure 5 above.

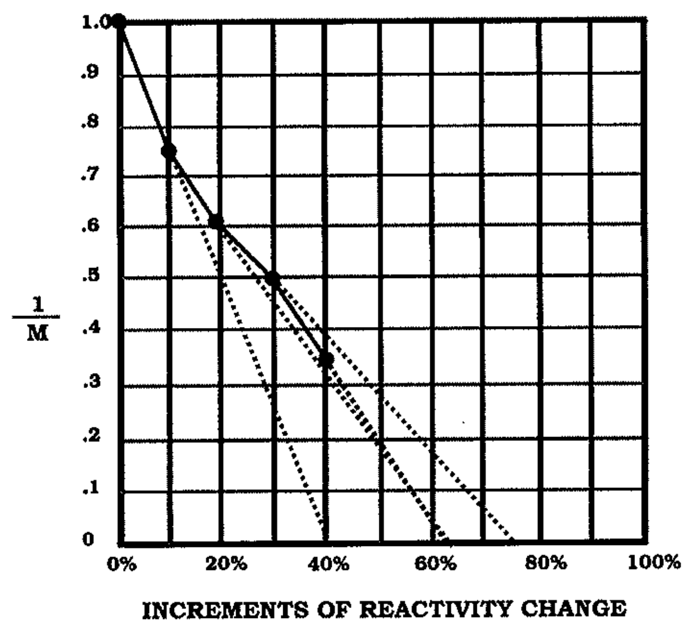


Figure 6 1/M Graph for control rods 40% withdrawn

Withdrawing control rods to 40% results in an equilibrium count rate of 59 cps.

20 cps / 59 cps results in a 1/M of 0.34. After plotting, the new predicted criticality is 62% withdrawn. See Figure 6 above.

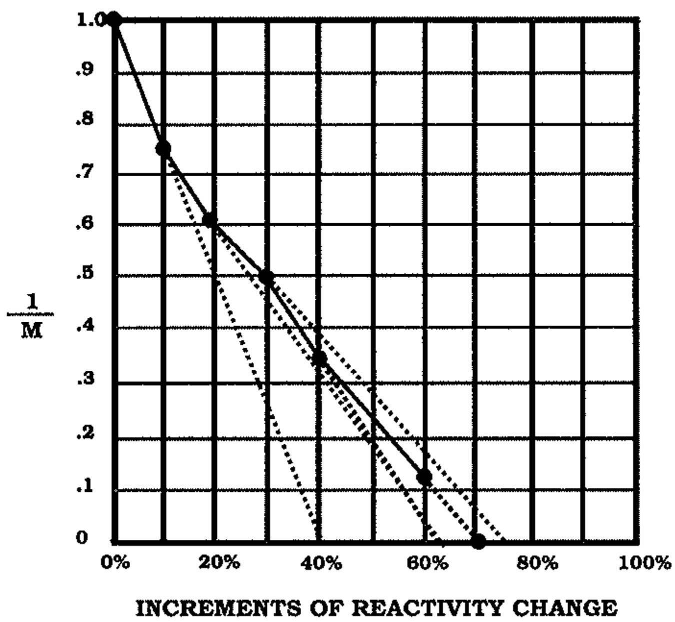


Figure 7 1/M Graph for control rods 60% withdrawn

Withdrawing the control rods to 60% results in a count rate of 160 cps, with the 1/M value of 0.125. The predicted criticality is now 70% withdrawn. See Figure 7 above.

The following is data from the 1/M sequence:

Control rod position	Count rate final	1/M value
10%	27	0.74
20%	33	0.61
30%	40	0.5

Control rod position	Count rate final	1/M value
40%	59	0.34
50%	80	0.25
60%	160	0.125

Remember the thumb rules: ***"Doubling the count rate 5 to 7 times we would be critical and If we double the count rate, we have reduced the margin to criticality by one half."***

Observe the gray sections of the 1/M sequence. At 30% withdrawn, the count rate doubled from the initial count rate of 20 cps and the 1/M value went from 1 to 0.5, which reduced the margin to criticality by one half. At 50% withdrawn, the count rate doubled a second time from the original

(2^2 x 20 = 80). The margin to criticality from the first doubling decreased by one half (from 0.5 to 0.25). At 60% withdrawn, the original 20 cps was doubled a third time (2^3 x 20 = 160). The 1/M value from the last doubling (50%) decreased by one half. So far we doubled the count rate three times. Two more doublings and we should be critical or supercritical. The operators, however, should realize that the reactor may go critical at any time during a reactor startup.

Ideally, the 1/M plot would be a straight line for each reactivity change that occurs – which is to say that the reactor response would be identical for identical reactivity parameter changes. Often this is not the case. The value of reactivity inserted by parameter changes varies with core conditions (rod worth, temperature coefficients, etc.) so the reactor's response can vary from what is predicted.

The variation can be either less subcritical multiplication than expected (conservative), or more than expected (non-conservative). The shape of the resulting 1/M plot reflects this trend.

Conservative plot: This is conservative because the actual critical position is later than predicted. The predicted critical condition moves further to the right.

Non-conservative: This is classified as a non-conservative 1/M because the actual critical position is earlier than predicted. The predicted critical value moves closer to the left with each subsequent reactivity change.

See Figure 8 below.

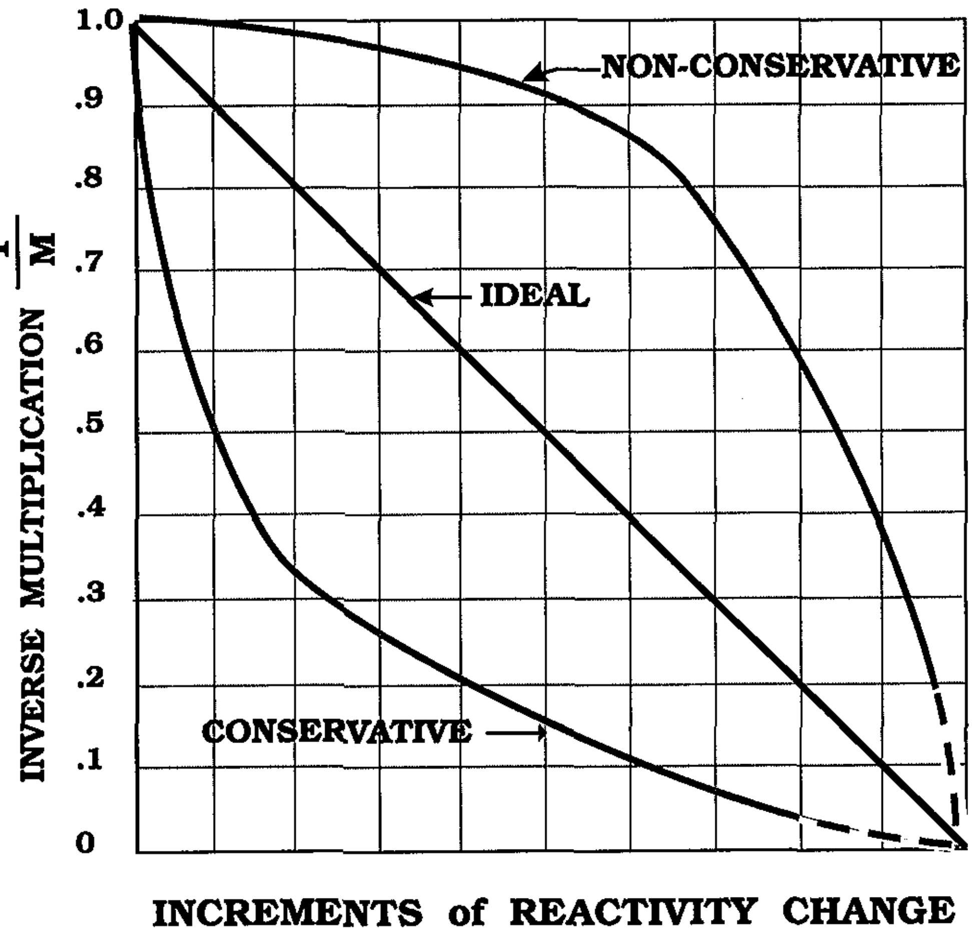


Figure 8 Conservative verse Non-Conservative 1/M Plots

REACTOR STARTUP AND APPROACH TO CRITICALITY

Objective 5

Before beginning reactor startup activities, the reactor operators (and/or the reactor engineering staff) should calculate the reactivity changes needed for each stage of the reactor power escalation from hot shutdown to full power. These calculations provide the following:

- Amount of dilution water needed
- Amount of boric acid needed
- Control bank positions for each plateau (ensures insertion limits are observed and sufficient control reactivity is reserved for future needs).
- Reactivity changes (magnitudes and rates) needed to compensate for fission product poisoning (the dilution/boration amounts are included above).
- Assurance that auxiliary systems can adequately support the startup.
- Forward-looking preparations such as these help ensure that the power escalation occurs without unplanned interruptions.

With the reactor shutdown (K_{eff} <1.0), the following factors will affect the reactivity in the core; control rod worth, RCS boron concentration, core age, fission product poisons, moderator temperature and fuel temperature. With the reactor shutdown, control rod worth is the dominant factor controlling K_{eff}. The shutdown margin is affected by the same factors as K_{eff}.

Objectives 6A, 6B, 7

Prior to commencing the actual startup of the reactor, the operator is required to estimate the point at which the reactor will go critical. This is accomplished by completing an Estimated Critical Boron (ECB) calculation and an Estimated Critical Position (ECP) determination. Both the ECB and ECP are mathematical calculations in which moderator temperature, power history, time since shutdown, core age, boron concentration, and control rod worth are used to predict when the reactor should go critical. To complete the ECB, the operator selects the control rod height at which he wants the reactor to go critical, and then, by means of a reactivity balance, he determines the change in RCS boron concentration required to achieve criticality at this desired rod position. Using the subsequent boron concentration adjustment (normally a dilution), an ECP calculation is performed. The boron concentration used in the ECP calculation is the sample boron concentration after boron concentration is adjusted.

The purpose of the ECP is to estimate the position of the control rods when criticality is obtained by pulling rods. This provides the operator with a benchmark by which he can judge the reactors response. Any deviation of the actual core parameters from those used in the ECP calculation will cause the actual critical rod position to be different from the ECP. Consider the case where the ECP is calculated for 10 hours following a trip from a long run at full power. Subsequent events cause the startup to be delayed until 16 hours after the trip. While the ECP uses Xenon-135 negative reactivity value at or near the post-trip peak, the actual critical position must balance less negative reactivity from Xenon than predicted. The actual critical position would be lower in the core (less withdrawn) than the estimate.

In general, the following guidance should be considered in evaluating what will cause the ECP and actual critical position (ACP) to differ:

- Actual condition has less negative reactivity than the predicted condition – ACP less withdrawn than ECP
- Actual condition has more negative reactivity than predicted condition – ACP more withdrawn than ECP

(Actual plant procedures may be used at this time to calculate an ECP)

Once the boron concentration as been adjusted to the desired value and the ECP has been completed, the reactor startup may begin. As a check on the ECP the operator is again required to use counts doubling. By recording the source range count level existing prior to control rod withdrawal and noting the rod position at which count level doubles, the operator can approximate actual critical rod height. Curves similar to Figure 9 may be used to assist the operator in making this determination.

The Inverse Count Rate Ratio (1/M) plot is used as described earlier. During a startup, a baseline inverse count ratio (ICR) is typically established after fully withdrawing the first control bank and allowing the count rate to stabilize. A point is then plotted on the graph at the intersection of the rod position (Shutdown Banks fully withdrawn) on the x-axis and the 1.0 line on the y-axis.

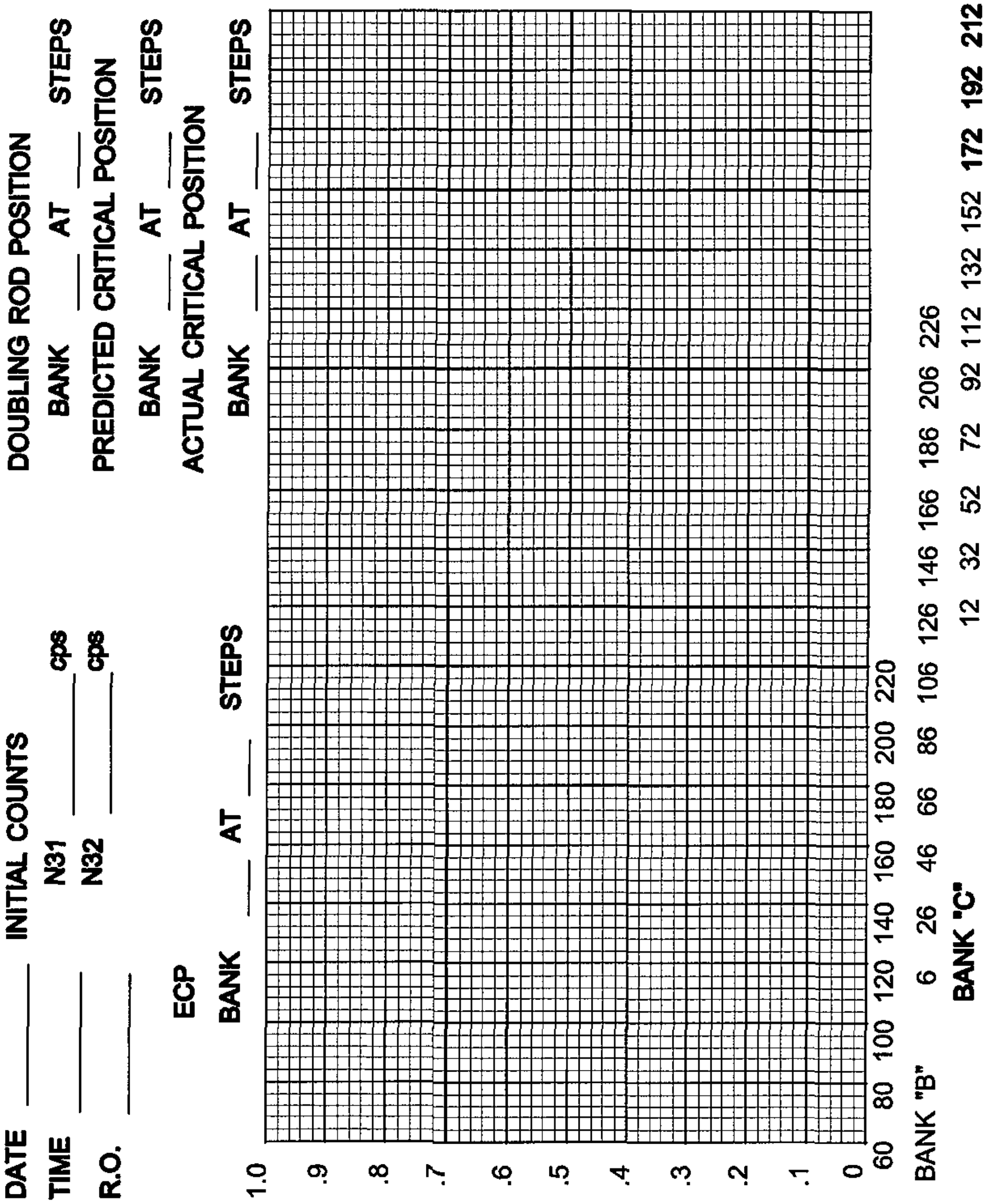


Figure 9 Inverse Count Ratio

If the estimated critical rod height determined from the ICR plot is outside the bounds of the ECP the startup must be terminated and all rods fully inserted or the rods are inserted to a lower position that ensures the reactor will remain subcritical. The reason for the difference must be identified and corrected before resuming startup activities. The most common errors are in the reactivity balance calculation of the ECP. If the reason for the discrepancy cannot be found, the reactor engineering staff must be notified so that an investigation can be initiated.

Objective 8

When rod motion is stopped in a subcritical reactor, the Source Range count rate will achieve a new equilibrium level and the SUR meter will eventually indicate zero, indicating that the reactor power level is no longer changing. As K_{eff} approaches 1.0, the time required for the Source Range count rate to reach the new equilibrium level following rod withdrawal will become longer and longer. When the reactor is close to criticality, it is possible for this time period to be several minutes. The Source Range instruments will be responding to the increased neutron flux levels as the control rods are withdrawn. With the insertion of positive reactivity there will be indication of a positive startup rate (SUR) and the count rate will increase correspondingly.

Objective 9, 10

As a minimum, the reactor operator will monitor the following parameters on the approach to criticality: reactor power level, reactor SUR, control rod position, boron concentration, and moderator temperature. The SUR meter is usually the first indication to respond to any reactivity change in the reactor.

Objectives 11, 12, 13, 14

An exactly critical reactor has a stable neutron count rate after rod motion has ceased. This is the same indication that is observed when the equilibrium count rate is reached due to subcritical multiplication. It is; therefore, difficult to determine when the reactor is exactly critical. It is; however, readily apparent when the reactor is supercritical. Therefore, the reactor is said to be CRITICAL when the reactor is slightly supercritical as indicated by a constant positive startup rate and a steadily increasing count rate with no rod motion or any other reactivity addition being made to the core. Normally, the source range count rate will have doubled five to seven times before criticality is achieved. Once criticality is achieved, the operators must continue to monitor reactor power level, reactor SUR, control rod position, boron concentration, and moderator temperature to ensure safe operation of the reactor.

INTERMEDIATE RANGE OPERATIONS

Objective 15

After criticality has been achieved, control rods are used to increase reactor power to 1×10^{-8} amps in the intermediate range. As power is increased, the intermediate range instruments will start to indicate increasing power and a positive SUR. The reactor operator will check the operation of the nuclear instrumentation by verifying correct overlap between the ranges (i.e. power increasing at the same rate on both) before blocking the source range. The power increase is stopped and power is stabilized to record actual critical conditions. Control rod positions, boron concentration, and RCS average loop temperatures are recorded. This data is typically reviewed by the station nuclear engineer for accuracy and is used as necessary to update the reactivity curves.

1×10^{-8} amps was selected as the standard point for taking the critical data for two reasons:

- It is well above the source range. Source neutron effects are negligible.
- It is below the point of adding heat to the RCS. The Doppler (fuel temperature) effect is not present and moderator temperature is constant so there is no moderator temperature coefficient effect.

Objective 16

After the critical data has been recorded, the control rods are withdrawn to increase power to the point of adding heat (approximately 3×10^{-6} amps). A key item to understand is that, in the intermediate range, any addition of positive or negative reactivity will produce a SUR that will remain constant (reactor power will increase or decrease at a constant logarithmic rate) until one of the following occurs:

- Reactor power enters the source range and an equilibrium count rate is established due to subcritical multiplication.
- An amount of reactivity equal to but opposite the initial reactivity addition is made.
- Reactor power reaches the point of adding heat and doppler (fuel temperature) and moderator temperature provide inherent stability effects as discussed below.

Objective 17

Fuel temperature is dependent on decay heat, heat produced directly from fission, ambient losses, and system losses. Up to about 0.1% power, decay heat is the major heat producer. *The point where heat production from fission becomes significant enough to overcome ambient losses is defined as the point of adding heat (POAH).*

Objectives 18, 19, & 20

As nuclear power increases above the point of adding heat, fuel and moderator temperatures increase. This results in a negative reactivity addition due to the Doppler and moderator temperature coefficients, which eventually causes the startup rate to fall to zero and power to stabilize. If there were no steam demand by the secondary, the reactor power would stabilize at the point of adding heat and the positive reactivity added by the control rods would be balanced by the reactivity due to changes in moderator temperature and fuel temperature. In most applications, power plants use steam dump valves (turbine bypass valves) to control the average moderator temperature during startup. These valves respond to changes in steam pressure to regulate the primary temperature.

As power increases above the point of adding heat, the power mismatch (reactor power > steam demand) causes the moderator temperature to begin to increase. This increases steam pressure and the dump valves/bypass valves open to pass more steam flow and reduce steam pressure back to setpoint. This increases secondary demand to match the increased reactor power. The negative reactivity associated with the power defect balances the positive reactivity added by the control rods. Power will stabilize above the point of adding heat, at approximately two percent of full power. As the moderator temperature increases, the coolant expands into the pressurizer causing increased level and pressure. Depending on the sensitivity of secondary instrumentation increased steam flow or pressure may also be observed.

POWER RANGE OPERATION

Reactor power is maintained at two percent while the secondary plant is prepared for power operations. When the turbine generator is ready for startup, the reactor operator withdraws

control rods as necessary to establish approximately 7-15 percent power. Steam is bled directly to the condenser by the steam dump system to maintain steam pressure.

Objective 21

As the main turbine draws more steam, the steam dump system will automatically reduce the amount of steam being bled to the condenser. RCS temperature and reactor power are maintained at approximately constant levels. When the turbine generator load has increased to match reactor power, the steam dumps will be fully closed. Power escalation continues by gradually opening the main turbine governor valves.

When power is increased above 15 percent, the control rods may be placed in automatic control. The control rods will step out automatically to raise T_{avg} in accordance with the ramped T_{avg} versus power program (Figure 10). Values shown are for a typical 4-loop Westinghouse plant. Actual values will vary from plant to plant.

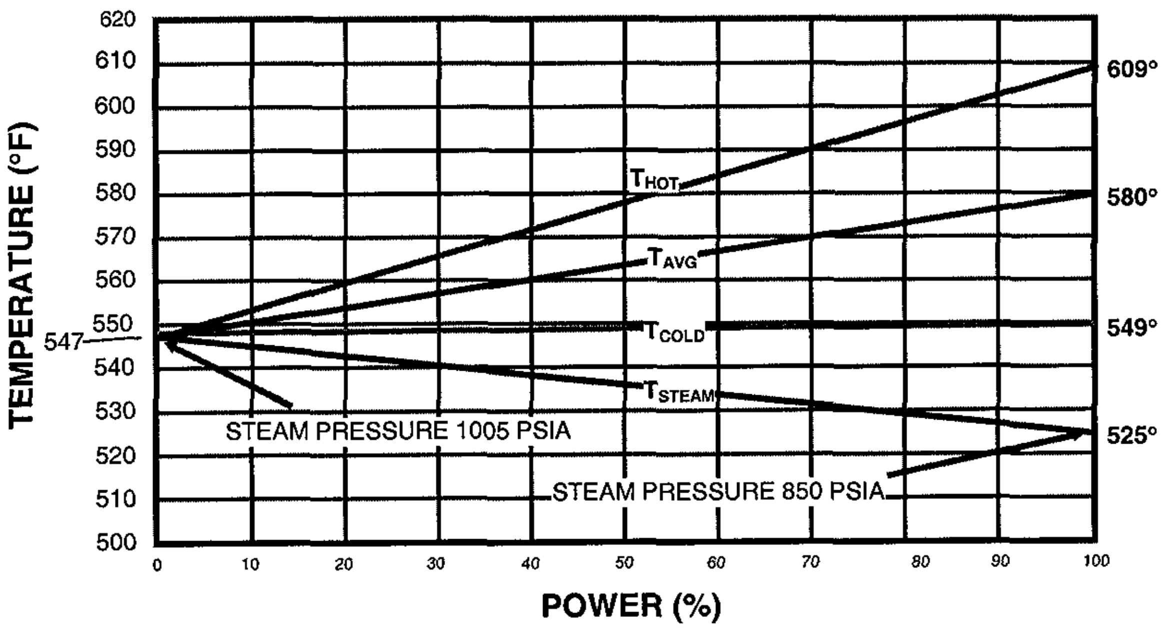


Figure 10 Reactor Coolant Temperature verses Reactor Power (Westinghouse)

Objective 22

For B&W plant designs, T_{ave} is ramped with control rods in manual from the POAH to 15% power using steam dumps while maintaining constant steam generator level. After 15% power, constant T_{ave} is maintained constant up to rated power and steam generator levels are adjusted as necessary.

If boron dilution is not commenced, the control rods will step out until the control bank is fully withdrawn. Further automatic rod withdrawal is then inhibited. Continued power increases will cause the RCS temperature to decrease, adding positive reactivity to compensate for the power defect.

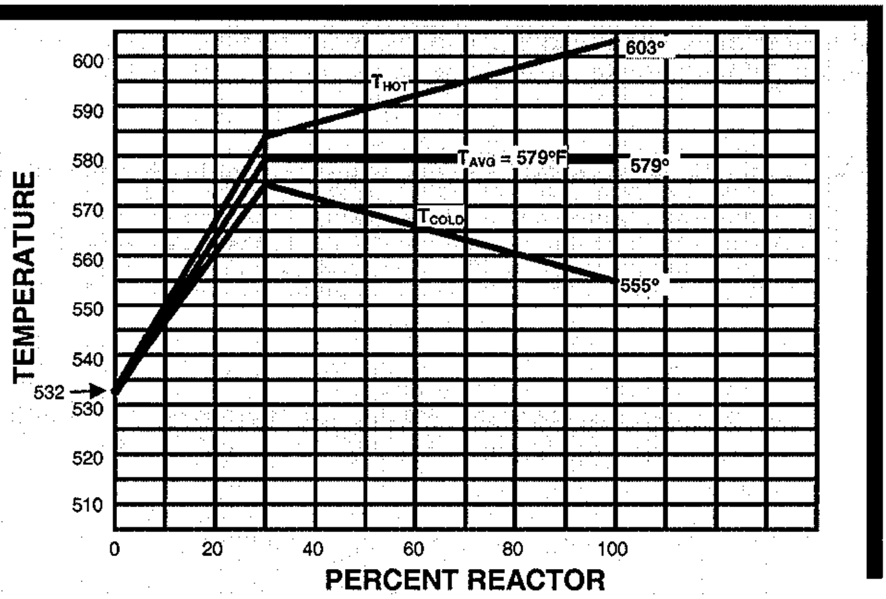


Figure 11 Reactor Coolant Temperature verses Reactor Power (B&W)

Decreasing RCS temperatures will cause lower steam pressures in the secondary plant and lower overall plant efficiency. To prevent the decrease in RCS temperature, a dilution must be commenced to follow the power increase rate.

Once the reactor has operated at power, xenon and other fission product poisons begin to build to their equilibrium values. This buildup adds negative reactivity to the reactor. Boron concentration changes or control rod withdrawal must compensate for this negative reactivity. The control rods are normally kept almost fully withdrawn to maintain the axial flux difference (axial imbalance or ΔI at ONS) within its technical specification requirements and to optimize fuel utilization. Xenon buildup is normally compensated for by dilution. Since xenon buildup is a relatively slow process, this presents no significant problem for the reactor operator.

LOAD CHANGES AT POWER

Objectives 23, 24, 25

Once power passes the point of adding heat into the power range of operation, the combined effects of moderator temperature feedback and fuel temperature feedback cause the reactor characteristics to change. Now secondary system steam demand controls the steady state reactor power level. If power is stable, the $K_{eff} = 1$ and the net reactivity is zero.

Assume first that control rods are withdrawn without any change in secondary system steam demand. The rod withdrawal adds positive reactivity, K_{eff} becomes greater than 1, and power starts to increase. Since steam demand hasn't changed, the additional energy from the output from the core causes the moderator temperature to increase as fuel temperature increases. Both of these add negative reactivity. The power increase stops when the reactivities are balanced but power is still greater than demand. Moderator temperature continues to increase, adding negative reactivity and driving K_{eff} below 1. Power now decreases and fuel temperature decreases, adding positive reactivity. When power decreases back to match steam demand, moderator temperature stops increasing, and power stops decreasing. The positive reactivity added by the rod withdrawal is balanced by an increase in moderator temperature. Reactor power has returned to its original value.

Objective 26

For control rod insertions, the opposite of the above occurs, the insertion adds negative reactivity, K_{eff} becomes less than 1, power starts to decrease, moderator and fuel temperature decreases adding positive reactivity, the power decrease stops when the reactivity is balanced but power is still less than steam demand. Moderator temperature continues to decrease, adding positive reactivity, power increases to match steam demand, moderator temperature stops decreasing, and power remains constant, equal to steam demand. The negative reactivity added by the rod insertion is balanced by a decrease in moderator temperature.

Now assume that steam demand is increased with no control rod motion available. As steam flow increases, the additional steam necessary is produced by taking more energy from the Reactor Coolant System. This causes T_c to begin decreasing. The colder water returning to the core provides better neutron moderation, adding positive reactivity and causing reactor power to begin increasing. The increasing fuel temperature adds negative reactivity. Eventually, the positive reactivity added by the decrease in moderator temperature will be offset by the increase in fuel temperature and reactor power will be equal to steam demand. The negative reactivity associated with the increased power level completely balances the positive reactivity added by the reduced moderator temperature and the reactor is stable at a higher power level and lower moderator temperature.

Normally it is most economical to maintain the power level of nuclear generating plants as close to full rated power as possible and allow fossil-fueled plants to vary their output as the demand on the electric distribution system change. Often this is not possible, as when several nuclear plants are serving the same distribution system. When this is the case, the power level of the plant will vary with the electrical demand. Varying power level with external electrical demand is known as load follow.

The major concern during load follow operations is the asymmetric xenon distribution that can result from power changes. In order to prevent large distortions in the axial power profile which could lead to peaking factors outside of design limits, there are strict technical specification requirements that limit the allowed axial flux difference (ΔI). The indicated axial flux difference must be maintained within the allowable target band shown by the dotted lines on Figure 12, which is typical for a Westinghouse and a B&W plant. Normally, this is accomplished by maintaining control rods almost fully withdrawn from the core. Minimizing control rod movement during power operation minimizes axial flux shifts that can initiate xenon oscillations.

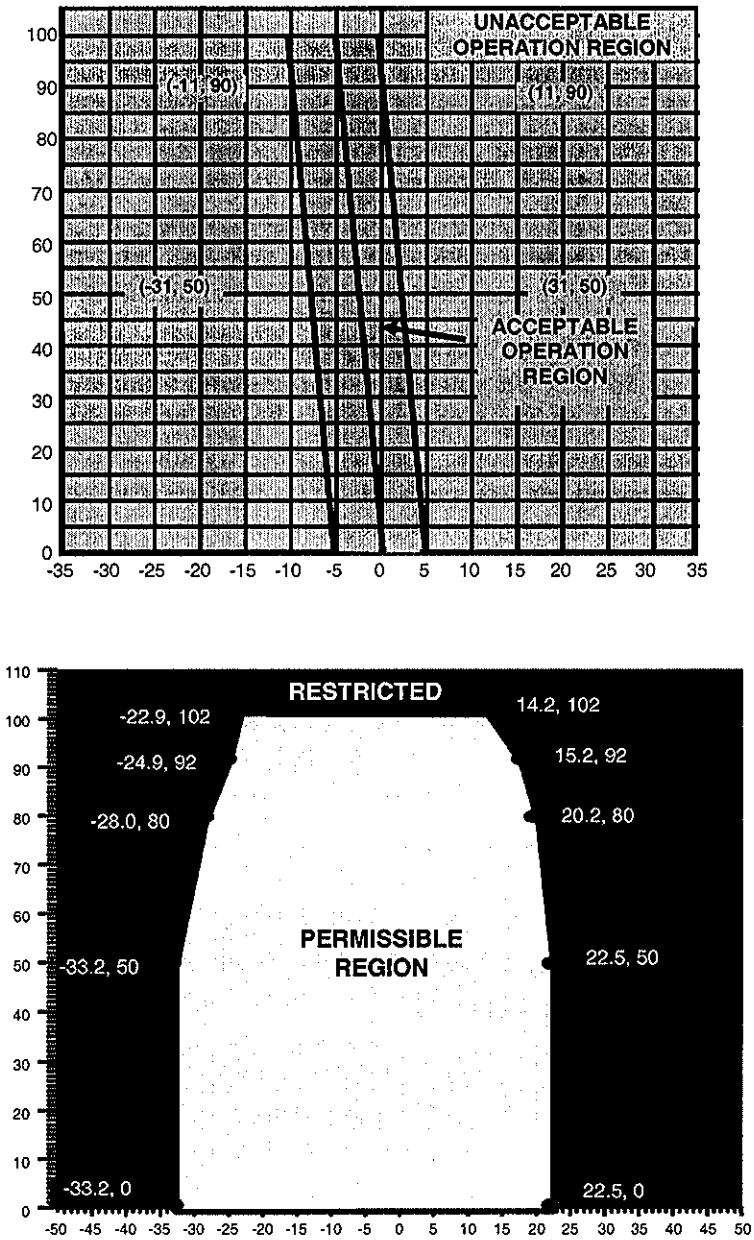


Figure 12 Westinghouse (top) and B&W (bottom) AFD allowable target bands

Reactivity changes during load follow operations are made by changing the RCS boron concentration. This helps keep the control rods out of the core. As reactor power changes, however, the axial flux will tend to shift because of moderator temperature reactivity effects. Control rods are used to maintain the axial flux distribution with the limits of Figure 12. Changing reactor power upsets equilibrium conditions and induces transient xenon behavior. The reactor operator must be aware of the direction of the resulting reactivity change and compensate with boron concentration adjustments to maintain RCS temperature. Rod movement in or out of the core for reactivity control is undesirable because it distorts the natural axial distribution of the flux and may initiate a xenon transient.

A typical load follow cycle is shown in Figure 13. This cycle (12 hours at 100%, 3-hour ramp to 50%, 6 hours at 50%, 3-hour ramp to 100%) is timed to coincide with demand on the distribution system: the 12-hour full-load period during high demand and the 6-hour half-load period during low demand. Such a cycle would be repeated daily as long as load follow operations are necessary, with adjustments in the ramp rates and power plateaus made as needed to meet demand (these adjustments are determined by the System Load Dispatcher and implemented by the on-duty crew).

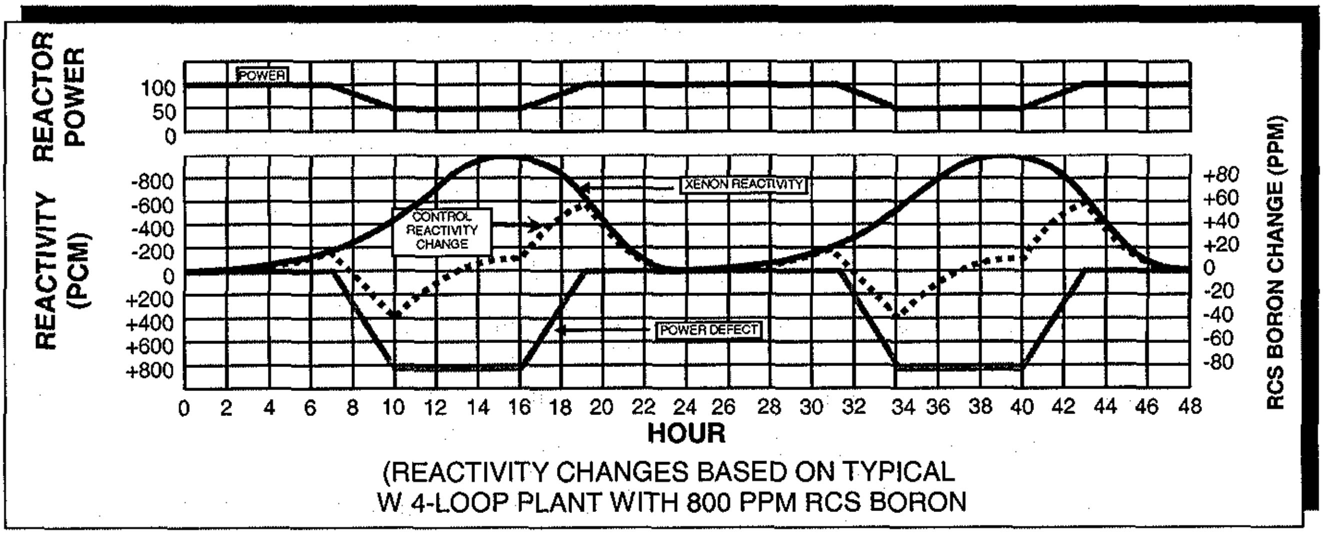


Figure 13 typical load follow cycle

As can be seen from the Net Reactivity Change graph in Figure 13, the load follow cycle involves ever-changing core reactivity, with fairly abrupt changes in direction. The reactor operator must closely monitor RCS temperature and adjust boron concentration accordingly. Recall that changes in RCS boron concentration are not immediate, they lag operator action by several minutes. The operator must be able to anticipate these changes to be able to maintain power and temperature stable.

Load follow operations at or near end-of- cycle are difficult for two reasons: the amount of water needed to dilute the RCS at low boron concentrations is very large and xenon transients are more likely to start and are more difficult to control at end of life. Therefore, towards end of core life, the water processing expense and the difficulty of controlling the axial flux distribution may preclude load follow.

BORON CONCENTRATION AND CORE LIFE

Objective 27

Boric acid is dissolved in the reactor coolant system to offset the excess reactivity associated with the extra fuel that is loaded to provide power operation over the design life of the core. The amount of dissolved boron to be used is determined by the excess reactivity remaining in the core after burnable poisons, in the form discrete rods or a boron coating of the fuel pellets, is established by core design. The use of boric acid and discrete burnable poisons reduces the amount of control rods needed to offset the excess reactivity at BOL. In addition, the dissolved boron and discrete burnable poisons allow for operation with all control rods fully withdrawn with the exception of the last control bank, which is almost fully withdrawn. This optimizes the neutron flux profile and minimizes power peaking, leading to lower power densities and even fuel burnup. The amount of boron dissolved in the reactor coolant system (and thus the amount of discrete poisons that must be used) is limited by its effect on the moderator temperature coefficient of reactivity. As was previously discussed, it is possible for the moderator temperature coefficient to become positive at low temperatures and high boron concentrations. The value of the moderator temperature coefficient is limited by Technical Specifications. Although Tech Specs allow for a slightly positive moderator temperature coefficient, it is a core design consideration to have a negative moderator temperature coefficient to provide inherent stability in conjunction with the fuel temperature coefficient. Over core life, the boron concentration is reduced to compensate for fuel burnup and the buildup of fission product poisons. Eventually, the boron concentration is reduced to a level where further dilution is insufficient to offset the fuel depletion and poison buildup and reactor power cannot be maintained at full power and normal operating temperature. At this point, the reactor is shut down and refueled.

REACTOR SHUTDOWN AND RCS COOLDOWN

A normal plant shutdown and cooldown are performed periodically for refueling or maintenance. Power reduction is performed by decreasing the external load on the turbine generator in conjunction with a boration of the RCS. This maintains control rod position and satisfies axial flux difference requirements and rod insertion limits. As power is decreased below 15 percent, the rods are placed to manual control and the reactor operator manipulates the rods as necessary to control RCS temperature. When the turbine generator load has been decreased to approximately 50 MW, the turbine is tripped and the control rods are positioned to maintain approximately two percent reactor power.

After power has been stabilized at approximately two percent, the reactor operator will record the information required for Reference Reactivity Data (RRD): power level, rod position, and actual boron concentration. This data will be used for calculation of shutdown margins and for subsequent ECPs. Once the RRD data has been recorded, the reactor is shutdown by fully inserting all control banks.

Prior to commencing the reactor cooldown, the RCS is borated to achieve the xenon free shutdown margin required by Technical Specifications for RCS temperature below 200°F (typically -1.0% $\Delta K/K$). Once this boration is completed and the RCS boron concentration has been verified by chemical analysis, the cooldown is performed. When cold shutdown conditions are reached, the shutdown margin is verified again. If adequate, the shutdown banks are fully inserted and the reactor trip breakers are opened.

RESPONSE TO A REACTOR TRIP

Objective 28

The actions taken by reactor operators following a reactor trip are dictated by approved station procedures. The procedures ensure that the reactor is shut down, the turbine is tripped, normal and/or emergency power sources are available and the plant response is as expected. If needed, compensatory actions are taken in accordance with the procedures.

After the initial prompt drop, reactor power decreases at a rate determined by the decay of DNP groups, eventually stabilizing at –1/3 DPM. The –1/3 DPM is determined by the decay constant of the longest-lived delayed neutron precursor group. The operator monitors the –1/3 DPM startup rate into the source range and verifies activation of the source range instruments. See Figure 14 below for a plot of reactor power versus time following a reactor trip.

Core thermal power will remain high for several seconds after the trip. There is a time lag of a few seconds for the heat generated in the fuel to be conducted into the coolant and the decay heat immediately following the prompt drop is approximately 7% of rated thermal power, assuming a trip from equilibrium full power operation. RCS temperature will be reduced by the steam dump system and will stabilize at no-load T_{avg}. Ten seconds after the trip, decay heat is still approximately 5% RTP and decreases to about 1% RTP in about eight hours.

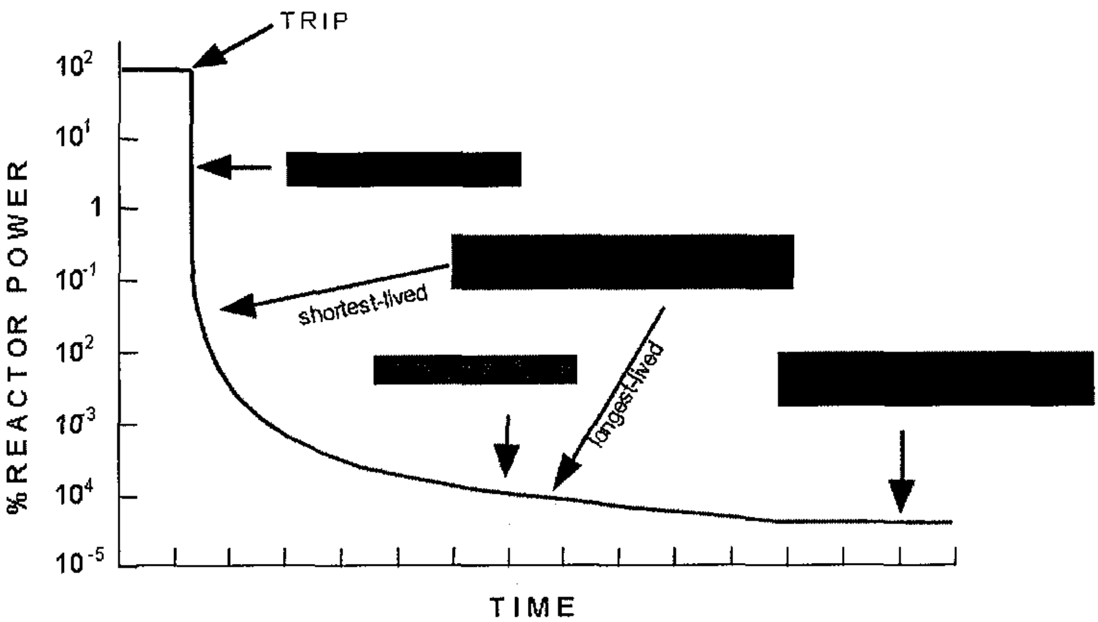


Figure 14 Reactor power versus time following a reactor trip

DECAY HEAT

Objective 29

Approximately 200 MeV of energy is released per fission event. The majority of this energy is in the form of the kinetic energy of the fission fragments and fission neutrons and appears instantaneously. Approximately 6-7% of the energy is released some time later as the fission products decay. This energy is absorbed by the surrounding fuel matrix and core materials and is known as decay heat.

Objectives 30A, B, & C

Once the reactor is shut down and the fission rate is very low, fission fragments continue to decay, releasing heat to the reactor core. The number of fission fragments undergoing decay is determined by the number of fissions and the time since they occurred, therefore the amount of decay heat being released in the core depends on core power history, including time since shutdown.

Figure 15 shows the decay heat for a uranium-fueled reactor versus time, assuming an infinite operating time (i.e., all fission products are in equilibrium). For decay heat considerations, a power reactor operating at full power for about a month can be considered to have been operating infinitely. For a 3000 MW_t power reactor, heat generation in the core is 45-50 MW one hour after a trip from an extended run at full power. This is more than enough heat to damage the fuel if cooling is lost. This illustrates why shutdown heat removal is so important, not just after a trip or shutdown, but literally for months afterward.

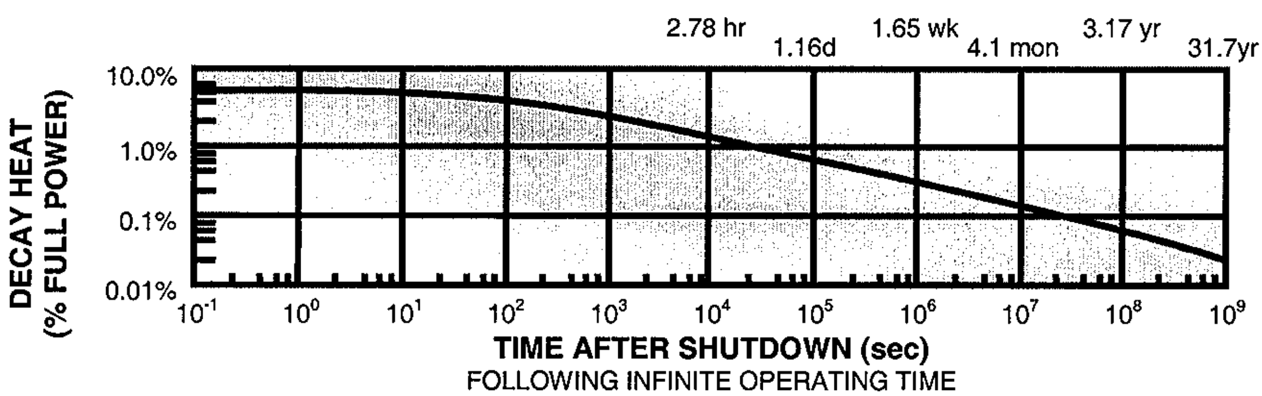


Figure 15 Decay heat versus time

The amount of decay heat produced following a reactor shutdown or trip depends upon the power history of the core. If it has operated for a long period of time at a high power the fission product inventory will be greater than if it had operated for a short period of time at low power. The percentage of decay heat at some time can be estimated from the following table:

Time after Trip from Full Power	**% of Full Power Generated as Decay Heat**
1 second	6.0
1 minute	4.5
30 minutes	2.0
1 hour	1.6
8 hours	1.0
24 hours	0.7
48 hours	0.6
1 week	0.5
1 month	0.1

Reference: Westinghouse Electric Corp., Thermal-Hydraulic Principles and Applications to the Pressurized Water Reactor, 1985

Table 1

GLOSSARY	
Critical -	The reactor is said to be critical as indicated by a constant positive SUR and a steady increasing count rate with no positive reactivity being added to the core.
Decay Heat -	Heat generated by the fuel after shutdown from the decay of fission products and their daughters.
Estimated Critical Position (ECP) -	Mathematical calculation in which moderator temperature, power history, time since shutdown, core age, boron concentration and control rod worth are used to predict the rod position at criticality.
Estimated Critical Boron (ECB) -	Mathematical calculation in which moderator temperature, power history, time since shutdown, core age, boron concentration and control rod worth are used to predict the boron concentration at criticality for a desired critical rod position.
1/M Plot -	A plot using the inverse count ratio following reactivity addition events to obtain a conservative estimate of critical rod position. May also be used during fuel loading to monitor for inadvertent criticality.
Point of Adding Heat (POAH) -	Point where heat production from fission becomes significant enough to overcome ambient losses.

OPERATING EXPERIENCE

SOER/SER 88-2, 91-24, 96-10, 90-3, 88-3, 86-16, 96-15

REFERENCES

INPO Case Study Materials 87-015, "Control Rod Mispositioning and Reactivity Events"

SOER 88-2 Premature Criticality

Objective 31

SIGNIFICANT OPERATING

EXPERIENCE REPORT: 88-2

PREMATURE CRITICALITY EVENTS DURING REACTOR STARTUP

EVENTS:

UNIT (TYPE): V.C. SUMMER (PWR) MCGUIRE 2 (PWR)

DOC NO/LER NO: 50-395/85003 50-370/NA

EVENT DATE: 2/28/85 5/17/85

NSSS/AE: WESTINGHOUSE/GILBERT WESTINGHOUSE/DUKE POWER COMPANY

UNIT (TYPE): SAN ONOFRE 3 (PWR)

DOC NO/LER NO: 50-362/NA

EVENT DATE: 4/13/86

NSSS/AE: COMBUSTION ENGINEERING/
BECHTEL COMPANY

UNIT (TYPE): VOGTLE 1 (PWR)

DOC NO/LER NO: 50-424/87032

EVENT DATE: 6/6/87

NSSS/AE: WESTINGHOUSE/BECHTEL/
SOUTHERN SERVICES

UNIT (TYPE): PALO VERDE 1 (PWR)

DOC NO/LER NO: 50-528/NA

EVENT DATE 5/14/88

NSSS/AE: COMBUSTION ENGINEERING/BECHTEL

SUMMARY:

Several premature criticality events have occurred at commercial nuclear plants in recent years. Some of these unanticipated criticalities were not recognized by the reactor operators and resulted in automatic reactor scrams. During one event, the reactor power rate of increase had reached 17 decades per minute (a 1.6 second period) at the time of the reactor scram. Factors contributing to these events included failure to monitor all pertinent indications, inaccurate core reactivity information, the use of trainees without proper supervision, and inaccuracies or lack of sufficient detail in procedures governing the approach to criticality. Methods of calculating proximity to criticality, such as inverse count rate ratio (1/M) plots or count rate doubling data, were not used. In most cases, the audio count rate speakers also were not in use.

Inadequate control of reactivity has the potential for damaging the core. Reliance on safety system actuations reduces the margin of safety. If the reactor protection system were to fail or the associated scram breakers delay opening (as has happened), significant power excursions can occur in a very short time period, with resultant core damage.

DESCRIPTION:

V. C. SUMMER

The plant had been operating near full power when it was shut down for rod control system repairs. An estimated critical condition calculation was performed, and after being shut down for 23 hours, the unit was brought critical for about three hours and then shut down again.

Approximately seven hours after the second shutdown, another estimated critical condition calculation was performed using the power history and other data from the earlier criticality. The estimated critical rod position was calculated to be 168 steps on Bank D.

During the subsequent startup, a trainee was manipulating control rods under the direct supervision of the shift supervisor. The shift supervisor planned to stop rod motion at about 100 steps on Bank D to check rod position indication and nuclear instrumentation. As the trainee pulled rods, the shift supervisor did not notice the rapidly increasing source range count rate. The estimated critical rod position was in error, and the reactor scrammed on high source range flux, with rods at 76 steps on Bank D. Analysis after the event revealed that criticality occurred at about 40 steps on Bank D, and the startup rate had reached approximately 17 decades per minute. A review of general operating procedures disclosed that the method used to calculate the estimated critical condition was not sufficiently accurate under non-equilibrium xenon and samarium conditions.

Because the power level had been low and the operating time had been brief during the previous criticality, the estimated critical condition calculation procedure did not produce an accurate average power for the xenon and samarium calculations. Additionally, the licensed operator performing the estimated critical condition calculations had not used the correct curves when determining xenon and samarium reactivity. The plant curve book provides a set of curves for three stages of core life (beginning, middle, and end). The operators used the middle-of-life curves, but the beginning-of-life curves more closely approximated fuel exposure history. This error contributed to the magnitude of the calculation errors.

The procedural inaccuracy and use of the incorrect curves did not fully account for the discrepancy between the calculated and actual critical rod position. At the utility's request, the vendor verified that an accurate estimated critical condition could be calculated using the computer codes that were in use at this and other units.

However, using a new, more refined technique, the vendor developed more accurate data relating reactivity to the previous reactor power history. This technique consistently produces a more accurate estimated critical condition.

MCGUIRE 2

McGuire 2 had been operating at 100 percent power when the unit was scrammed manually due to a main generator hydrogen leak. Repairs were completed later that day, and preparations were made for startup. An estimated critical condition calculation was performed using a computer program. The critical rod position was calculated to be 38 steps on Bank D control rods.

Operators began control rod withdrawal and were withdrawing Bank C when they observed startup rate indications that suggested that the reactor was approaching critical-ity. Rod motion was stopped, and it was determined that the unit had achieved criticality at 26 steps on Bank C. This critical rod position was 126 steps and about 1000 pcm (1 percent delta k/k) below the calculated estimated critical condition rod position. Criticality had occurred 21 steps below the minimum rod insertion limits, but shutdown margin was greater than required. All rods were immediately inserted as required by procedure.

Investigation disclosed that the computer program used to estimate critical rod position provided an incorrect xenon worth. The program had been revised during the previous fuel cycle, and engineering personnel had incorrectly read a control rod position, which led to entering an excessive xenon worth constant.

The program had been used several times with acceptable results; however, in each of these previous uses, xenon had not been a factor in reactivity calculations due to the power history of the reactor. In this event, the program failed to account for the transient xenon conditions.

SAN ONOFRE 3

The unit startup was being conducted to achieve criticality approximately 18 hours following a scram from 100 percent power. This was the first reactor startup conducted with significant xenon present in the core during cycle 2. The startup was performed by a non-licensed operator trainee under the direct supervision of a licensed assistant control operator. During the Unit 3 startup, the shift super-intendent was involved in Unit 2 operations. The Unit 3 startup proceeded normally during withdrawal of the two shutdown rod groups and part-length groups.

During withdrawal of the regulating rod groups, the trainee and the assistant control operator used the startup rate meter as the primary indicator of approach to criticality. Log power level and startup count rate were recorded on a chart-type recorder, but the chart was small and moved slowly, making data trending difficult.

Because the control room supervisor thought that criticality would not be achieved for approximately 15 to 20 minutes, he began briefing his relief on plant status. While the control room supervisor was engaged in discussions with his relief, the trainee continued pulling rods. The reactor achieved criticality at approximately 80 inches on Group 4, which was 280 inches of collective group rod travel earlier than the estimated critical condition. Criticality occurred with reactor power at approximately 1x10-5 percent.

This is roughly equivalent to an intermediate range nuclear instrument reading of 1x10-11 amps for a Babcock & Wilcox design and 5x10-11 amps for a Westinghouse design. Rod withdrawal continued to approximately 114 inches on Group 4. The alarm, "Hi Log Power Level Bypass Permissive," occurred at 1x10-4 percent power. The control room supervisor directed the control operator's attention to this alarm and accompanied him to the control board to take the necessary bypass action.

At this point, the control room supervisor realized that the reactor had achieved criticality. The trainee also recognized criticality based on a sustained startup rate in excess of 1 decade per minute with no rod motion. Control rods were inserted, and the reactor coolant system was borated. Because of rod insertion the core protection calculator determined an unsatisfactory relationship existed between rod position and sensed power and caused a reactor scram.

Subsequent to the event, it was determined that the Unit 3 physics data for the time-dependent effect of xenon had not been updated to include changes found to be necessary for Unit 2. This resulted in an error in the estimated critical condition.

VOGTLE 1

The reactor was being taken critical approximately 37 hours after a scram from 75 percent power. The estimated critical condition was calculated to be 45 steps on Bank D, with a boron concentration of 1120 ppm. Criticality occurred at about 80 steps on Bank C, an error of about 600 pcm (0.6 percent delta k/k), with a boron concentration of 1120 ppm.

The startup was being performed by an unlicensed trainee under the supervision of the reactor operator. Control rod withdrawal was being performed by procedure, which required pausing every 50 steps to allow counts to stabilize and to monitor rod alignment. The reactor operator and trainee were using the source range counts recorder trace (on slow speed) as the primary means to evaluate reactor response. Shift personnel judged the source range counts and startup rate meters to be too erratic to be read reliably. The computer display of these parameters was not used by the operators since they would have to turn away from the control panels to see the display.

Control rods were withdrawn from 20 steps to 70 steps on Bank C. The reactor operator and trainee determined the count rate had stabilized and had begun another 50 step withdrawal to 120 steps on Bank C when a low pressure alarm on a safety injection accumulator was received. The low pressure condition placed the unit in a limiting condition for operation, and the shift supervisor directed the trainee to stop rod withdrawal. Post-event analysis of data indicates rod position was approximately 80 steps on Bank C at this point, and that reactor criticality had been achieved but had not been detected by the operator.

After about a one-minute delay the shift supervisor decided to proceed with the startup and assigned another operator to respond to the accumulator pressure alarm. The reactor operator and trainee then observed the source range counts recorder and incorrectly interpreted counts as stable when in fact the counts continued to increase without rod motion. The reactor operator directed the trainee to proceed with the startup.

The reactor operator then became involved with startup temperature logs and did not supervise the trainee or observe reactor response. Additionally, the shift supervisor had turned his attention to logging the accumulator pressure alarm and the associated limiting condition for operation and did not observe the trainee's activities, rod control system operation, or reactor response.

The trainee withdrew Bank C control rods about 20 steps in a region of relatively high rod worth, and startup rate increased rapidly. The trainee observed the rapidly increasing source range counts but did not interpret it as excessive. The intermediate range nuclear instrument current increased rapidly, and at 1x10-10 amps, the P-6 permissive interlock bistable indicators lit. This permissive interlock allows manual blocking of the source range high flux scram. The reactor scrammed on source range high flux about nine seconds later.

Subsequent to the event, it was determined that the estimated critical condition calculation procedure did not provide adequate instruction in the use of the boron worth tables. In addition, the procedure did not provide sufficient accuracy in determining the boron and samarium correction factors for boron worth. Several erroneous estimated critical condition calculations had occurred prior to this event. For example, one month earlier an estimated critical condition calculation had been in error by about 750 pcm (0.75 percent delta k/k). Operations management and the on-shift operations supervisors were aware of the previous estimated critical condition errors, but the reactor operator was not.

PALO VERDE 1

Unit 1 achieved criticality with reactivity conditions about 850 pcm (0.85 percent delta k/k) before the estimated critical condition. The reactor went critical on control rod Group 3 at 60 inches of travel. The startup rate was about 0.25 decades per minute. Reactor power increased to 1x10-3 percent power (at 1x10-4 percent power the core protection calculator trip was enabled).

The allowable minimum rod height to ensure adequate shutdown margin (power dependent insertion limit) is control rod Group 3 at 60 inches withdrawn. The operators determined this limit was not met and decided to reinsert control rods and assess the reactivity conditions. (Later calculations determined that the reactor actually achieved criticality at 55 inches, and that adequate shutdown margin had been maintained.) Because of this rod insertion the core protection calculator determined an unsatisfactory relationship existed between rod position and sensed power and caused a reactor scram when Group 3 was inserted to 25 inches. The operators monitored count rate throughout the approach to criticality and were aware of the proximity to the power dependent insertion limit.

By procedure, an inverse count rate ratio plot is required to be started when Group 3 is at 60 inches withdrawn. This plot was not performed because the initiating conditions had not been met. The error in the estimated critical condition resulted from the following:

- An inaccuracy in the computer program that calculates xenon level resulted in a 270 pcm error.
- Failure to calculate a new estimated critical condition after a 3.5 hour deviation from the calculated startup time – Xenon decay during this time period resulted in a 340 pcm error.

Note: About one month earlier, a startup when the reactor had appreciable xenon achieved criticality about 450 pcm before the predicted critical condition. Reactor engineering was resolving this discrepancy at the time of this reactor scram.

SIGNIFICANCE:

These events are significant because they demonstrate a lack of control over reactivity changes. Inadequate control of reactivity has the potential for damaging the core. Although reactor protection system features are designed to prevent core damage during startup reactivity transients, these events unnecessarily challenge the reactor protection system. If the reactor protection system (or its associated scram breakers) were to fail, significant power excursions could occur in a very short time period.

ANALYSIS/DISCUSSION:

Use of Available Instrumentation

Failure to correctly use or interpret indications of approaching criticality contributed to several of the events described above. In the V. C. Summer event, the rapidly increasing count rate went unnoticed by the shift supervisor. In the Vogtle event, the source range count rate and startup rate meters were judged to be erratic, and the computer display of these parameters was not used. Also, the source range counts chart recorder was in slow speed, making it difficult to trend and evaluate reactor response. In addition, the audio count rate speaker was not in use.

The tendency to become fixed on one indication during startup must be avoided. All available instrumentation should be aligned and scanned periodically during rod withdrawal to provide indication of approaching criticality.

OPERATOR AND SUPERVISORY ATTENTION

Operators have repeatedly failed to anticipate criticality when pulling control rods but not yet near the estimated critical condition. During large xenon transients, critical conditions are difficult to estimate accurately and are subject to large errors. Operators should not rely on the critical rod position predicted by the estimated critical condition calculation to guide rod withdrawal during startup. Operators should anticipate criticality whenever rods are being withdrawn and should understand and use all available indications of neutron flux level and rate of change to effectively monitor the approach to criticality.

Methods such as the inverse count rate ratio plot and count rate doubling use actual neutron flux level changes to monitor changing core reactivity conditions during rod withdrawal. Periodically stopping rod motion and allowing subcritical multiplication to stabilize flux level during the startup will permit effective monitoring of the approach to criticality. In addition, gross errors in the calculated estimated critical condition can be detected early enough to allow corrective action before achieving criticality.

In four of the seven events discussed above, trainees were involved in pulling control rods or monitoring nuclear instrumentation. Conduct of shift turnover during the approach to criticality provided an additional distraction to personnel in two of the events. During a reactor startup, a large amount of positive reactivity is added within a relatively short time period; thus, strict operator and supervisory attention is required. Distractions, especially when trainees are involved in performing the startup, should be avoided.

ESTIMATED CRITICAL CONDITION

In all six of the PWR premature criticality events described above, an inaccurate calculation of estimated critical condition was a contributing factor. Inaccurate estimation of xenon reactivity under non-equilibrium conditions contributed to the estimated critical condition error in five of these events.

The primary purpose of the estimated critical condition calculation in a PWR is to confirm expected core reactivity characteristics for a given set of conditions, such as core age, power history, and temperature. Based on this calculation, boron concentration is adjusted as necessary to ensure that an acceptable rod position will exist when criticality is achieved.

Estimated critical condition calculations usually provide a reasonably accurate prediction of critical rod position. However, inaccuracies in manual or computerized calcula-tion methods under non-equilibrium xenon conditions have resulted in significant errors at several plants. The process used to generate data for these calculations should be periodically verified to be accurate.

GLOSSARY	
Critical -	The reactor is said to be critical as indicated by a constant positive SUR and a steady increasing count rate with no positive reactivity being added to the core.
Decay Heat -	Heat generated by the fuel after shutdown from the decay of fission products and their daughters.
Estimated Critical Position (ECP) -	Mathematical calculation in which moderator temperature, power history, time since shutdown, core age, boron concentration and control rod worth are used to predict the rod position at criticality.
Estimated Critical Boron (ECB) -	Mathematical calculation in which moderator temperature, power history, time since shutdown, core age, boron concentration and control rod worth are used to predict the boron concentration at criticality for a desired critical rod position.
1/M Plot -	A plot using the inverse count ratio following reactivity addition events to obtain a conservative estimate of critical rod position. May also be used during fuel loading to monitor for inadvertent criticality.
Point of Adding Heat (POAH) -	Point where heat production from fission becomes significant enough to overcome ambient losses.

EXAMPLE QUESTIONS:

1. The reactor is critical at 10^{-8} amps. To increase power to the point of adding heat, the control operator should perform which of the following actions?

   A. Increase steam demand.

   B. Withdraw control rods.

   C. Increase turbine load.

   D. Decrease coolant temperature.

2. During a reactor startup from a Xenon-free condition, after recording critical data, the operator establishes a positive startup rate to continue increasing power. Soon power begins leveling off and the startup rate decreases toward zero, well below the expected power level for the point of adding heat. A probable cause for this is:

   A. Xenon buildup in the core.

   B. inadvertent boration.

   C. integral rod worth decreases as power increases.

   D. shutdown margin increases as rods are withdrawn.

3. The "point of adding heat" (POAH) is defined as that power level where the reactor is producing enough heat:

   A. for doppler coefficient to produce a positive reactivity feedback.

   B. for void coefficient to produce a positive reactivity feedback.

   C. to cause an observable temperature increase in fuel and coolant.

   D. to support main turbine operations.

4. After recording critical data during a reactor startup, the reactor operator withdraws the control rods to continue the startup. Along with the reactor coolant system (RCS) temperature, which one of the following two parameters are monitored to determine when the reactor reaches the "point of adding heat" (POAH)?

   A. RCS pressure, steam generator level

   B. Reactor power, reactor startup rate

   C. Steam pressure, turbine load

   D. RCS flow, presssurizer level

5. During a reactor startup, reactor startup rate response to reaching the "point of adding heat" is described as:

   A. increasing due to increasing turbine load.

   B. increasing due to decreasing reactor coolant system (RCS) flow.

   C. decreasing due to decreasing RCS pressure.

   D. decreasing due to increasing fuel temperature.

6. The point of adding heat is that power level where:

   A. fuel temperature equals coolant temperature.

   B. nuclear heat is observable to the operator.

   C. fuel centerline temperature approaches design limits.

   D. nuclear heat is sufficient to cause clad creep.

7. Which one of the following parameters most significantly affects reactivity in a critical reactor below the point of adding heat?

   A. Coolant inventory

   B. Control rod position

   C. Reactor coolant pressure

   D. Core flux level

8. If positive reactivity is added to a critical reactor well below the point of adding heat, power will increase:

   A. only during the reactivity addition.

   B. linearly.

   C. asymptotically.

   D. exponentially.

9. A reactor has just achieved criticality. A subsequent 5 second rod withdrawal is performed. After the initial prompt jump, the neutron population:

   A. increases linearly.

   B. increases exponentially.

   C. decreases to the original value.

   D. stays constant.

10. A reactor is critical well below the point of adding heat when a small amount of positive reactivity is added to the core. If the same amount of negative reactivity is then added to the core, reactor power will level off:

    A. somewhat higher than the initial power level.

    B. somewhat lower than the initial power level.

    C. at the initial power level.

    D. at the subcritical multiplication equilibrium level.

11. During a reactor startup, after reaching criticality, the operator has established a positive startup rate. Upon reaching the point of adding heat, the startup rate should:

    A. double.

    B. increase slowly.

    C. not change until rods are inserted.

    D. decrease.

12. During reactor startup, after recording critical data, the operator inserts enough positive reactivity into the core to cause power to increase. After a few minutes, power begins to level off without operator action. This is an indication of reaching the:

    A. rated thermal power.

    B. DNBR safety limits.

    C. design limits for fuel centerline temperature.

    D. point of adding heat.

13. Which of the following parameters is an indication of reactor power reaching the point of adding heat?

    A. Boron concentration increase

    B. Startup rate increase

    C. Pressurizer level increase

    D. Power level increase

14. After taking critical data during a reactor startup, the operator establishes a 1 decade per minute startup rate (26 second reactor period) to increase power to the point of adding heat. How much negative reactivity must be added by doppler and moderator temperature coefficients to stop the power increase at POAH?

Assume $\lambda_{eff} = 0.1\ sec^{-1}$ $\beta_{eff} = 0.00579$

A. 0.10%Δk/k or 100 pcm

B. 0.16% Δk/k or 160 pcm

C. 1.0% Δk/k or 1,000 pcm

D. 1.6% Δk/k or 1,600 pcm

15. During a reactor startup, the first coefficient to begin inserting negative reactivity into the core at the point of adding heat is:

A. doppler coefficient.

B. void coefficient.

C. moderator temperature coefficient.

D. pressure coefficient.

16. During a xenon-free reactor startup, critical data were inadvertently taken two decades below the required intermediate range (IR) level. The critical data were taken again at the proper IR level with the same reactor coolant temperatures and boron concentration.

The critical rod position taken at the proper IR level ________ the critical rod position taken two decades below the proper IR level.

A. is less than

B. is the same as

C. is greater than

D. cannot be compared to

17. The reactor is critical below the point of adding heat with a startup rate of 0.5 decades per minute. Which one of the following will decrease FIRST when the reactor reaches the point of adding heat?

A. Pressurizer level

B. RCS temperature

C. Reactor power

D. Startup rate

18. Given a critical reactor operating below the point of adding heat, what reactivity effects are associated with reaching the point of adding heat?

    A. There are no reactivity effects since the reactor is critical.

    B. The increase in fuel temperature will begin to create a positive reactivity effect.

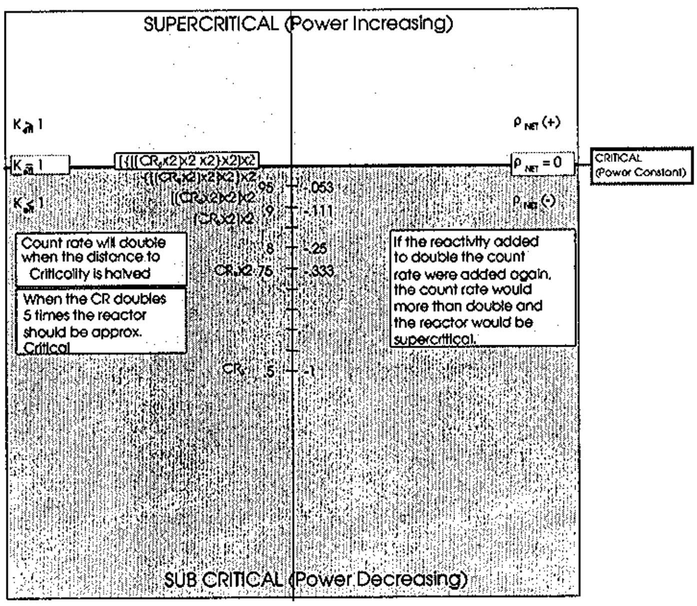
    C. The decrease in fuel temperature will begin to create a negative reactivity effect.

    D. The increase in fuel temperature will begin to create a negative reactivity effect.

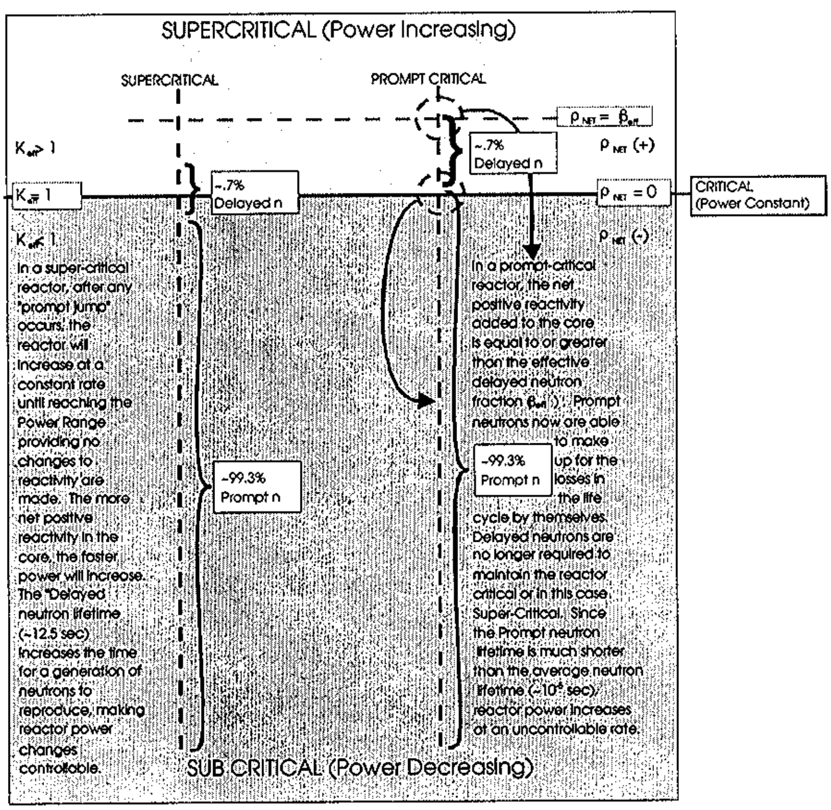
ANSWERS

1. B
2. B
3. C
4. B
5. D
6. B
7. B
8. D
9. B
10. A
11. D
12. D
13. C
14. B
15. A
16. B
17. D
18. D

K/eff ρ THUMBRULES

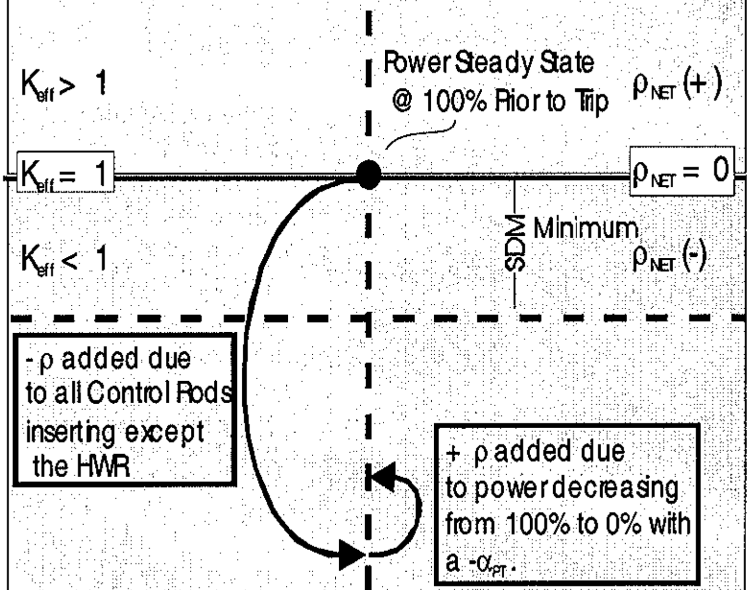


K/ eff ρ PROMPT CRITICAL



K_{eff} / ρ REACTOR TRIP

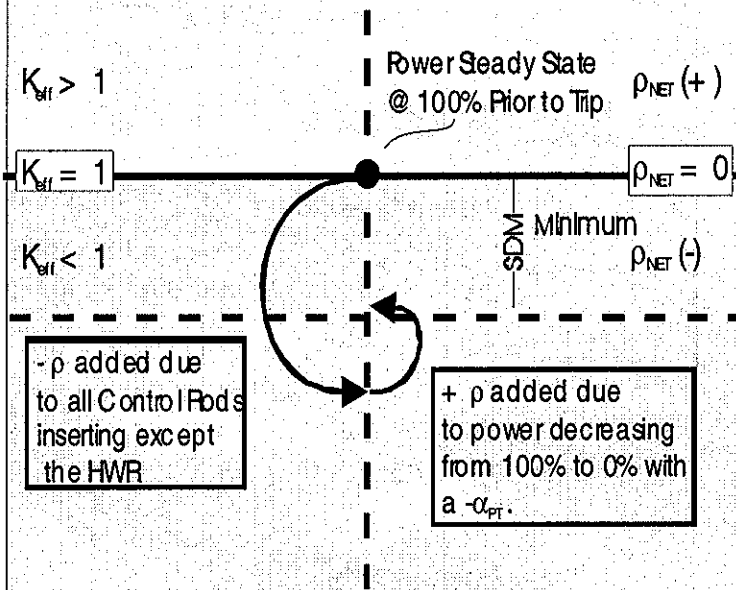
SUPERCRITICAL (Power Increasing)



With CRs above the rod insertion limits prior to reaching criticality, enough negative reactivity is added by the CR during a trip (inc. the HWR stuck fully out) to shut down the reactor to a level below the minimum SDM.

SUBCRITICAL (Power Decreasing)

SUPERCRITICAL (Power Increasing)



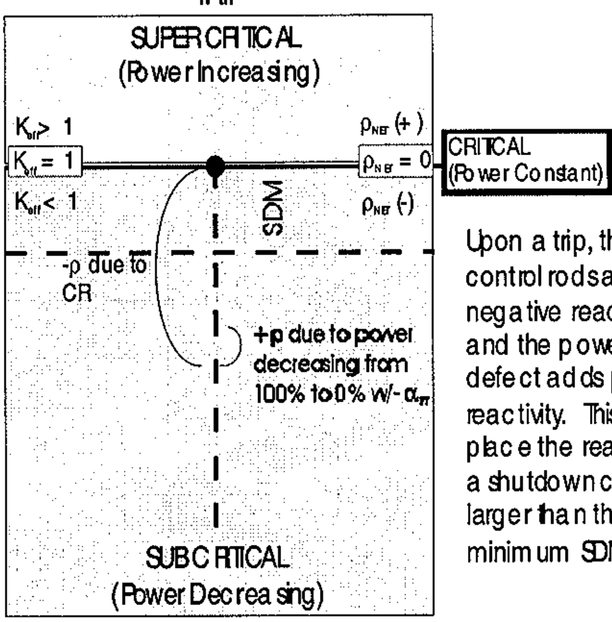
With the CRs below the rod insertion limits when criticality is obtained a reactor trip may not ensure that the reactor will be shutdown below the minimum shutdown margin. This condition could be achieved by having too low of a boron concentration during a start up. Excessively diluting boron adds positive reactivity that should be added by withdrawing control rods. At startup, there is less negative reactivity in the core therefore, less positive reactivity is needed to bring the reactor critical. This results in a lower rod height at criticality and therefore a smaller amount of negative reactivity added upon a trip.

SUBCRITICAL (Power Decreasing)

KP-3

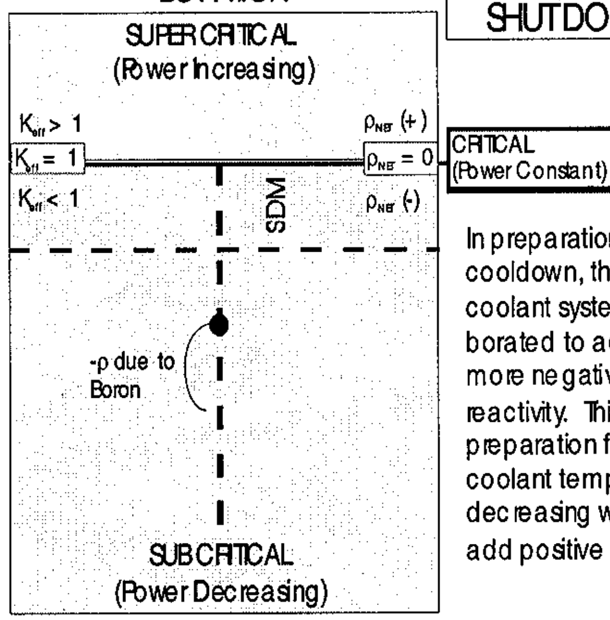
K_{eff} / ρ SHUTDOWN

TRIP



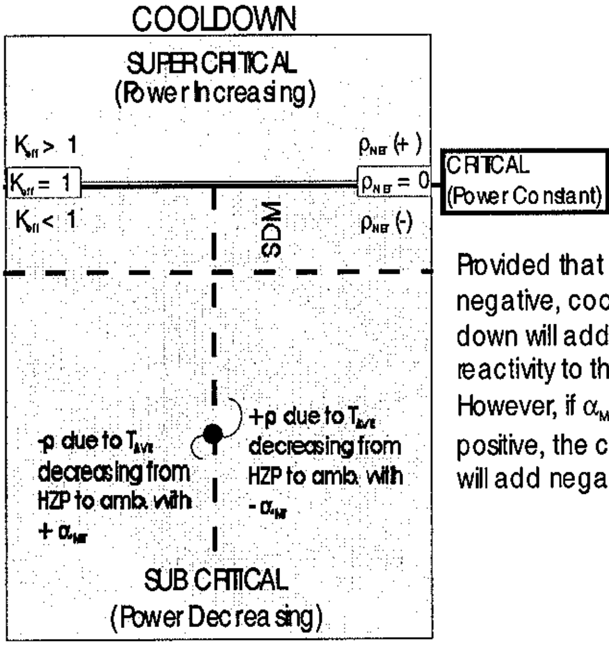
Upon a trip, the control rods add negative reactivity and the power defect adds positive reactivity. This should place the reactor in a shutdown condition larger than the minimum SDM.

BORATION



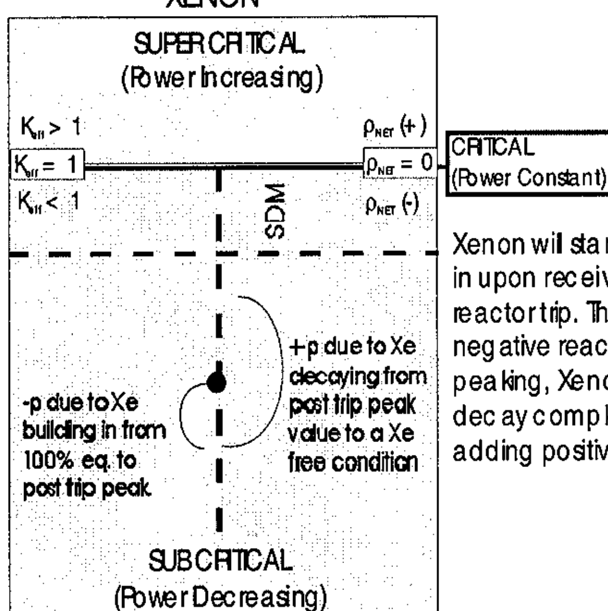
In preparation for cooldown, the coolant system is borated to add more negative reactivity. This is in preparation for reactor coolant temperature decreasing which will add positive reactivity.

COOLDOWN



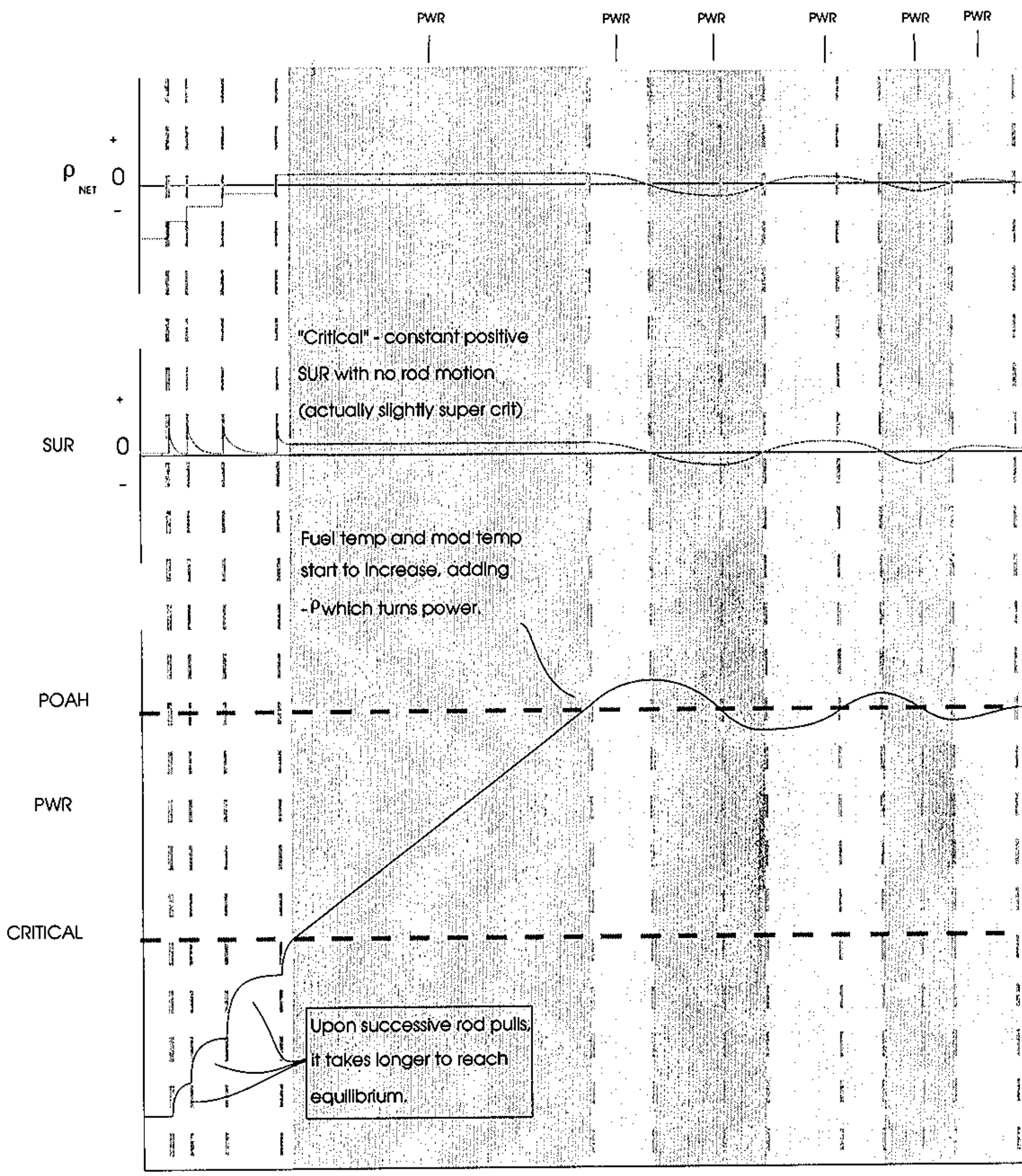
Provided that α_{MT} is negative, cooling down will add positive reactivity to the core. However, if α_{MT} is positive, the cooldown will add negative.

XENON



Xenon will start to build in upon receiving a reactor trip. This will add negative reactivity. After peaking, Xenon will then decay completely, adding positive reactivity.

KP-4



Once power increases above the POAH, T_{Fuel} will provide **immediate** feedback (ρ) to turn power. T_{Mod} will also increase and (if undermoderated), add ($-\rho$) but, moderator temperature has a partial time lag due to loop transient time. This time lag causes the reactivity feedback from T_{Mod} to overshoot. This results in power oscillating around the POAH. After reaching a steady state condition, the Rx will be Critical, Power and T_{Mod} will be at the POAH, and the (ρ+) inserted by the control rods will be compensated for by an increase in T_{Mod}

POWER ASCENSION

SRO Question # 16 *Bank Question: 1054* *Answer: C*

1 Pt Given the following conditions on Unit 1:

- Unit 1 is at 100% power.
- 'A', 'B', and 'C' VL AHU are running
- 'A' and 'C' VL AHUs have tripped and will not restart
- Attempts to start 'D' VL AHU were unsuccessful
- Average temperature in lower containment for past 365 days has been 105 degrees.
- Maintenance indicated it will take two days to repair the VL AHUs.
- Containment lower compartment temperature is 126 degrees and steady.

Which one (1) of the following describes the required Technical Specification actions to address the high containment temperature?

Reference Provided Tech Spec 3.6.5

A. Restore temperature to within limits in 8 hours.

B. Reduce temperature to <125 degrees in 72 hours.

C. No action is required to address high containment temperature.

D. Be in Mode 3 in 14 hours.

Distracter Analysis:

A. Incorrect: Plausible.
B. Incorrect: Plausible.
C. Correct.
D. Incorrect: Plausible.

LEVEL: SRO
KA: SYS 022 G2.1.12 (2.9/4.0)
SOURCE: NEW
LEVEL OF KNOWLEDGE: Analysis
AUTHOR: CWS
LESSON: OP-MC-CNT-VUL
OBJECTIVES: OP-MC-CNT-VUL Obj. 11
REFERENCES: OP-MC-CNT-VUL pages 15 & 17
Tech Spec 3.6.5
OBJECTIVES: OP-MC-ADM-TS Obj. 2
REFERENCES: Tech Spec 3.6.5 and Bases
OP-MC-ADM-TS pages 29, 31

Comment

This question accurately determines a senior operator's ability to analyze plant conditions and apply the appropriate technical specification. By the literal interpretation of the technical specification, answer "C" is the correct answer, and the operator is not required to take any action to address the high containment temperature.

However, the operational philosophy at McGuire Nuclear Station is to take conservative action to ensure the plant is operated well within the technical specification limiting conditions for operation and other plant operating documents. As such, answer "B" would be a conservative, expected action taken at McGuire Nuclear Station to ensure containment temperature is maintained within allowable guidelines.

Proposed Resolution

The facility recommends that answers "B" and "C" both be accepted as a correct answer to this question.

Justification

As part of the briefing prior to taking the written examination, the candidates were read Appendix E, *Polices and Guidelines for Taking NRC Examinations*, Part B, Written Examination Guidelines. The information contained in the briefing instructed operators to answer questions based on the actual plant.

The expectations at McGuire Nuclear station are that all available actions will be implemented to reduce Containment temperature to less than 125 degrees as soon as possible. Procedures are in place for the operators to respond to Containment temperature if it exceeds 120 degrees. One procedure addresses all available equipment related to the Containment Cooling Fans (VL). It directs the operator to place all available fans in service, at their maximum capabilities to limit the temperature increase. That same procedure directs the operator to place additional Containment Cooling Water pumps (RV) in service to assist in limiting the temperature increase. We have also provided the operator with an additional procedure which aligns the Nuclear Service Water system (RN) to the Standby Nuclear Service Water Pond (SNSWP). This supply is typically at a lower water temperature than the normal RN supply, Low Level Intake (LLI). Once aligned to the SNSWP, the operator aligns the RN system Non-Essential Header such that the Containment Cooling Fans (VL) is supplied from that cooler water source (SNSWP). This RN alignment is also utilized to supply the Reactor Coolant Pump Motor coolers during times where the motor coolers are increasing above the normal range.

Plant references are included with this comment package:

- OP/1/A/6450/001, Containment System Ventilation
- OP/1/A/6400/006, Nuclear Service Water

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be:

a. ≥ 75°F and ≤ 100°F for the containment upper compartment, and

b. ≥ 100°F and ≤ 120°F for the containment lower compartment.

--NOTES--

1. The minimum containment average air temperature in MODES 2, 3, and 4 may be reduced to 60°F.

2. Containment lower compartment temperature may be between 120°F and 125°F for up to 90 cumulative days per calendar year provided lower compartment temperature average over the previous 365 days is less than 120°F. Within this 90 cumulative day period, lower compartment temperature may be between 125°F and 135°F for 72 cumulative hours.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limits.	A.1 Restore containment average air temperature to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment upper compartment average air temperature is within limits.	24 hours
SR 3.6.5.2	Verify containment lower compartment average air temperature is within limits.	24 hours

Duke Power Company McGuire Nuclear Station **Nuclear Service Water System** **Continuous Use**	Procedure No. **OP/ 1/A/6400/006** Revision No. **154** Electronic Reference No. **MC004756**

PERFORMANCE

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(ISSUED) - PDF Format

Revision History (significant issues, limited to one page)

Rev 154 10/14/02: The following changes are included in this revision:

- In Enclosures 4.2 and 4.3 added sections to provide guidance to perform manual backwash with the discharge going to the sump instead of RC.
- Made various changes to Enclosure 4.14 per System Engineer to correct procedure deficiencies and ensure RN Pump Minimum flow requirements are met.
- In response to PIP 02-3754 added notes throughout the procedure prior to starting RN Pumps, "Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms".

Rev 153 (09/27/01) Per MM 12006, increased the D/P set point at which KC is declared inoperable. The caution prior to Step 3.2 in Enclosure 4.9 was changed from 11.5 psid to 13.5 psid.

Rev 152 (8/13/01) Enclosure 4.13 step 3.2 changed check off box to a RO sign off box.

Rev 151 (04/18/01)

- Encl. 4.2 and 4.3 changed for 5 minutes to at least 5 minutes and deleted reference to PIP 0-M96-2223 (Comp actions have been added to enclosure).
- Encl. 4.5, 4.12 and 4.14 added PP to place white tags on breakers and stickers on control boards.
- Encl. 4.12 added step to check SNSWP level.
- Encl. 4.10 deleted note referring to failing air to control valves and making RN inoperable. Valves are failed using hydraulic bypass and inoperability of RN is covered in associated section of enclosure.

Rev 150 (04/08/01) Incorporated outstanding changes

149A - Enclosure 4.14 deleted reference to Encl. 4.8 to start a pump
149B - replaced Enclosure 4.2 per MM-11224 (RN Backwash, A Train)
149C - replaced Enclosure 4.3 per MM-11224 (RN Backwash, B Train)

Rev 149 (8-8-00)
•New Enclosure 4.14 to direct aligning A Train to supply the Non Ess Header from the SNSWP.
•Enclosure 4.8
-Added step to evaluate effect of losing cooling water to affected Ess Header prior to securing train.
-Changed title to "Shifting Operational Alignment of RN Pumps".

Unit 1

Nuclear Service Water System

1. Purpose

To describe proper operation of the RN System.

2. Limits and Precautions

2.1 Automatic backwash of RN Strainer occurs at 1.86 psid (55 INWC). Exceeding automatic initiation set point will **NOT** cause RN System problems. {PIP 0-M96-2223}

2.2 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

2.3 RN suction alignments to RC Supply Crossover may introduce Asiatic Clams into the RN System. {Generic Letter 89-13}

2.4 NS HX drain down (voids) may occur when NS HX outlet valve opened. {PIP 1-M94-0538}

2.5 Only RN Train B should be aligned to SNSWP during normal operation due to problems with RN header air accumulation.

2.6 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

2.7 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2.8 Maximum SNSWP temperature is 82°F. (ITS 3.7.8)

3. Procedure

See Section 4.

Unit 1

4. Enclosures

4.1 Startup

4.2 Manual Backwash of RN Strainer 1A

4.3 Manual Backwash of RN Strainer 1B

4.4 Filling the SNSWP

4.5 Shifting Suction From Low Level Intake to the SNSWP and Realignment

4.6 Shifting Suction From Low Level Intake to RC Supply Crossover

4.7 Shifting Suction From RC Supply Crossover to Low Level Intake

4.8 Shifting Operational Alignment of RN Pumps

4.9 KC Heat Exchanger Flush and Realignment

4.10 VC Chiller Flush and Realignment

4.11 Cooling the SNSWP Using Low Level Intake

4.12 Defeating Automatic Alignment of RN Trains

4.13 Manual Automatic Backwash of RN Strainers 1A & 1B

4.14 Supplying the Non-Essential Header from the SNSWP

End of Body

Unit 1

1. Limits and Precautions

1.1 Automatic backwash of RN Strainer occurs at 1.86 psid (55 INWC). Exceeding automatic initiation set point will **NOT** cause RN System problems. {PIP 0-M96-2223}

1.2 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.3 RN suction alignments to RC Supply Crossover may introduce Asiatic clams into the RN System. {Generic Letter 89-13}

1.4 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.5 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 RN System suction and discharge to RC System is available.

_____ 2.2 RV aligned per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

_____ 2.3 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

☐ Section 3.3 Starting 1A RN Pump
☐ Section 3.4 Starting 1B RN Pump

3.3 Starting 1A RN Pump

_____ 3.3.1 Ensure 1A RN Pump Base Plate Drain Sump Pump switch in "ON".

☐ 3.3.2 Ensure lit "Train A Modulating Valve Reset".

_____ 3.3.3 **IF** required, vent 1A RN Pump using 1RN-248 (1A RN Pump Vent).

_____ DV _____ 3.3.4 **IF** 1A RN Pump is the first RN Pump to be started, unlock and throttle 3 turns open 1RN-29 (1A RN Pump Disch Isol).

_____ 3.3.5 Ensure 1RN-86A Mode Select in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.3.6 Start 1A RN Pump.

_____ 3.3.7 Ensure 1RN-86A (A KC HX Inlet Isol) fully opens.

_____ DV _____ 3.3.8 **IF** throttled per Step 3.3.4 continue to throttle open 1RN-29 (1A RN Pump Disch Isol).

_____ DV _____ 3.3.9 **WHEN** 1A RN Pump discharge header pressure reaches 100 psig, ensure fully open and locked 1RN-29 (1A RN Pump Disch Isol).

_____ 3.3.10 Control KC temperature as desired for KC Train A by throttling 1RN-89A (RN to A KC HX Control).

3.4 Starting 1B RN Pump

_____ 3.4.1 Ensure 1B RN Pump Base Plate Drain Sump Pump switch in "ON".

☐ 3.4.2 Ensure lit "Train B Modulating Valve Reset".

_____ 3.4.3 **IF** required, vent 1B RN Pump using 1RN-250 (1B RN Pump Vent).

_____ DV _____ 3.4.4 **IF** 1B RN Pump is the first RN Pump to be started, unlock and throttle 3 turns open 1RN-31 (1B RN Pump Disch Isol).

_____ 3.4.5 Ensure 1RN-187B Mode Select in "AUTO".

NOTE: Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.4.6 Start 1B RN Pump.

_____ 3.4.7 Ensure 1RN-187B (B KC HX Inlet Isol) fully opens.

_____ DV _____ 3.4.8 **IF** throttled per Step 3.4.4 continue to throttle open 1RN-31 (1B RN Pump Disch Isol).

_____ DV _____ 3.4.9 **WHEN** 1B RN Pump discharge header pressure reaches 100 psig, ensure fully open and locked 1RN-31 (1B RN Pump Disch Isol).

_____ 3.4.10 Control KC temperature as desired for KC Train B by throttling 1RN-190B (RN to B KC HX Control).

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 Automatic backwash of RN Strainer occurs at 1.86 psid (55 INWC). Exceeding automatic initiation set point will **NOT** cause RN System problems. {PIP 0-M96-2223}

2. Initial Conditions

_____ 2.1 Normal strainer backwash system non-functioning **AND** manual automatic strainer backwash unavailable.

_____ 2.2 1A RN Pump operating.

_____ 2.3 **IF** necessary to manually turn strainer drum, Maintenance standing by per MP/0/A/7150/032 (Nuclear Service Water Strainer Corrective Maintenance).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Complete one of the following:

_____ SRO 3.2.1 Declare RN Train A inoperable.

OR

_____ SRO 3.2.2 Ensure the following compensatory measures in place:

_____ SRO 3.2.2.1 Conduct a pre-job brief.

_____ SRO 3.2.2.2 Two designated operators stationed at 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).

_____ SRO 3.2.2.3 Designated operator in control room prepared to alert operators at 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol) to close valve upon receiving an ESF Actuation signal.

_____ SRO 3.2.2.4 Communication established between designated control room personnel and dedicated operators at 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).

_____ 3.3 Notify Control Room a manual RN Strainer Backwash in progress.

NOTE:	Manual Backwash of RN Strainer 1A with Strainer Discharge to RC is the preferred method.

3.4 Perform the following sections as applicable:

- ☐ Section 3.5, Manual Backwash of RN Strainer 1A with Strainer Discharge to RC
- ☐ Section 3.6, Manual Backwash of RN Strainer 1A with Strainer Discharge to WZ Sump A

3.5 Manual Backwash of RN Strainer 1A with Strainer Discharge to RC

☐ 3.5.1 Check open 1RN-23 (1A RN Strainer Backflush Manual Drain Isol).

3.5.2 Open 1RN-22A (1A RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.5.2.1 Close 1RNAS0222 (Air Supply Isol For 1RN-22A).
DV

_____ _____ 3.5.2.2 Open 1RNAS0221 (Air Supply Bypass For 1RN-22A).
DV

☐ 3.5.2.3 Check open 1RN-22A.

_____ _____ 3.5.3 Open 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).
DV

NOTE:	Per Engineering guidance, acceptable RN Strainer DP post backwash should be less than or equal to 30 INWC per 1RNPS5000 (1A RN Pump Strainer DP).

☐ 3.5.4 Allow strainer to backwash for at least 5 minutes.

_____ _____ 3.5.5 Close 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).
DV

NOTE:	Positioning the following valves vents air supply to 1RN-22A (1A RN Strainer Backflush Auto Drain Isol). Checking 1RN-22A closed assures safety related vent path will allow 1RN-22A to close on ESF Signal.

3.5.6 Close 1RN-22A (1A RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.5.6.1 Close 1RNAS0221 (Air Supply Bypass For 1RN-22A).
DV

_____ _____ 3.5.6.2 Open 1RNAS0222 (Air Supply Isol For 1RN-22A).
DV

☐ 3.5.6.3 Check closed 1RN-22A.

_____ 3.5.7 **IF** RN Train A was declared inoperable, evaluate RN Train A operability.
SRO

_____ 3.5.8 **IF** in place, evaluate removing compensatory measures.
SRO

CAUTION: There is the potential to flood WZ Sump A when performing Section 3.6.

3.6 Manual Backwash of RN Strainer 1A with Strainer Discharge to WZ Sump A

3.6.1 Close 1RN-23 (1A RN Strainer Backflush Manual Drain Isol) as follows:

_____ DV _____ 3.6.1.1 Place 1RNAS0023 (3-Way Air Control For 1RN-23) in vent position.

☐ 3.6.1.2 Check closed 1RN-23.

NOTE: A separate NLO should be stationed to monitor WZ Sump A level and manually operate WZ Sump A Pumps to decrease sump level if required.

_____ DV _____ 3.6.2 Throttle open 1RN-482 (RN Strainer 1A Backflush To Sump Isol).

_____ DV _____ 3.6.3 **IF** WZ Sump A Pumps are unable to maintain WZ Sump A level throttle closed 1RN-482 (RN Strainer 1A Backflush To Sump Isol).

3.6.4 Open 1RN-22A (1A RN Strainer Backflush Auto Drain Isol) as follows:

_____ DV _____ 3.6.4.1 Close 1RNAS0222 (Air Supply Isol For 1RN-22A).

_____ DV _____ 3.6.4.2 Open 1RNAS0221 (Air Supply Bypass For 1RN-22A).

☐ 3.6.4.3 Check open 1RN-22A.

_____ DV _____ 3.6.5 Open 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).

NOTE: Per Engineering guidance, acceptable RN Strainer DP post backwash should be less than or equal to 30 INWC per 1RNPS5000 (1A RN Pump Strainer DP).

☐ 3.6.6 Allow strainer to backwash for at least 5 minutes.

_____ DV _____ 3.6.7 Close 1RN-20 (RN Strainer 1A Backflush Manual Supply Isol).

NOTE: Positioning the following valves vents air supply to 1RN-22A (1A RN Strainer Backflush Auto Drain Isol). Checking 1RN-22A closed assures safety related vent path will allow 1RN-22A to close on ESF Signal.

3.6.8 Close 1RN-22A (1A RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.6.8.1 Close 1RNAS0221 (Air Supply Bypass For 1RN-22A).
DV

_____ _____ 3.6.8.2 Open 1RNAS0222 (Air Supply Isol For 1RN-22A).
DV

☐ 3.6.8.3 Check closed 1RN-22A.

_____ _____ 3.6.9 Close 1RN-482 (RN Strainer 1A Backflush To Sump Isol).
DV

3.6.10 Open 1RN-23 (1A RN Strainer Backflush Manual Drain Isol) as follows:

_____ _____ 3.6.10.1 Position 1RNAS0023 (3-Way Air Control For 1RN-23) to align air supply to 1RN-23.
DV

☐ 3.6.10.2 Check open 1RN-23.

_____ 3.6.11 **IF** WZ Sump A Pumps operated to maintain sump level, ensure in "AUTO".

_____ 3.6.12 **IF** RN Train A was declared inoperable, evaluate RN Train A operability.
SRO

_____ 3.6.13 **IF** in place, evaluate removing compensatory measures.
SRO

End of Enclosure

1. Limits and Precautions

1.1 Automatic backwash of RN Strainer occurs at 1.86 psid (55 INWC). Exceeding automatic initiation set point will **NOT** cause RN System problems. {PIP 0-M96-2223}

2. Initial Conditions

_____ 2.1 Normal strainer backwash system non-functioning **AND** manual automatic strainer backwash unavailable.

_____ 2.2 1B RN Pump operating.

_____ 2.3 **IF** necessary to manually turn strainer drum, Maintenance standing by per MP/0/A/7150/032 (Nuclear Service Water Strainer Corrective Maintenance).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Complete one of the following:

_____ (SRO) 3.2.1 Declare RN Train B inoperable.

OR

3.2.2 Ensure the following compensatory measures in place:

_____ (SRO) 3.2.2.1 Conduct a pre-job brief.

_____ (SRO) 3.2.2.2 Two designated operators stationed at 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).

_____ (SRO) 3.2.2.3 Designated operator in control room prepared to alert operators at 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol) to close valve upon receiving an ESF Actuation signal.

_____ (SRO) 3.2.2.4 Communication established between designated control room personnel and dedicated operators at 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).

_____ 3.3 Notify Control Room a manual RN strainer backwash in progress.

NOTE: Manual Backwash of RN Strainer 1B with Strainer Discharge to RC is the preferred method.

3.4 Perform the following sections as applicable:

- ☐ Section 3.5, Manual Backwash of RN Strainer 1B with Strainer Discharge to RC
- ☐ Section 3.6, Manual Backwash of RN Strainer 1B with Strainer Discharge to WZ Sump B

3.5 Manual Backwash of RN Strainer 1B with Strainer Discharge to RC

☐ 3.5.1 Check open 1RN-27 (1B RN Strainer Backflush Manual Drain Isol).

3.5.2 Open 1RN-26B (1B RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.5.2.1 Close 1RNAS0262 (Air Supply Isol For 1RN-26B).
DV

_____ _____ 3.5.2.2 Open 1RNAS0261 (Air Supply Bypass For 1RN-26B).
DV

☐ 3.5.2.3 Check open 1RN-26B.

_____ _____ 3.5.3 Open 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).
DV

NOTE: Per Engineering guidance, acceptable RN Strainer DP post backwash should be less than or equal to 30 INWC per 1RNPS5010 (1B RN Pump Strainer DP).

☐ 3.5.4 Allow strainer to backwash for at least 5 minutes.

_____ _____ 3.5.5 Close 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).
DV

NOTE: Positioning the following valves vents air supply to 1RN-26B (1B RN Strainer Backflush Auto Drain Isol). Checking 1RN-26B closed assures safety related vent path will allow 1RN-26B to close on ESF Signal.

3.5.6 Close 1RN-26B (1B RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.5.6.1 Close 1RNAS0261 (Air Supply Bypass For 1RN-26B).
DV

_____ _____ 3.5.6.2 Open 1RNAS0262 (Air Supply Isol For 1RN-26B).
DV

☐ 3.5.6.3 Check closed 1RN-26B.

_____ 3.5.7 **IF** RN Train B was declared inoperable, evaluate RN Train B operability.
SRO

_____ 3.5.8 **IF** in place, evaluate removing compensatory measures.
SRO

Unit 1

CAUTION: There is the potential to flood WZ Sump B when performing Section 3.6.

3.6 Manual Backwash of RN Strainer 1B with Strainer Discharge to WZ Sump B

3.6.1 Close 1RN-27 (1B RN Strainer Backflush Manual Drain Isol) as follows:

_____ _____ 3.6.1.1 Place 1RNAS0027 (3-Way Air Control For 1RN-27) in vent position.
DV

☐ 3.6.1.2 Check closed 1RN-27.

NOTE: A separate NLO should be stationed to monitor WZ Sump B level and manually operate WZ Sump B Pumps to decrease sump level if required.

_____ _____ 3.6.2 Throttle open 1RN-483 (RN Strainer 1B Backflush To Sump Isol).
DV

_____ _____ 3.6.3 <u>**IF**</u> WZ Sump B Pumps are unable to maintain WZ Sump B level throttle closed 1RN-483 (RN Strainer 1B Backflush To Sump Isol).
DV

3.6.4 Open 1RN-26B (1B RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.6.4.1 Close 1RNAS0262 (Air Supply Isol For 1RN-26B).
DV

_____ _____ 3.6.4.2 Open 1RNAS0261 (Air Supply Bypass For 1RN-26B).
DV

☐ 3.6.4.3 Check open 1RN-26B.

_____ _____ 3.6.5 Open 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).
DV

NOTE: Per Engineering guidance, acceptable RN Strainer DP post backwash should be less than or equal to 30 INWC per 1RNPS5010 (1B RN Pump Strainer DP).

☐ 3.6.6 Allow strainer to backwash for at least 5 minutes.

_____ _____ 3.6.7 Close 1RN-24 (RN Strainer 1B Backflush Manual Supply Isol).
DV

Unit 1

NOTE: Positioning the following valves vents air supply to 1RN-26B (1B RN Strainer Backflush Auto Drain Isol). Checking 1RN-26B closed assures safety related vent path will allow 1RN-26B to close on ESF Signal.

3.6.8 Close 1RN-26B (1B RN Strainer Backflush Auto Drain Isol) as follows:

_____ _____ 3.6.8.1 Close 1RNAS0261 (Air Supply Bypass For 1RN-26B).
DV

_____ _____ 3.6.8.2 Open 1RNAS0262 (Air Supply Isol For 1RN-26B).
DV

☐ 3.6.8.3 Check closed 1RN-26B.

_____ _____ 3.6.9 Close 1RN-483 (RN Strainer 1B Backflush To Sump Isol).
DV

3.6.10 Open 1RN-27 (1B RN Strainer Backflush Manual Drain Isol) as follows:

_____ _____ 3.6.10.1 Position 1RNAS0027 (3-Way Air Control For 1RN-27) to align air supply to 1RN-27.
DV

☐ 3.6.10.2 Check open 1RN-27.

_____ 3.6.11 **IF** WZ Sump B Pumps operated to maintain sump level, ensure in "AUTO".

_____ 3.6.12 **IF** RN Train B was declared inoperable, evaluate RN Train B operability.
SRO

_____ 3.6.13 **IF** in place, evaluate removing compensatory measures.
SRO

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 RN suction alignments to RC Supply Crossover to may introduce Asiatic clams into the RN System. {Generic Letter 89-13}

1.3 Only RN Train B should be aligned to SNSWP during normal operation due to problems with RN header air accumulation.

1.4 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.5 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

1.6 Maximum SNSWP temperature is 82°F. (ITS 3.7.8)

2. Initial Conditions

_____ 2.1 One or more of the following conditions:

- ☐ It is desired to fill SNSWP.
- ☐ Desired to align RN discharge to SNSWP to unwater as directed by RP.

_____ 2.2 RN in operation per Enclosure 4.1 (Startup).

_____ 2.3 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, Raising SNSWP level with RN Train A in service
- ☐ Section 3.4, Raising SNSWP level with RN Train B in service

Unit 1

3.3 Raising SNSWP level with RN Train A in service

☐ 3.3.1 Leave RN Train A in service to supply in-plant loads.

_____ 3.3.2 **IF** necessary, vent 1B RN Pump using 1RN-250 (1B RN Pump Vent).

_____ 3.3.3 Ensure "1RN-187B Mode Select" in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.3.4 Start 1B RN Pump.

_____ 3.3.5 Ensure 1RN-187B (B KC HX Inlet Isol) fully opens.

_____ 3.3.6 **WHEN** RN Train B operating, align the following:

_____ 3.3.6.1 Open 0RN-152B (Train 1B & 2B Disch to SNSWP).

_____ 3.3.6.2 Close 0RN-283AC (Train 1B & 2B Disch to RC).

_____ 3.3.6.3 Close 0RN-284B (Train 1B & 2B Disch to RC).

☐ 3.3.7 Allow SNSWP to fill to greater than 740 feet on 0RNP6000 (Standby NSW Pond Level).

_____ 3.3.8 **WHEN** desired level reached, stop 1B RN Pump as follows:

_____ 3.3.8.1 Place "1RN-187B Mode Select" to "MANUAL".

_____ 3.3.8.2 Depress "CLOSE" for 1RN-187B (B KC HX Inlet Isol).

_____ 3.3.8.3 Monitor RN to KC HX Flow and stop 1B RN Pump prior to 1RN-187B (B KC HX Inlet Isol) being fully closed.

_____ 3.3.8.4 Place "1RN-187B Mode Select" to "AUTO".

3.3.9 Realign RN Train B to normal discharge as follows:

_____ 3.3.9.1 Open 0RN-283AC (Train 1B & 2B Disch to RC).

_____ 3.3.9.2 Open 0RN-284B (Train 1B & 2B Disch to RC).

_____ 3.3.9.3 Close 0RN-152B (Train 1B & 2B Disch to SNSWP).

3.4 Raising SNSWP level with RN Train B in service

NOTE: SNSWP high temperature limit is 82°F (ITS 3.7.8).

3.4.1 Align the following:

_____ 3.4.1.1 Open 0RN-152B (Train 1B & 2B Disch to SNSWP).

_____ 3.4.1.2 Close 0RN-283AC (Train 1B & 2B Disch to RC).

_____ 3.4.1.3 Close 0RN-284B (Train 1B & 2B Disch to RC).

☐ 3.4.2 Allow SNSWP to fill to greater than 740 feet on 0RNP6000 (Standby NSW Pond Level).

_____ 3.4.3 **WHEN** desired level reached **OR** SNSWP temperature 78°F, realign RN Train B to normal discharge as follows:

_____ 3.4.3.1 Open 0RN-283AC (Train 1B & 2B Disch to RC).

_____ 3.4.3.2 Open 0RN-284B (Train 1B & 2B Disch to RC).

_____ 3.4.3.3 Close 0RN-152B (Train 1B & 2B Disch to SNSWP).

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 **<u>IF</u>** RN flow is greater than 20,000 gpm **<u>AND</u>** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 Only RN Train B should be aligned to SNSWP during normal operation due to problems with RN header air accumulation.

1.3 **<u>WHEN</u>** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.4 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 RN system in operation per Enclosure 4.1 (Startup).

_____ 2.2 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, Aligning RN Train B to the SNSWP
- ☐ Section 3.4, Aligning RN Train B to Low Level Intake
- ☐ Section 3.5, Aligning RN Train A to the SNSWP
- ☐ Section 3.6, Aligning RN Train A to Low Level Intake

Unit 1

3.3 Aligning RN Train B to the SNSWP

☐ 3.3.1 Check SNSWP level greater than 739.5 feet.

_____ 3.3.2 **IF** desired to isolate both non-essential headers, perform the following:

_____ 3.3.2.1 Ensure adequate RV Pumps to provide satisfactory Containment cooling.

3.3.2.2 Close:

- _____ 1RN-42A (AB Non ESS Supply Isol)
- _____ Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

3.3.3 Open:

- _____ 0RN-9B (Train 1B & 2B SNSWP Supply)
- _____ 0RN-152B (Train 1B & 2B Disch to SNSWP)

3.3.4 Close:

- _____ 0RN-283AC (Train 1B & 2B Disch to RC)
- _____ 0RN-284B (Train 1B & 2B Disch to RC)
- _____ 0RN-10AC (Train 1B & 2B LLI Supply)

3.4 Aligning RN Train B to Low Level Intake

3.4.1 Open:

- _____ 0RN-10AC (Train 1B & 2B LLI Supply)
- _____ 0RN-283AC (Train 1B & 2B Disch to RC)
- _____ 0RN-284B (Train 1B & 2B Disch to RC)

3.4.2 Close:

- _____ 0RN-152B (Train 1B & 2B Disch to SNSWP)
- _____ 0RN-9B (Train 1B & 2B SNSWP Supply)

3.4.3 Ensure open:

- _____ 1RN-42A (AB Non ESS Supply Isol)
- _____ Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

3.5 Aligning RN Train A to SNSWP

NOTE:
- Compensatory Actions are required when aligning RN Train A to SNSWP. Compensatory Actions are NOT required when performing RN Valve Stroke Timing and Slave Relay Testing. {PIR 0-M92-0074}
- Unit 1 and 2 TD CA Pumps are degraded but will remain operable provided respective Train B RN Essential supply to each TD CA Pump remains operable.

☐ 3.5.1 Prepare tagout per PP-364.

☐ 3.5.2 Check SNSWP level greater than 739.5 feet.

_____ SRO 3.5.3 Declare the following tracking entries in TSAIL: (RN Train A supplies will be isolated.)

☐ TD CA (Unit 1 A Train Assured Makeup)
☐ TD CA (Unit 2 A Train Assured Makeup)

3.5.4 Ensure closed:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.5.5 Place Control Board Switch Labels for the following:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.5.6 Tag open:

DV			Tag#
_____ DV	_____	• 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A)	________
_____ DV	_____	• Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A)	________

Unit 1

_____ 3.5.7 **IF** desired to isolate both non-essential headers, perform the following:

_____ 3.5.7.1 Ensure adequate RV Pumps to provide satisfactory Containment cooling.

3.5.7.2 Close:

_____ • 1RN-42A (AB Non ESS Supply Isol)
_____ • Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

3.5.8 Open:

_____ • 0RN-7A (Train 1A & 2A SNSWP Supply)
_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP)

3.5.9 Close:

_____ • 0RN-147 AC (Train 1A & 2A Disch to RC)
_____ • 0RN-12 AC (Train 1A & 2A LLI Supply)

3.6 Aligning RN Train A to Low Level Intake

3.6.1 Open:

_____ • 0RN-12 AC (Train 1A & 2A LLI Supply)
_____ • 0RN-147 AC (Train 1A & 2A Disch to RC)

3.6.2 Close:

_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP)
_____ • 0RN-7A (Train 1A & 2A SNSWP Supply)

3.6.3 Remove tag and close:

			Tag #
_____ DV	_____	• 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A)	_____________
_____ DV	_____	• Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A)	_____________

3.6.4 Remove Control Board Switch Labels from the following:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.6.5 Ensure closed:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

_____ 3.6.6 Evaluate TSAIL tracking entries for the following:
SRO

☐ TD CA (Unit 1 A Train Assured Makeup)
☐ TD CA (Unit 2 A Train Assured Makeup)

3.6.7 Ensure open:

_____ • 1RN-42A (AB Non ESS Supply Isol)
_____ • Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

End of Enclosure

1. Limits and Precautions

1.1 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 RN suction alignments to RC Supply Crossover may introduce Asiatic clams into the RN System. {Generic Letter 89-13}

1.3 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.4 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 RC System in operation per OP/1/B/6400/001 A (Condenser Circulating Water Systems).

_____ 2.2 RN System in operation per Enclosure 4.1 (Startup).

_____ 2.3 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

_____ 3.2 Ensure adequate RV Pumps are running and aligned to both Units Containment headers to ensure satisfactory containment cooling.

3.3 Perform the following sections as applicable:

☐ Section 3.4, Aligning RN Train A Suction to RC Supply Crossover
☐ Section 3.5, Aligning RN Train B Suction to RC Supply Crossover

Unit 1

3.4 Aligning RN Train A Suction to RC Supply Crossover

_____ 3.4.1 **<u>IF</u>** desired to isolate both non-essential headers, perform the following:

_____ 3.4.1.1 Ensure adequate RV Pumps to provide satisfactory Containment cooling.

3.4.1.2 Close:

- _____ 1RN-42A (AB Non ESS Supply Isol)
- _____ Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

CAUTION: Take care during the following operation to ensure pump suction is **<u>NOT</u>** lost.

3.4.2 Open:

- _____ 0RN-2B (Train 1A & 2A RC Supply)
- _____ 0RN-3A (Train 1A & 2A RC Supply)

3.4.3 Close:

- _____ 0RN-12AC (Train 1A & 2A LLI Supply)
- _____ 0RN-13A (Train 1A & 2A LLI Supply)

3.5 Aligning RN Train B Suction to RC Supply Crossover

_____ 3.5.1 **IF** desired to isolate both non-essential headers, perform the following:

_____ 3.5.1.1 Ensure adequate RV Pumps to provide satisfactory Containment cooling.

3.5.1.2 Close:

- _____ 1RN-42A (AB Non ESS Supply Isol)
- _____ Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

CAUTION: Take care during the following operation to ensure pump suction is **NOT** lost.

3.5.2 Ensure open:

- _____ 0RN-4AC (Train 1B & 2B RC Supply)
- _____ 0RN-5B (Train 1B & 2B RC Supply)

3.5.3 Close:

- _____ 0RN-10AC (Train 1B & 2B LLI Supply)
- _____ 0RN-11B (Train 1B & 2B LLI Supply)

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 **<u>IF</u>** RN flow is greater than 20,000 gpm **<u>AND</u>** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 RN suction alignments to RC Supply Crossover may introduce Asiatic clams into the RN System. {Generic Letter 89-13}

1.3 **<u>WHEN</u>** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.4 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 RC System in operation per OP/1/B/6400/001 A (Condenser Circulating Water Systems).

_____ 2.2 RN System suction from RC Supply Crossover.

_____ 2.3 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

_____ DV _____ 3.2 Ensure open 1RN-1 (LLI Supply to RN), located at Dam.

3.3 Perform the following sections as applicable:

- ☐ Section 3.4, Aligning RN Train A Suction to Low Level Intake
- ☐ Section 3.5, Aligning RN Train B Suction to Low Level Intake

Unit 1

CAUTION: Take care during the following operation to ensure pump suction is **NOT** lost.

3.4 Aligning RN Train A Suction to Low Level Intake

3.4.1 Open:

_____ • 0RN-12AC (Train 1A & 2A LLI Supply)
_____ • 0RN-13A (Train 1A & 2A LLI Supply)

3.4.2 Close:

_____ • 0RN-2B (Train 1A & 2A RC Supply)
_____ • 0RN-3A (Train 1A & 2A RC Supply)

3.4.3 Ensure open:

_____ • 1RN-42A (AB Non ESS Supply Isol)
_____ • Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

CAUTION: Take care during the following operation to ensure pump suction is **NOT** lost.

3.5 Aligning RN Train B Suction to Low Level Intake

3.5.1 Open:

_____ • 0RN-10AC (Train 1B & 2B LLI Supply)
_____ • 0RN-11B (Train 1B & 2B LLI Supply)

_____ 3.5.2 Close 0RN-5B (Train 1B & 2B RC Supply)

3.5.3 Ensure open:

_____ • 1RN-42A (AB Non ESS Supply Isol)
_____ • Unit 2 valve 2RN-42A (AB Non ESS Supply Isol)

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.3 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 Essential Trains of RN are tied together with discharge pressure on both Trains.

_____ 2.2 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

_____ 3.2 **IF** KC HX RN corrected DP is greater than 7.5 on in service HX, evaluate performing High Velocity Flush prior to shifting trains.

☐ 3.3 Monitor RV Pumps and containment temperatures while shifting operational alignment of RN Pumps.

3.4 Perform the following sections as applicable:

- ☐ Section 3.5, Starting 1A RN Pump
- ☐ Section 3.6, Stopping 1A RN Pump
- ☐ Section 3.7, Starting 1B RN Pump
- ☐ Section 3.8, Stopping 1B RN Pump

Unit 1

3.5 Starting 1A RN Pump

_____ 3.5.1 **IF** necessary, vent 1A RN Pump using 1RN-248 (1A RN Pump Vent).

_____ 3.5.2 Ensure "1RN-86A Mode Select" in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.5.3 Start 1A RN Pump.

_____ 3.5.4 Ensure full open 1RN-86A (A KC HX Inlet Isol).

_____ 3.5.5 Throttle 1RN-89A (RN to A KC HX Control) to control KC temperature.

3.6 Stopping 1A RN Pump

_____ 3.6.1 Evaluate effect of loss of cooling on A Train Essential Header.

_____ 3.6.2 Place "1RN-86A Mode Select" in "MANUAL".

_____ 3.6.3 Depress close for 1RN-86A (A KC HX Inlet Isol).

_____ 3.6.4 Monitor RN to KC HX Flow and stop 1A RN Pump prior to 1RN-86A (A KC HX Inlet Isol) being fully closed.

_____ 3.6.5 Place "1RN-86A Mode Select" in "AUTO".

3.7 Starting 1B RN Pump

_____ 3.7.1 **IF** necessary, vent 1B RN Pump using 1RN-250 (1B RN Pump Vent).

_____ 3.7.2 Ensure "1RN-187B Mode Select" in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.7.3 Start 1B RN Pump.

_____ 3.7.4 Ensure full open 1RN-187B (B KC HX Inlet Isol).

_____ 3.7.5 Throttle 1RN-190B (RN to B KC HX Control) to control KC temperature.

3.8 Stopping 1B RN Pump

_____ 3.8.1 Evaluate effect of loss of cooling on B Train Essential Header.

_____ 3.8.2 Place "1RN-187B Mode Select" in "MANUAL".

_____ 3.8.3 Depress close for 1RN-187B (B KC HX Inlet Isol).

_____ 3.8.4 Monitor RN to KC HX Flow and stop 1B RN Pump prior to HX Inlet valve being fully closed.

_____ 3.8.5 Place "1RN-187B Mode Select" in "AUTO".

End of Enclosure

1. Limits and Precautions

1.1 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 **WHEN** Unit operating conditions allow, normal RN flow rate through KC HX should be limited to 2000 - 2500 gpm. Operation at this desired flow range ensures HX outlet valves and associated piping experience minimal flow induced vibration. {PIP-99-0157}

1.3 Minimum flow protection for RN Pump provided by RN flow to KC HX. With total train flow rate less than set point, approximately 3000 gpm, the affected control loop will override manual inputs to ensure adequate pump protection. {PIP-99-0157}

2. Initial Conditions

_____ 2.1 Essential Trains of RN are tied together with discharge pressure on both Trains.

_____ 2.2 **IF** performing a High Velocity Flush, KC HX RN corrected DP is greater than 7.5 psid.

_____ 2.3 **IF** performing a Super Flush, KC HX RN corrected DP is greater than 9.0 psid.

3. Procedure

□ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

CAUTION: **IF** either KC Train HX corrected DP increases to 13.5 psid, the associated KC Train must be declared inoperable. Accurate DP readings can only be obtained with RN flow rate to KC HX greater than or equal to 2500 gpm.

3.2 Perform the applicable section:

□ Section 3.3, 1A KC HX High Velocity Flush
□ Section 3.4, 1A KC HX Super Flush
□ Section 3.5, 1B KC HX High Velocity Flush
□ Section 3.6, 1B KC HX Super Flush

3.3 1A KC HX High Velocity Flush

_____ 3.3.1 **<u>IF</u>** necessary, vent 1A RN Pump using 1RN-248 (1A RN Pump Vent).

_____ 3.3.2 Ensure "1RN-86A Mode Select" in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.3.3 Ensure 1A RN Pump in operation.

_____ 3.3.4 Ensure full open 1RN-86A (A KC HX Inlet Isol).

NOTE:	• RN to KC HX Flow must be greater than 2500 gpm to validate KC HX DP readings. • A High Velocity Flush of KC HX is performed by opening KC HX control valve to maximum position allowed by travel stops. A typical High Velocity Flush results in 7000 - 8000 gpm flow through KC HX.

☐ 3.3.5 Monitor RN and RV Pumps suction pressures while performing High Velocity Flush.

_____ 3.3.6 Throttle 1RN-89A (RN to A KC HX Control) to 6000 gpm.

_____ 3.3.7 **<u>IF</u>** additional flow is required for flush, evaluate the following:
SRO

☐ Placing 1B RN Pump in service per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps)
☐ Starting one or more RV Pumps

_____ 3.3.8 Record performance of High Velocity Flush and initial DP reading in RO Logbook. **<u>IF</u>** available, use OAC point M1P1222 (KC HX A RN Corrected DP) (15 min Avg).

_____ 3.3.9 Open 1RN-89A (RN to A KC HX Control), from Control Room, to maximum position.

NOTE:	Best results are achieved by flushing KC HX for 25 - 35 minutes.

☐ 3.3.10 Flush 1A KC HX for at least 15 minutes but **<u>NOT</u>** greater than 2 hours.

_____ 3.3.11 Throttle 1RN-89A (RN to A KC HX Control) to 6000 gpm for final DP readings.

_____ 3.3.12 Record final DP reading in RO Logbook. **IF** available, use computer point M1P1222 (KC HX A RN Corrected DP) (15 min Avg).

_____ 3.3.13 Throttle 1RN-89A (RN to A KC HX Control) as desired to control KC temperature.

_____ 3.3.14 **IF** desired, secure one of the operating Unit 1 RN Trains per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.3.15 **IF** additional RV Pumps were started, return RV to desired alignment.

3.4 1A KC HX Super Flush

NOTE:
- To perform a Super Flush of KC HX, KC HX control valve travel stops are moved to allow for maximum flow possible which will cause affected train to be inoperable.
- RN to KC HX Flow must be greater than 2500 gpm to validate KC HX DP readings. (Readings approaching 10,000 gpm are optimal for Super Flush.)

☐ 3.4.1 Monitor RN and RV Pumps suction pressures while performing Super Flush.

_____ 3.4.2 Ensure 1A and 1B RN Pumps in service per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.4.3 Throttle 1RN-89A (RN to A KC HX Control) to 6000 gpm.

_____ 3.4.4 Record performance of Super Flush and initial DP reading in RO Logbook. **IF** available, use computer point M1P1222 (KC HX A RN Corrected DP) (15 min Avg).

_____ 3.4.5 Declare 1A RN inoperable.
SRO

NOTE: 1RN-86A (A KC HX Inlet Isol) can be manually opened past its electrical throttle position.

_____ 3.4.6 Locally open 1RN-86A (A KC HX Inlet Isol) to full open position.

NOTE:
- Maximum Super Flush duration is 1 hour.
- RN Train separation is **NOT** required. Leaving RN Trains cross-tied will maximize header pressure and reduce potential for YC Chiller trips. {PIP 99-4170}

_____ 3.4.7 Throttle 1RN-89A (RN to A KC HX Control) as required to allow IAE to adjust travel stops to full open position, per IP/0/A/3219/016 (Setting Throttle Valve Stops).

Unit 1

_____ 3.4.8 Slowly open 1RN-89A (RN to A KC HX Control) to its maximum position (12,000 - 14,000 gpm as indicated on 1RNP5040 Pump Flow)), or OAC Point M1A1395 (1A KC HX RN Wide Range Flow), greater than 10,000 gpm.

☐ 3.4.9 Flush 1A KC HX for 25 - 35 minutes.

_____ 3.4.10 Ensure 1RN-89A (RN to A KC HX Control) has remained full open.

_____ 3.4.11 Throttle 1RN-89A (RN to A KC HX Control) as required to have IAE reposition and lock travel stops for 1RN-89A (RN To A KC HX Control) per IP/0/A/3219/016 (Setting Throttle Valve Stops).

_____ 3.4.12 Throttle 1RN-89A (RN to A KC HX Control) to 6000 gpm for final DP readings.

_____ 3.4.13 Record final DP reading in RO Logbook. **IF** available, use OAC point M1P1222 (KC HX A RN Corrected DP) (15 min Avg).

_____ 3.4.14 Place "1RN-86A Mode Select" in "MANUAL" and stroke from open to closed and back to open to check proper operation.

_____ 3.4.15 Place "1RN-86A Mode Select" in "AUTO".

_____ SRO 3.4.16 **WHEN** 1A KC HX Super Flush complete **AND** travel stops for 1RN-89A (RN to A KC HX Control) have been repositioned, evaluate 1A RN operability.

_____ 3.4.17 Throttle 1RN-89A (RN to A KC HX Control) as desired to control KC temperature.

_____ 3.4.18 **IF** desired, secure one of the operating Unit 1 RN Trains per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.4.19 **IF** additional RV Pumps were started, return RV to desired alignment.

3.5 1B KC HX High Velocity Flush

_____ 3.5.1 **IF** necessary, vent 1B RN Pump using 1RN-250 (1B RN Pump Vent).

_____ 3.5.2 Ensure 1RN-187B (B KC HX Inlet Isol) in "AUTO".

NOTE: Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.5.3 Ensure 1B RN Pump in operation.

_____ 3.5.4 Ensure 1RN-187B (B KC HX Inlet Isol) fully opens.

NOTE:	
	• RN to KC HX Flow must be greater than 2500 gpm to validate KC HX DP readings. • A High Velocity Flush of KC HX is performed by opening KC HX control valve to maximum position allowed by travel stops. A typical High Velocity Flush results in 7000 - 8000 gpm flow through KC HX.

☐ 3.5.5 Monitor RN and RV Pumps suction pressures while performing High Velocity Flush.

_____ 3.5.6 Throttle 1RN-190B (RN to B KC HX Control) to 6000 gpm.

_____ 3.5.7 **IF** additional flow is required for flush, evaluate the following:
SRO

☐ Placing 1A RN Pump in service per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps)
☐ Starting one or more RV Pumps

_____ 3.5.8 Record performance of High Velocity Flush and initial DP reading in RO Logbook. **IF** available, use OAC Point M1P1223 (KC HX B RN Corrected DP) (15 min Avg).

_____ 3.5.9 Open 1RN-190B (RN to B KC HX Control), from Control Room, to maximum position.

NOTE:	Best results are achieved by flushing KC HX for 25 - 35 minutes.

☐ 3.5.10 Flush 1B KC HX for at least 15 minutes but **NOT** greater than 2 hours.

_____ 3.5.11 Throttle 1RN-190B (RN to B KC HX Control) to 6000 gpm for final DP readings.

_____ 3.5.12 Record final DP reading in RO Logbook. **IF** available, use OAC Point M1P1223 (KC HX B Corrected DP) (15 min Avg).

_____ 3.5.13 Throttle 1RN-190B (RN to B KC HX Control) as desired to control KC temperature.

_____ 3.5.14 **IF** desired, secure one of the operating Unit 1 RN Trains per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.5.15 **IF** additional RV Pumps were started, return RV to desired alignment.

3.6 1B KC HX Super Flush

NOTE:	• To perform a Super Flush of KC HX, KC HX control valve travel stops are moved to allow for maximum flow possible which will cause affected train to be inoperable. • RN to KC HX Flow must be greater than 2500 gpm to validate KC HX DP readings. (Readings approaching 10,000 gpm are optimal for Super Flush.)

☐ 3.6.1 Monitor RN and RV Pumps suction pressures while performing Super Flush.

_____ 3.6.2 Ensure 1A and 1B RN Pumps in service per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.6.3 Throttle 1RN-190B (RN to B KC HX Control) to 6000 gpm.

_____ 3.6.4 Record performance of Super Flush and initial DP reading in RO Logbook. **<u>IF</u>** available, use OAC Point M1P1223 (KC HX B RN corrected DP) (15 min Avg).

_____ 3.6.5 Declare 1B RN inoperable.
SRO

NOTE:	1RN-187B (B KC HX Inlet Isol) can be manually opened past its electrical throttle position.

_____ 3.6.6 Locally open 1RN-187B (B KC HX Inlet Isol) to full open position.

NOTE:	• Maximum Super Flush duration is 1 hour. • RN Train separation is **<u>NOT</u>** required. Leaving RN Trains cross-tied will maximize header pressure and reduce potential for YC Chiller trips. {PIP 99-4170}

_____ 3.6.7 Throttle 1RN-190B (RN to B KC HX Control) as required to allow IAE to adjust travel stops to full open position per IP/0/A/3219/016 (Setting Throttle Valve Stops).

_____ 3.6.8 Slowly open 1RN-190B (RN to B KC HX Control) to maximum position (12,000 - 14,000 gpm as indicated on 1RNP-5050 (1B RN Pump Flow) or OAC Point M1A1401 (B KC HX B Wide Range Flow), greater than 10,000 gpm.

☐ 3.6.9 Flush 1B KC HX for 25 - 35 minutes.

_____ 3.6.10 Ensure 1RN-190B (RN to B KC HX Control) has remained full open.

_____ 3.6.11 Throttle 1RN-190B (RN to B KC HX Control) as required to have IAE reposition and lock travel stops for 1RN-190B (RN to B KC HX Control) per IP/0/A/3219/016 (Setting Throttle Valve Stops).

_____ 3.6.12 Throttle 1RN-190B (RN to B KC HX Control) to 6000 gpm for final DP readings.

_____ 3.6.13 Record final DP reading in RO Logbook. **IF** available, use OAC Point M1P1223 (KC HX B RN Corrected DP) (15 min Avg).

_____ 3.6.14 Place "1RN-187B Mode Select" in "MANUAL" **AND** stroke from open to closed and back to open to check proper operation.

_____ 3.6.15 Place "1RN-187B Mode Select" in "AUTO".

_____ SRO 3.6.16 **WHEN** 1B KC HX Super Flush complete **AND** travel stop for 1RN-190B (RN to B KC HX Control) has been repositioned, evaluate 1B RN operability.

_____ 3.6.17 Throttle 1RN-190B (RN to B KC HX Control) as desired to control KC temperature.

_____ 3.6.18 **IF** desired, secure one of the operating Unit 1 RN Trains per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.6.19 **IF** additional RV Pumps were started, return RV to desired alignment.

End of Enclosure

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 VC Chiller to be flushed has RN supplied from Unit 1.

_____ 2.2 HVAC crew has requested High Velocity Flush **OR** Super Flush of A or B VC Chiller.

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

☐ Section 3.3, A VC Chiller High Velocity Flush
☐ Section 3.4, A VC Chiller Super Flush
☐ Section 3.5, B VC Chiller High Velocity Flush
☐ Section 3.6, B VC Chiller Super Flush

3.3 A VC Chiller High Velocity Flush

_____ 3.3.1 **IF** chiller to be flushed in service, ensure HVAC crew available in case of chiller trip.

☐ 3.3.2 Start idle Train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.3.3 **WHEN** both RN Trains operating, close 1RN-40A (Train A to Non-Ess Hdr Isol)

_____ DV _____ HVAC 3.3.4 Fail open 1RN-445 (Control Rm A/C Cond A Contr) (located 750+9, GG54-55, above KC Pump Motor) using "Hydraulic Control Bypass" located on valve actuator.

_____ 3.3.5 **WHEN** flush complete (approximately 30 minutes or time specified by WCC SRO), perform the following:

_____ DV _____ HVAC 3.3.5.1 Return 1RN-445 (Control Rm A/C Cond A Contr) to normal using "Hydraulic Control Bypass" located on valve actuator.

_____ 3.3.5.2 Open 1RN-40A (Train A to Non-Ess Hdr Isol).

_____ 3.3.5.3 **IF** desired, shutdown one train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

Unit 1

3.4 A VC Chiller Super Flush

_____ 3.4.1 **IF** chiller to be flushed in service, ensure HVAC crew available in case of chiller trip.

_____ 3.4.2 Have Operations Test Group install RN to VC Flow Test Gauge (0 - 10 psid) for VC Chiller A, at 0RNPG-6110 (767, HH54).

☐ 3.4.3 Start idle Train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.4.4 **WHEN** both Trains of RN operating, close 1RN-40A (Train A to Non-Ess Hdr Isol).

_____ DV _____ HVAC 3.4.5 Fail open 1RN-445 (Control Rm A/C Cond A Contr) (located 750+9, GG54-55, above KC Pump Motor) using "Hydraulic Control Bypass" located on valve actuator.

_____ SRO 3.4.6 Declare RN Train A inoperable.

_____ DV _____ 3.4.7 Remove lock and throttle open 1RN-97 (VC A Control Bypass) (located 1' West of GG54, 750+10) to achieve 8.2 - 10 psid as read on Test Gauge at HH54, 767 near 0RNPG-6110.

_____ 3.4.8 **WHEN** flush complete (approximately 30 minutes or time specified by WCC SRO), perform the following:

- _____ DV _____ 3.4.8.1 Lock closed 1RN-97 (VC A Control Bypass).
- _____ SRO 3.4.8.2 Evaluate RN Train A operability.
- _____ DV _____ HVAC 3.4.8.3 Return 1RN-445 (Control Rm A/C Cond A Contr) to normal using "Hydraulic Control Bypass" located on valve actuator.
- _____ 3.4.8.4 Open 1RN-40A (Train A to Non-Ess Hdr Isol).
- _____ 3.4.8.5 **IF** desired, shutdown one train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).
- _____ 3.4.8.6 Have Operations Test Group remove RN to VC flow Test Gauge at 0RNPG-6110 for VC Chiller A.
- ☐ 3.4.8.7 Check 0RNPG-6110 reads plant conditions.

3.5 B VC Chiller High Velocity Flush

_____ 3.5.1 **IF** chiller to be flushed in service, ensure HVAC crew available in case of chiller trip.

☐ 3.5.2 Start idle Train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.5.3 **WHEN** both RN Trains operating, close 1RN-41B (Train B to Non-Ess Hdr Isol).

_____ DV _____ HVAC 3.5.4 Fail open 1RN-460 (Control Rm A/C Cond B Contr) (located 750+11, GG54, above 2A1 KC Pump) using "Hydraulic Control Bypass" located on valve actuator.

_____ 3.5.5 **WHEN** flush complete (approximately 30 minutes **OR** time specified by WCC SRO), perform the following:

_____ DV _____ HVAC 3.5.5.1 Return 1RN-460 (Control Rm A/C Cond B Contr) to normal using "Hydraulic Control Bypass" located on valve actuator.

_____ 3.5.5.2 Open 1RN-41B (Train B to Non-Ess Hdr Isol).

_____ 3.5.5.3 **IF** desired, shutdown one train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

3.6 B VC Chiller Super Flush

_____ 3.6.1 **IF** chiller to be flushed in service, ensure HVAC crew available in case of chiller trip.

_____ 3.6.2 Have Operations Test Group install RN to VC Flow Test Gauge (0 - 10 psid) for VC Chiller B, at 0RNPG6120 (767, GG55).

☐ 3.6.3 Start idle Train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.6.4 **WHEN** both Trains of RN operating, close 1RN-41B (Train B to Non-Ess Hdr Isol).

_____ DV _____ HVAC 3.6.5 Fail open 1RN-460 (Control Rm A/C Cond B Contr) (located 750+11, GG54, above 2A1 KC Pump) using "Hydraulic Control Bypass" located on valve actuator.

_____ SRO 3.6.6 Declare RN Train B inoperable.

Unit 1

_____ _____ 3.6.7 Remove lock and throttle open 1RN-198 (Control Rm A/C Cond B Cntrl Bypass) to achieve 8.2 - 10.0 psid as read on Test Gauge at 767, GG55, near 0RNPG6120.
DV

_____ 3.6.8 **WHEN** flush complete (approximately 30 minutes **OR** time specified by WCC SRO), perform the following:

_____ _____ 3.6.8.1 Lock closed 1RN-198 (Control Rm A/C Cond B Cntrl Bypass).
DV

_____ 3.6.8.2 Evaluate RN Train B operability.
SRO

_____ _____ 3.6.8.3 Return 1RN-460 (Control Rm A/C Cond B Contr) to normal using "Hydraulic Control Bypass" located on valve actuator.
DV HVAC

_____ 3.6.8.4 Open 1RN-41B (Train B to Non-Ess Hdr Isol).

_____ 3.6.8.5 **IF** desired, shutdown one train of RN per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

_____ 3.6.8.6 Have Operations Test Group remove RN to VC flow Test Gauge at 0RNPG6120 for VC Chiller B.

☐ 3.6.8.7 Check 0RNPG6120 reads plant conditions.

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

1.2 Maximum SNSWP temperature is 82°F. (ITS 3.7.8)

2. Initial Conditions

_____ 2.1 RN System aligned to Low Level Intake.

_____ 2.2 Evaluation made for any adverse affects on operating RN Trains (both units).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following steps to cool SNSWP.

_____ 3.2.1 **IF** RN Train A operating:

_____ 3.2.1.1 Initiate cooling by opening 0RN-9B (Train 1B & 2B SNSWP Supply).

☐ 3.2.1.2 Allow SNSWP to overflow until desired temperature is achieved.

_____ 3.2.1.3 **WHEN** desired temperature reached, close 0RN-9B (Train 1B & 2B SNSWP Supply).

_____ 3.2.2 **IF** RN Train B operating:

_____ 3.2.2.1 Initiate cooling by opening 0RN-7A (Train 1A & 2A SNSWP Supply).

☐ 3.2.2.2 Allow SNSWP to overflow until desired temperature achieved.

_____ 3.2.2.3 **WHEN** desired temperature reached, close 0RN-7A (Train 1A & 2A SNSWP Supply).

Unit 1

CAUTION:	WHEN aligning both Trains of RN to SNSWP for cooling, operating RN Pump and RV Pump suction pressures must be closely monitored to ensure adequate NPSH.

_____ 3.2.3 IF SNSWP is NOT cooling quickly enough, align both Trains of RN to SNSWP by opening the following:

- _____ • 0RN-7A (Train 1A & 2A SNSWP Supply)
- _____ • 0RN-9B (Train 1B & 2B SNSWP Supply)

_____ 3.2.4 WHEN desired temperature is reached, close:

- _____ • 0RN-7A (Train 1A & 2A SNSWP Supply)
- _____ • 0RN-9B (Train 1B & 2B SNSWP Supply)

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 Only RN Train B should be aligned to SNSWP during normal operation due to problems with RN header air accumulation.

1.2 **IF** RN flow is greater than 20,000 gpm **AND** suction is from Low Level Intake, RV Pumps may lose suction.

2. Initial Conditions

None

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, Defeating RN Train A Automatic Alignment Capability
- ☐ Section 3.4, Restoring RN Train A Automatic Alignment Capability
- ☐ Section 3.5, Defeating RN Train B Automatic Alignment Capability
- ☐ Section 3.6, Restoring RN Train B Automatic Alignment Capability

3.3 Defeating RN Train A Automatic Alignment Capability

☐ 3.3.1 Prepare tagout per PP-403.

☐ 3.3.2 Check SNSWP level greater than 739.5 feet.

NOTE:	Unit 1 and 2 TD CA Pumps are degraded but will remain operable provided respective Train B RN Essential supply to each TD CA Pump remains operable.

_____ SRO 3.3.3 Declare the following tracking entries in TSAIL: (RN Train A supplies will be isolated.)

- ☐ TD CA (Unit 1 A Train Assured Makeup)
- ☐ TD CA (Unit 2 A Train Assured Makeup)

Unit 1

3.3.4 Ensure closed:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.3.5 Place Control Board Switch Labels for the following:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)
_____ • 0RN-7A (Train 1A &2A SNSWP Supply) (U1 Control Board)
_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP) (U1 Control Board)
_____ • 0RN-7A (Train 1A &2A SNSWP Supply) (U2 Control Board)
_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP) (U2 Control Board)

3.3.6 Tag open:

			Tag #
_____ DV	_____	• 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A)	_____
_____ DV	_____	• Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A)	_____

_____ 3.3.7 Open 0RN-7A (Train 1A &2A SNSWP Supply).

_____ 3.3.8 Open 0RN-149A (Train 1A & 2A Disch to SNSWP).

3.3.9 Tag open:

			Tag #
_____ DV	_____	• 1EMXH-3C (SNSWP Supply A Shutoff Vlv 0RN-7A)	_____
_____ DV	_____	• 1EMXH-6A (SNSWP Discharge A Isolation Vlv 0RN-149A)	_____

3.3.10 Isolate RN Train A Discharge to RC per one of the following:

_____ 3.3.10.1 Close 0RN-147AC (Train 1A & 2A Disch to RC).

- _____ Place Control Board Switch Label on 0RN-147AC (Train 1A & 2A Disch to RC) (U1 Control Board)
- _____ Place Control Board Switch Label on 0RN-147AC (Train 1A & 2A Disch to RC) (U2 Control Board)
- _____ Place Control Board Switch Label on 0RN-147AC (Train 1A & 2A Disch to RC) (SSF)
- _____ (DV) _____ Tag open 1EMXH1-3B (CCW Discharge A Isolation Vlv 0RN-147AC) Tag # __________

OR

_____ 3.3.10.2 Close 0RN-148A (Train 1A & 2A Disch to RC).

- _____ Place Control Board Switch Label on 0RN-148A (Train 1A & 2A Disch to RC) (U1 Control Board)
- _____ Place Control Board Switch Label on 0RN-148A (Train 1A & 2A Disch to RC) (U2 Control Board)
- _____ (DV) _____ Tag open 1EMXH1-1E (CCW Discharge A Isolation Vlv 0RN-148A) Tag # __________

3.3.11 Isolate RN Train A suction from LLI per one of the following:

_____ 3.3.11.1 Close 0RN-12AC (Train 1A & 2A LLI Supply).

_____ • Place Control Board Switch Label on 0RN-12AC (Train 1A & 2A LLI Supply) (U1 Control Board)

_____ • Place Control Board Switch Label on 0RN-12AC (Train 1A & 2A LLI Supply) (U2 Control Board)

_____ • Place Control Board Switch Label on 0RN-12AC (Train 1A & 2A LLI Supply) (SSF)

_____ (DV) _____ • Tag open 1EMXH1-2B (Low Level Supply A Shutoff Vlv 0RN-12AC) Tag # ___________

OR

_____ 3.3.11.2 Close 0RN-13A (Train 1A & 2A LLI Supply).

_____ • Place Control Board Switch Label on 0RN-13A (Train 1A & 2A LLI Supply) (U1 Control Board)

_____ • Place Control Board Switch Label on 0RN-13A (Train 1A & 2A LLI Supply) (U2 Control Board)

_____ (DV) _____ • Tag open 1EMXH-4C (Low Level Supply A Shutoff Vlv 0RN-13A) Tag # ___________

3.4 Restoring RN Train A Automatic Alignment Capability

3.4.1 Remove tag and close:

			Tag #
_____ (DV)	_____	• 1EMXH-3C (SNSWP Supply A Shutoff Vlv 0RN-7A)	___________
_____ (DV)	_____	• 1EMXH-6A (SNSWP Discharge A Isol Vlv 0RN-149A)	___________

_____ 3.4.2 Remove tag from the following:

		Tag #
_____	• 1EMXH-4C (Low Level Supply A Shutoff Vlv 0RN-13A)	___________
	OR	
_____	• 1EMXH1-2B (Low Level Supply A Shutoff 0RN-12AC)	___________

Unit 1

3.4.3 Remove tag from the following:

Tag #

- _____ • 1EMXH1-1E (CCW Discharge A Isolation Vlv 0RN-148A) ________

OR

- _____ • 1EMXH1-3B (CCW Discharge A Isolation Vlv 0RN-147AC) ________

3.4.4 Ensure closed:

- _____ DV _____ • 1EMXH-4C (Low Level Supply A Shutoff Vlv 0RN-13A)
- _____ DV _____ • 1EMXH1-2B (Low Level Supply A Shutoff Vlv 0RN-12AC)
- _____ DV _____ • 1EMXH1-1E (CCW Discharge A Isolation Vlv 0RN-148A)
- _____ DV _____ • 1EMXH1-3B (CCW Discharge A Isolation Vlv 0RN-147AC)

3.4.5 Ensure Control Board Switch Labels removed from the following:

- _____ • 0RN-7A (Train 1A &2A SNSWP Supply) (U1 Control Board)
- _____ • 0RN-149A (Train 1A & 2A Disch to SNSWP) (U1 Control Board)
- _____ • 0RN-13A (Train 1A & 2A LLI Supply) (U1 Control Board)
- _____ • 0RN-148A (Train 1A & 2A Disch to RC) (U1 Control Board)
- _____ • 0RN-12AC (Train 1A & 2A LLI Supply) (U1 Control Board)
- _____ • 0RN-147AC (Train 1A & 2A Disch to RC) (U1 Control Board)
- _____ • 0RN-7A (Train 1A &2A SNSWP Supply) (U2 Control Board)
- _____ • 0RN-149A (Train 1A & 2A Disch to SNSWP) (U2 Control Board)
- _____ • 0RN-13A (Train 1A & 2A LLI Supply) (U2 Control Board)
- _____ • 0RN-148A (Train 1A & 2A Disch to RC) (U2 Control Board)
- _____ • 0RN-12AC (Train 1A & 2A LLI Supply) (U2 Control Board)
- _____ • 0RN-147AC (Train 1A & 2A Disch to RC) (U2 Control Board)
- _____ • 0RN-12AC (Train 1A & 2A LLI Supply) (SSF)
- _____ • 0RN-147AC (Train 1A & 2A Disch to RC) (SSF)

Unit 1

____ 3.4.6 **WHEN** desired to align RN Train A to LLI, perform the following:

3.4.6.1 Ensure open:

- ____ • 0RN-13A (Train 1A & 2A LLI Supply)
- ____ • 0RN-12AC (Train 1A & 2A LLI Supply)
- ____ • 0RN-148A (Train 1A & 2A Disch to RC)
- ____ • 0RN-147AC (Train 1A & 2A Disch to RC)

3.4.6.2 Close:

- ____ • 0RN-7A (Train 1A &2A SNSWP Supply)
- ____ • 0RN-149A (Train 1A & 2A Disch to SNSWP)

3.4.6.3 Remove tag and close:

			Tag #
____ DV	____	• 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A)	________
____ DV	____	• Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A)	________

3.4.6.4 Remove Control Board Switch Labels from the following:

- ____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
- ____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.4.6.5 Ensure closed:

- ____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
- ____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

____ SRO 3.4.6.6 Evaluate TSAIL tracking entries for the following:

- ☐ TD CA (Unit 1 A Train Assured Makeup)
- ☐ TD CA (Unit 2 A Train Assured Makeup)

Unit 1

3.5 Defeating RN Train B Automatic Alignment Capability

☐ 3.5.1 Prepare tagout per PP-405.

_____ 3.5.2 Evaluate Tech Specs for the following:
SRO

- ☐ Unit 1 RN Train B
- ☐ Unit 2 RN Train B

3.5.3 Ensure open:

- _____ • 0RN-10AC (Train 1B & 2B LLI Supply)
- _____ • 0RN-11B (Train 1B & 2B LLI Supply)
- _____ • 0RN-283AC (Train 1B & 2B Disch to RC)
- _____ • 0RN-284B (Train 1B & 2B Disch to RC)

3.5.4 Place Control Board Switch Labels for the following:

- _____ • 0RN-10AC (Train 1B & 2B LLI Supply) (U1 Control Board)
- _____ • 0RN-11B (Train 1B & 2B LLI Supply) (U1 Control Board)
- _____ • 0RN-283AC (Train 1B & 2B Disch to RC) (U1 Control Board)
- _____ • 0RN-284B (Train 1B & 2B Disch to RC) (U1 Control Board)
- _____ • 0RN-10AC (Train 1B & 2B LLI Supply) (U2 Control Board)
- _____ • 0RN-11B (Train 1B & 2B LLI Supply) (U2 Control Board)
- _____ • 0RN-283AC (Train 1B & 2B Disch to RC) (U2 Control Board)
- _____ • 0RN-284B (Train 1B & 2B Disch to RC) (U2 Control Board)
- _____ • 0RN-10AC (Train 1B & 2B LLI Supply) (SSF)
- _____ • 0RN-283AC (Train 1B & 2B Disch to RC) (SSF)

3.5.5 Tag open:

			Tag #
_____ DV	_____	• 1EMXH1-1D (Low Level Supply B Shutoff Vlv 0RN-10AC)	_____
_____ DV	_____	• Unit 2 breaker 2EMXH-F2D (Low Level Supply B Shutoff Vlv 0RN-11B)	_____
_____ DV	_____	• 1EMXH1-3C (CCW Discharge B Isolation Vlv 0RN-283AC)	_____
_____ DV	_____	• Unit 2 breaker 2EMXH-F3C (CCW Discharge B Isolation Vlv 0RN-284B)	_____

Unit 1

3.6 Restoring RN Train B Automatic Alignment Capability

3.6.1 *Remove tag and close:*

			Tag #
____ DV	____	• 1EMXH1-1D (Low Level Supply B Shutoff Vlv 0RN-10AC)	________
____ DV	____	• Unit 2 breaker 2EMXH-F2D (Low Level Supply B Shutoff Vlv 0RN-11B)	________
____ DV	____	• 1EMXH1-3C (CCW Discharge B Isolation Vlv 0RN-283AC)	________
____ DV	____	• Unit 2 breaker 2EMXH-F3C (CCW Discharge B Isolation Vlv 0RN-284B)	________

3.6.2 Remove Control Board Switch Labels from the following:

____ • 0RN-10AC (Train 1B & 2B LLI Supply) (U1 Control Board)
____ • 0RN-11B (Train 1B & 2B LLI Supply) (U1 Control Board)
____ • 0RN-283AC (Train 1B & 2B Disch to RC) (U1 Control Board)
____ • 0RN-284B (Train 1B & 2B Disch to RC) (U1 Control Board)
____ • 0RN-10AC (Train 1B & 2B LLI Supply) (U2 Control Board)
____ • 0RN-11B (Train 1B & 2B LLI Supply) (U2 Control Board)
____ • 0RN-283AC (Train 1B & 2B Disch to RC) (U2 Control Board)
____ • 0RN-284B (Train 1B & 2B Disch to RC) (U2 Control Board)
____ • 0RN-10AC (Train 1B & 2B LLI Supply) (SSF)
____ • 0RN-283AC (Train 1B & 2B Disch to RC) (SSF)

____ SRO 3.6.3 Evaluate operability of the following:

☐ Unit 1 RN Train B
☐ Unit 2 RN Train B

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 Automatic backwash of RN Strainer occurs at 1.86 psid (55 INWC). Exceeding automatic initiation set point will **NOT** cause RN System problems. {PIP-0-M96-2223}

2. Initial Conditions

None

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

_____ 3.2 Notify Control Room a manual automatic RN Strainer Backwash is in progress.

3.3 Perform the following sections as applicable:

- ☐ Section 3.4, Manual Automatic Backwash 1A RN Strainer
- ☐ Section 3.5, Manual Automatic Backwash 1B RN Strainer

3.4 Manual Automatic Backwash 1A RN Strainer

☐ 3.4.1 Check 1A RN Pump running.

_____ 3.4.2 On "TB 232" "Nuclear Ser Wtr Strainer 1A Backwash" depress and release black "OPEN" pushbutton.

NOTE:	Operation of 1RN-21A (RN Strainer 1A Backflush Auto Supply) may be erratic due to system pressure fluctuations. {PIP 00-1611}

☐ 3.4.3 Check by visual observation 1RN-21A (RN Strainer 1A Backflush Auto Supply) and 1RN-22A (RN Strainer 1A Backflush Auto Drn) cycle open.

_____ 3.4.4 **WHEN** *approximately 5 minutes* have elapsed, check 1RN-21A (RN Strainer 1A Backflush Auto Supply) and 1RN-22A (RN Strainer 1A Backflush Auto Drn) cycle closed.

_____ 3.4.5 **IF** 1A RN Strainer differential pressure is greater than 50 INWC as read on 1RNPS5000 (RN Pump Strnr A Diff Press), notify Plant SRO for further evaluation.

Unit 1

3.5 Manual Automatic Backwash 1B RN Strainer

☐ 3.5.1 Check 1B RN Pump running.

_____ 3.5.2 On "TB 233" "Nuclear Ser Wtr Strainer 1B Backwash" depress and release black "OPEN" pushbutton.

NOTE:	Operation of 1RN-25B (B RN Strainer Backflush Auto Supply Isol) may be erratic due to system pressure fluctuations. {PIP 00-1611}

☐ 3.5.3 Check by visual observation 1RN-25B (B RN Strainer Backflush Auto Supply Isol) and 1RN-26B (B RN Strainer Backflush Auto Drn) cycle open.

_____ 3.5.4 **WHEN** approximately 5 minutes have elapsed, check 1RN-25B (B RN Strainer Backflush Auto Supply Isol) and 1RN-26B (B RN Strainer Backflush Auto Drn) cycle closed.

_____ 3.5.5 **IF** 1B RN Strainer differential pressure is greater than 50 INWC as read on 1RNPS5010 (RN Pump Strainer B Diff Press), notify Plant SRO for further evaluation.

End of Enclosure

Unit 1

1. Limits and Precautions

1.1 Maximum SNSWP temperature is 82°F. (ITS 3.7.8)

2. Initial Conditions

_____ 2.1 Engineering has recommended aligning Unit 1 RN Non-Essential Header to the SNSWP for additional cooling.

______________________ _____/_____
Person Contacted Date Time

_____ 2.2 RN Train B in operation.

2.3 The following valves open:

- _____ • 0RN-10AC (Train 1B & 2B LLI Supply)
- _____ • 0RN-11B (Train 1B & 2B LLI Supply)
- _____ • 0RN-283AC (Train 1B & 2B Disch to RC)
- _____ • 0RN-284B (Train 1B & 2B Disch to RC)

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections, as applicable:

- ☐ Section 3.3, Aligning RN Train A to the Non-Essential Header
- ☐ Section 3.4, Returning RN Train A to Normal Alignment

3.3 Aligning RN Train A to the Non-Essential Header

NOTE:
- __IF__ any RV Pumps are in operation minimum flow requirements for 1A RN Pump may be unattainable through the Unit 1 RN Non-Essential Header. Minimum flow requirements may be met with flow through 1A KC HX **__OR__** securing RV Pumps.
- Unit 1 RN Train A is operable and available during performance of this enclosure.

☐ 3.3.1 Prepare tagout per PP-364.

☐ 3.3.2 Check SNSWP level greater than 739.5 feet.

_____ 3.3.3 Ensure Unit 2 RN Train A shutdown.

Unit 1

NOTE:	RV Pump operation during performance of this enclosure is undesirable due to potentially moving warmer less oxygenated water from the LLI through Unit 1 Non-Essential Header to the SNSWP.

_____ 3.3.4 **IF** RV Pump(s) in operation perform the following:

☐ 3.3.4.1 List operating RV Pump(s): ______________

_____ 3.3.4.2 Notify System Engineer to determine if operating RV Pump(s) may be secured.

______________________ _____/_____
Person Contacted Date Time

_____ 3.3.4.3 **IF** System Engineer determines RV Pump(s) may be secured, secure operating RV Pump(s) per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

NOTE:	Unit 1 and 2 TD CA Pumps are degraded but will remain operable provided respective Train B RN Essential supply to each TD CA Pump remains operable.

_____ SRO 3.3.5 Declare the following tracking entries in TSAIL: (RN Train A supplies will be isolated).

- ☐ TD CA (Unit 1 A Train Assured Makeup)
- ☐ TD CA (Unit 2 A Train Assured Makeup)

3.3.6 Ensure closed:

- _____ 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
- _____ Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.3.7 Place Control Board Switch Labels for the following:

- _____ 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
- _____ Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.3.8 Tag open:

			Tag #
_____ DV	_____	• 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A)	__________
_____ DV	_____	• Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A)	__________

Unit 1

3.3.9 Open:

_____ • 0RN-7A (Train 1A &2A SNSWP Supply)
_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP)

3.3.10 Close:

_____ • 0RN-147AC (Train 1A & 2A Disch to RC)
_____ • 0RN-12AC (Train 1A & 2A LLI Supply)

_____ 3.3.11 Ensure 1B "RN to KC HX Flow" greater than 3500 gpm.

3.3.12 Ensure open:

_____ • 1RN-42A (AB Non Ess Supply Isol)
_____ • 1RN-40A (Train A to Non Ess Hdr Isol)

NOTE:	Flow through RN Non-Essential Header should meet minimum flow requirements for 1A RN Pump if 1RN-86A (A KC HX Inlet Isol) is closed.

_____ 3.3.13 Ensure "1RN-86A Mode Select" in "AUTO".

NOTE:	Starting RN Pumps may cause associated 4KV Bus OAC low voltage alarms.

_____ 3.3.14 Start 1A RN Pump.

3.3.15 Close one of the following:

_____ • 1RN-41B (Train B to Non Ess Hdr Isol)
<u>OR</u>
_____ • 1RN-43A (Train B to Non Ess Hdr Isol)

NOTE:	**<u>IF</u>** any RV Pumps are required to be in operation 1RN-89A (RN to A KC HX Control) may need to remain throttled open to meet 1A RN Pump minimum flow requirements through the 1A KC HX.

_____ 3.3.16 Maintain 1A RN Pump flow greater than 3000 gpm by throttling 1RN-89A (RN to A KC HX Control) to reduce 1A KC HX flow as much as possible.

☐ 3.3.17 Maintain SNSWP temperature less than 82°F. (ITS 3.7.8)

Unit 1

3.4 Returning RN Train A to Normal Alignment

3.4.1 Ensure open:

_____ • 1RN-41B (Train B to Non Ess Hdr Isol)
_____ • 1RN-43A (Train B to Non Ess Hdr Isol)

3.4.2 Open:

_____ • 0RN-147AC (Train 1A & 2A Disch to RC)
_____ • 0RN-12AC (Train 1A & 2A LLI Supply)

3.4.3 Close:

_____ • 0RN-7A (Train 1A &2A SNSWP Supply)
_____ • 0RN-149A (Train 1A & 2A Disch to SNSWP)

_____ 3.4.4 **IF** desired to secure 1A RN Pump, perform the following:

_____ 3.4.4.1 Place "1RN-86A Mode Select" in "MANUAL".

_____ 3.4.4.2 Depress close for 1RN-86A (A KC HX Inlet Isol).

_____ 3.4.4.3 Monitor RN to KC HX Flow and stop 1A RN Pump prior to 1RN-86A (A KC HX Inlet Isol) being fully closed.

_____ 3.4.4.4 Place "1RN-86A Mode Select" in "AUTO".

3.4.5 Remove tag and close:

Tag #

_____ DV _____ • 1EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 1A Vlv 1CA-86A) __________
_____ DV _____ • Unit 2 breaker 2EMXA-F6C (T/D Aux FDWP Suct From RN Hdr 2A Vlv 2CA-86A) __________

3.4.6 Remove Control Board Switch Labels from the following:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

3.4.7 Ensure closed:

_____ • 1CA-86A (U1 TD CA Pump Suction From 1A RN Isol)
_____ • Unit 2 valve 2CA-86A (U2 TD CA Pump Suction From 2A RN Isol)

Unit 1

_____ 3.4.8 Evaluate TSAIL tracking entries for the following:

SRO

- ☐ TD CA (Unit 1 A Train Assured Makeup)
- ☐ TD CA (Unit 2 A Train Assured Makeup)

NOTE:	**IF** 1A RN Pump in operation, 1RN-86A (A KC HX Inlet Isol) will open when selected to "AUTO".

_____ 3.4.9 Ensure "1RN-86A Mode Select" in "AUTO".

_____ 3.4.10 **IF** desired to secure one train of RN, secure per Enclosure 4.8 (Shifting Operational Alignment of RN Pumps).

End of Enclosure

Duke Power Company McGuire Nuclear Station **Containment Ventilation System** **Continuous Use**	Procedure No. **OP/ 1/A/6450/001** Revision No. **032** Electronic Reference No. **MC004759**

PERFORMANCE

* * * * * * * * * * UNCONTROLLED FOR PRINT * * * * * * * * * *

(ISSUED) - PDF Format

Revision History (significant issues, limited to one page)

Rev 032 (06/17/03)

- Body and Enclosure 4.2
  -Changed L&P 2.3 and 1.3 respectively to read "**IF** three VL AHUs operating in "HIGH", "B", "C", and "D" should be in operation to maximize cooling for DRPI cabinets.".
- Enclosure 4.2
  -Added bullet to note prior to step 3.7.1 stating running 4 VL AHUs in low speed is preferred alignment.
  -Changed step 3.7.3.3 to read "**IF** four VL AHUs operating in "LOW", continue in this alignment unless challenging LCWAT Tech Spec lower limit, then stop one VL AHU per Section 3.9.".
  -Deleted steps 3.7.3.4 and 3.7.5.5.
  -Added note prior to step 3.3.4 to read ""HIGH" speed light should be lit 1 second after shifting VL AHU to "HIGH".
  -Revised step 3.3.4 to read "**IF** desired to operate in high speed **AND** "LOW" light lit for minimum of 1 minute, select "HIGH" for operating VL AHU(s):".
  -Added new step 3.3.5 to direct shifting VL AHUs from high to low speed.
  -Added new step 3.2 to direct maintaining Lower Containment Temp within Tech Spec limits.
- Enclosure 4.1
  - Added new step 3.2 to direct maintaining Lower Containment Temp within Tech Spec limits.
- Enclosure 4.5
  -Changed note prior to step 3.3.11.6 and 3.4.11.6 to read ""HIGH" speed light should be lit 1 second after shifting VL AHU to "HIGH".
  -Changed step 3.3.11.6 and 3.4.11.6 to read "**IF** desired to operate in high speed **AND** "LOW" light lit for minimum of 1 minute, select "HIGH" for operating VL AHU(s):".

Rev 031 (09/25/02)

- Added a note and steps to Enclosure 4.2 to ensure proper VL AHU operation for VR fan operation. At least 2 VL AHU should be in operation when 3 VR fans are operating. {PIP M-02-00884)
- Added steps to Enclosure 4.5 to provide better guidance for realigning VL AHU following an SI. The AHUs should be running and in high speed following an SI.

Unit 1

Containment Ventilation System

1. Purpose

Provide instructions for the operation of Containment Ventilation System.

2. Limits and Precautions

2.1 All VL AHUs should be operated in the same speed.

2.2 Only one Pipe Tunnel Booster Fan should be operated at a time.

2.3 **IF** three VL AHUs operating in "HIGH", "B", "C", and "D" should be in operation to maximize cooling for DRPI cabinets.

2.4 **WHEN** two VL AHUs operating, VL AHUs on opposite sides of Containment Building should be in operation.

3. Procedure

See Section 4.

4. Enclosures

4.1 Upper Containment Ventilation Operation

4.2 Lower Containment Ventilation Operation

4.3 Control Rod Drive Ventilation System Operation

4.4 Incore Instrument Ventilation System Operation

4.5 Realignment Of Containment Ventilation Systems Following SI

End Of Body

Unit 1

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 Cooling water is available to VU AHUs per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

NOTE:	VU System temperature controllers are set to maintain preferred normal operating Upper Containment temperature 78 - 82 °F.

☐ 3.2 Maintain Upper Containment temperature within limits of Tech Spec 3.6.5 (Containment Air Temperature).

3.3 Perform the following sections as applicable:

- ☐ Section 3.4, VU AHU Startup
- ☐ Section 3.5, Operating VU AHUs in "MAX COOL"
- ☐ Section 3.6, VU AHU Shutdown

3.4 VU AHU Startup

3.4.1 Ensure in "AUTO":

_____ • 1A RA Fan Mode Select
_____ • 1B RA Fan Mode Select
_____ • 1C RA Fan Mode Select
_____ • 1D RA Fan Mode Select

3.4.2 Place desired VU AHU(s) in "NORM":

_____ • 1A VU AHU Mode Select
_____ • 1B VU AHU Mode Select
_____ • 1C VU AHU Mode Select
_____ • 1D VU AHU Mode Select

3.4.3 Check Return Air Fan "ON" with associated operating VU AHU(s):

☐ 1A RA Fan
☐ 1B RA Fan
☐ 1C RA Fan
☐ 1D RA Fan

NOTE:	OAC points M1A1204, M1A1210, M1A1216, and M1A1222 can be used as indication of containment temperature.

3.5 Operating VU AHUs in "MAX COOL"

_____ 3.5.1 **IF** additional cooling is needed, perform the following:

3.5.1.1 Place desired VU AHU(s) in "MAX COOL":

_____ • 1A VU AHU Mode Select
_____ • 1B VU AHU Mode Select
_____ • 1C VU AHU Mode Select
_____ • 1D VU AHU Mode Select

3.5.1.2 Check lit for all VU AHUs selected to "MAX COOL":

☐ 1A VU AHU Thermostat BYPASSED
☐ 1B VU AHU Thermostat BYPASSED
☐ 1C VU AHU Thermostat BYPASSED
☐ 1D VU AHU Thermostat BYPASSED

_____ 3.5.2 **<u>WHEN</u>** "MAX COOL" no longer required, perform the following:

3.5.2.1 Place affected VU AHU in "NORM":

- _____ • 1A VU AHU Mode Select
- _____ • 1B VU AHU Mode Select
- _____ • 1C VU AHU Mode Select
- _____ • 1D VU AHU Mode Select

3.5.2.2 Check dark for VU AHUs in "NORM":

- ☐ 1A VU AHU Thermostat BYPASSED
- ☐ 1B VU AHU Thermostat BYPASSED
- ☐ 1C VU AHU Thermostat BYPASSED
- ☐ 1D VU AHU Thermostat BYPASSED

3.6 *VU AHU Shutdown*

3.6.1 Place in "OFF" for desired VU AHU(s):

- _____ • 1A VU AHU Mode Select
- _____ • 1B VU AHU Mode Select
- _____ • 1C VU AHU Mode Select
- _____ • 1D VU AHU Mode Select

3.6.2 Check Return Air Fan "OFF" with associated shutdown VU AHU(s):

- ☐ 1A RA Fan
- ☐ 1B RA Fan
- ☐ 1C RA Fan
- ☐ 1D RA Fan

End Of Enclosure

1. Limits and Precautions

1.1 All VL AHUs should be operated in the same speed.

1.2 Only one Pipe Tunnel Booster Fan should be operated at a time.

1.3 IF three VL AHUs operating in "HIGH", "B", "C", and "D" should be in operation to maximize cooling for DRPI cabinets.

1.4 WHEN two VL AHUs operating, VL AHUs on opposite sides of Containment Building should be in operation.

2. Initial Conditions

_____ 2.1 IF Unit 1 in Modes 1 - 5, cooling water is available to VL AHUs per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

☐ 3.2 Maintain Lower Containment temperature within limits of Tech Spec 3.6.5 (Containment Air Temperature).

3.3 Perform the following sections as applicable:

- ☐ Section 3.4, Starting VL AHUs
- ☐ Section 3.5, Starting Pipe Tunnel Booster Fan
- ☐ Section 3.6, Starting S/G Booster Fans
- ☐ Section 3.7, Starting Pressurizer Booster Fan
- ☐ Section 3.8, Optimizing VL System Configuration During Normal Operation
- ☐ Section 3.9, Stopping VL AHUs
- ☐ Section 3.10, Stopping Pipe Tunnel Booster Fan
- ☐ Section 3.11, Stopping S/G Booster Fans
- ☐ Section 3.12, Stopping Pressurizer Booster Fan

3.4 Starting VL AHUs

3.4.1 Ensure in "AUTO":

_____ • 1LCVU-D-1 (1A VL AHU Suction Damper Cntrl)
_____ • 1LCVU-D-2 (1B VL AHU Suction Damper Cntrl)
_____ • 1LCVU-D-3 (1C VL AHU Suction Damper Cntrl)
_____ • 1LCVU-D-4 (1D VL AHU Suction Damper Cntrl)

NOTE:	"LOW" speed light should be lit 7 seconds after starting VL AHU.

3.4.2 Select "LOW" for desired VL AHU(s):

_____ • 1A VL AHU Mode Select
_____ • 1B VL AHU Mode Select
_____ • 1C VL AHU Mode Select
_____ • 1D VL AHU Mode Select

3.4.3 Check "OPEN" on operating VL AHU(s):

☐ 1LCVU-D-5 (1A VL AHU Disch Damper)
☐ 1LCVU-D-6 (1B VL AHU Disch Damper)
☐ 1LCVU-D-7 (1C VL AHU Disch Damper)
☐ 1LCVU-D-8 (1D VL AHU Disch Damper)

NOTE:	"HIGH" speed light should be lit 1 second after shifting VL AHU to "HIGH".

_____ 3.4.4 **IF** desired to operate in high speed **AND** "LOW" light lit for minimum of 1 minute, select "HIGH" for operating VL AHU(s):

_____ • 1A VL AHU Mode Select
_____ • 1B VL AHU Mode Select
_____ • 1C VL AHU Mode Select
_____ • 1D VL AHU Mode Select

NOTE:	"LOW" speed light should be lit 7 seconds after shifting VL AHU to "LOW".

_____ 3.4.5 **IF** desired to shift from high speed to low speed, select "LOW" for operating VL AHU(s):

_____ • 1A VL AHU Mode Select
_____ • 1B VL AHU Mode Select
_____ • 1C VL AHU Mode Select
_____ • 1D VL AHU Mode Select

NOTE:	Pipe Tunnel Booster Fans may be operated at different speed from VL AHUs.

3.5 Starting Pipe Tunnel Booster Fan

3.5.1 Place one of the following in "HIGH":

_____ • 1A Pipe Tunnel Booster Fan
OR
_____ • 1B Pipe Tunnel Booster Fan

_____ 3.5.2 **IF** neither fan is available for high speed operation, place one of the following in "LOW":

_____ • 1A Pipe Tunnel Booster Fan
OR
_____ • 1B Pipe Tunnel Booster Fan

3.5.3 Check "OPEN" for operating fan:

☐ 1PTBF-D-1 (1A Pipe Tunnel Bstr Fan Disch)
OR
☐ 1PTBF-D-2 (1B Pipe Tunnel Bstr Fan Disch)

3.6 S/G Booster Fan Startup

3.6.1 Select "ON" for desired fan(s):

_____ • A S/G Booster Fan
_____ • B S/G Booster Fan
_____ • C S/G Booster Fan
_____ • D S/G Booster Fan

3.7 Pressurizer Booster Fan Startup

3.7.1 Position "Press Booster" to desired fan:

_____ • Fan A
OR
_____ • Fan B

3.8 Optimizing VL System Configuration During Normal Operation

NOTE:
- VL System may be operated in other than "optimized configuration" when conditions warrant. Some examples of warranted conditions are as follows:
  - Minimizing Lower Containment temperature for personnel entry
  - Minimizing Lower Containment temperature during shutdown
  - Minimizing Lower Containment temperature to alleviate stress on equipment
  - VL configuration to allow for equipment maintenance
- Preferred VL System configuration is 4 VL AHUs operating in low speed due to improved air flow distribution and minimal ventilation equipment stress.
- A minimum of 2 VL AHUs should be in operation when 3 VR fans are operating to ensure proper VR system flow.

☐ 3.8.1 Record Lower Containment Weighted Average Temperature (LCWAT) using OAC point M1P0755. ________ °F

_____ SRO 3.8.2 <u>**IF**</u> LCWAT indicates less than 115 °F, evaluate securing any operating RV pumps per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

_____ 3.8.3 <u>**IF**</u> LCWAT indicates less than 110 °F, perform one of the following:

_____ 3.8.3.1 <u>**IF**</u> four VL AHUs operating in "HIGH", stop one VL AHU per Section 3.9.

_____ 3.8.3.2 <u>**IF**</u> three VL AHUs operating in "HIGH", place all VL AHUs in "LOW" per Section 3.4.

_____ 3.8.3.3 <u>**IF**</u> four VL AHUs operating in "LOW", continue in this alignment unless challenging LCWAT Tech Spec lower limit, then stop one VL AHU per Section 3.9.

_____ 3.8.4 <u>**IF**</u> LCWAT indicates 115 - 120 °F, evaluate placing RV pumps in service per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

Unit 1

_____ 3.8.5 **IF** LCWAT indicates greater than 120 °F, perform one of the following:

_____ 3.8.5.1 **IF** four VL AHUs operating in "HIGH", place RV pumps in service per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

_____ 3.8.5.2 **IF** three VL AHUs operating in "HIGH", place remaining VL AHU in "HIGH" per Section 3.4.

_____ 3.8.5.3 **IF** four VL AHUs operating in "LOW", perform the following:

_____ A. Stop one VL AHU per Section 3.9.

_____ B. Place three VL AHUs in "HIGH" per Step 3.4.4.

_____ 3.8.5.4 **IF** three VL AHUs operating in "LOW", start remaining VL AHU in "LOW" per Section 3.4.

3.9 Stopping VL AHUs

3.9.1 Select "OFF" for desired VL AHU(s):

- _____ 1A VL AHU Mode Select
- _____ 1B VL AHU Mode Select
- _____ 1C VL AHU Mode Select
- _____ 1D VL AHU Mode Select

3.9.2 Check "CLOSED" for secured VL AHU(s):

- ☐ 1LCVU-D-5 (1A VL AHU Disch Damper)
- ☐ 1LCVU-D-6 (1B VL AHU Disch Damper)
- ☐ 1LCVU-D-7 (1C VL AHU Disch Damper)
- ☐ 1LCVU-D-8 (1D VL AHU Disch Damper)

3.10 Stopping Pipe Tunnel Booster Fan

3.10.1 Ensure in "OFF":

_____ • 1A Pipe Tunnel Booster Fan
_____ • 1B Pipe Tunnel Booster Fan

3.11 Stopping S/G Booster Fans

3.11.1 Place in "OFF":

_____ • S/G A Booster Fan
_____ • S/G B Booster Fan
_____ • S/G C Booster Fan
_____ • S/G D Booster Fan

3.12 Stopping Pressurizer Booster Fan

_____ 3.12.1 Place "Press Booster" in "OFF".

End Of Enclosure

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 **IF** swapping currently running VR fans, OSM concurrence has been received.
OSM

_____ 2.2 **IF** VR System is being shutdown, Control Rod Drive units are de-energized.

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, VR System Startup
- ☐ Section 3.4, Operating VR System for Additional Cooling
- ☐ Section 3.5, VR System Shutdown

3.3 VR System Startup

3.3.1 Check at least 2 VL AHUs in operation per Enclosure 4.2 (Lower Containment Ventilation Operation).

_____ 3.3.2 Contact VR System Engineer for desired operating configuration.

_______________________ _____/_____
Person Contacted Date Time

☐ 3.3.3 Record recommended operating configuration:

_____ CRD Vent Fan
_____ CRD Vent Fan
_____ CRD Vent Fan

3.3.4 Place recommended fans in "START":

_____ • 1A CRD Vent Fan
_____ • 1B CRD Vent Fan
_____ • 1C CRD Vent Fan
_____ • 1D CRD Vent Fan

3.3.5 Check "OPEN" for operating fans:

- ☐ 1CRDM-D1 (A CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D2 (B CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D3 (C CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D4 (D CRD Vent Fan Disch Damper)

3.4 Operating VR System for Additional Cooling

_____ 3.4.1 **IF** additional cooling required, perform the following:

3.4.1.1 Place idle CRD Vent Fan in "START":

- _____ • 1A CRD Vent Fan
- _____ • 1B CRD Vent Fan
- _____ • 1C CRD Vent Fan
- _____ • 1D CRD Vent Fan

3.4.1.2 Check "OPEN" for operating CRD Vent Fans:

- ☐ 1CRDM-D1 (A CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D2 (B CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D3 (C CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D4 (D CRD Vent Fan Disch Damper)

_____ 3.4.2 **IF** additional cooling no longer required, perform the following:

3.4.2.1 Stop one CRD Vent Fan:

- _____ • 1A CRD Vent Fan
- _____ • 1B CRD Vent Fan
- _____ • 1C CRD Vent Fan
- _____ • 1D CRD Vent Fan

3.4.2.2 Check "CLOSED" for idle CRD Vent Fan:

- ☐ 1CRDM-D1 (A CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D2 (B CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D3 (C CRD Vent Fan Disch Damper)
- ☐ 1CRDM-D4 (D CRD Vent Fan Disch Damper)

Unit 1

3.5 VR System Shutdown

3.5.1 Depress "STP" for operating CRD Vent Fans:

_____ • 1A CRD Vent Fan
_____ • 1B CRD Vent Fan
_____ • 1C CRD Vent Fan
_____ • 1D CRD Vent Fan

3.5.2 Check "CLOSED":

☐ 1CRDM-D1 (A CRD Vent Fan Disch Damper)
☐ 1CRDM-D2 (B CRD Vent Fan Disch Damper)
☐ 1CRDM-D3 (C CRD Vent Fan Disch Damper)
☐ 1CRDM-D4 (D CRD Vent Fan Disch Damper)

End Of Enclosure

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 Cooling water is being supplied to VT AHUs per OP/0/A/6400/009 (Containment Ventilation Cooling Water System).

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, VT System Startup
- ☐ Section 3.4, VT System Additional Cooling
- ☐ Section 3.5, VT System Shutdown

3.3 VT System Startup

3.3.1 Place desired VT AHU in "ON":

- _____ 1A VT AHU Incore Inst Room
- _____ 1B VT AHU Incore Inst Room

3.4 VT System Additional Cooling

3.4.1 Place operating VT AHU in "MAX":

- _____ 1A VT AHU Incore Inst Room
- _____ 1B VT AHU Incore Inst Room

_____ 3.4.2 **IF** two VT AHUs required for additional cooling, place idle AHU in "MAX":

- _____ 1A VT AHU Incore Inst Room
- _____ 1B VT AHU Incore Inst Room

Unit 1

_____ 3.4.3 **WHEN** additional cooling is no longer required, perform one of the following:

_____ 3.4.3.1 **IF** two VT AHUs operating in "MAX", perform the following:

A. Place one VT AHU in "OFF":

- _____ 1A VT AHU Incore Inst Room
  OR
- _____ 1B VT AHU Incore Inst Room

B. Place operating VT AHU in "ON":

- _____ 1A VT AHU Incore Inst Room
- _____ 1B VT AHU Incore Inst Room

_____ 3.4.3.2 **IF** one VT AHU operating in "MAX, place operating VT AHU in "ON":

- _____ 1A VT AHU Incore Inst Room
  OR
- _____ 1B VT AHU Incore Inst Room

3.5 VT System Shutdown

3.5.1 Place in "OFF":

- _____ 1A VT AHU Incore Inst Room
- _____ 1B VT AHU Incore Inst Room

End Of Enclosure

1. Limits and Precautions

None

2. Initial Conditions

_____ 2.1 SI and D/G Sequencer reset.

3. Procedure

☐ 3.1 Evaluate all outstanding R&Rs that may impact performance of this procedure.

3.2 Perform the following sections as applicable:

- ☐ Section 3.3, A Train Realignment
- ☐ Section 3.4, B Train Realignment

3.3 A Train Realignment

_____ (SRO) 3.3.1 **IF** 1ETA is being supplied by D/G, evaluate loads on the D/G.

3.3.2 Place in "OFF":

- _____ 1A VU AHU Mode Select
- _____ 1C VU AHU Mode Select

3.3.3 Place in "STOP":

- _____ 1A RA Fan Mode Select
- _____ 1C RA Fan Mode Select

3.3.4 Place in "OFF" for any non-operating fans:

- _____ 1A VL AHU Mode Select
- _____ 1C VL AHU Mode Select
- _____ 1A VT AHU Incore Inst Room
- _____ 1A Pipe Tunnel Booster Fan

3.3.5 Ensure "HIGH" selected for operating VL AHUs:

- _____ 1A VL AHU Mode Select
- _____ 1C VL AHU Mode Select

3.3.6 Place in "STOP" for any non-operating fans:

_____ • 1A CRD Vent Fan
_____ • 1C CRD Vent Fan

3.3.7 Reset and close:

_____ DV _____ • 1EMXC-1A (1A Lower Containment Cooling Unit Motor)

_____ DV _____ • 1EMXC-2A (1C Lower Containment Cooling Unit Motor)

_____ DV _____ • 1EMXC-3C (1A Control Rod Drive Vent Fan Motor)

_____ DV _____ • 1EMXC-3D (1C Control Rod Drive Vent Fan Motor)

_____ DV _____ • 1EMXC-8C (1A Incore Instrument Room AHU Motor)

_____ DV _____ • 1EMXC-3B (Pzr Compt Fan A Normal Source To Transfer Switch)

_____ DV _____ • 1EMXC-6B (1A Upper Containment AHU Motor)

_____ DV _____ • 1EMXC-6C (1C Upper Containment AHU Motor)

_____ DV _____ • 1EMXC-6A (1A Containment Pipe Tunnel Booster Fan Motor)

_____ DV _____ • 1EMXC-7C (1C Upper Containment Return Air Fan Motor)

_____ DV _____ • 1EMXC-8D (1A Upper Containment Return Air Fan Motor)

NOTE:
- Alternate power for A Train is 1LXD-1D.
- Alternate power must be available to re-transfer to normal power source.

_____ 3.3.8 **IF** 1LXD de-energized, go to Step 3.3.11.

3.3.9 Turn the following transfer switches to "RETRANSFER" in switchgear room ETA. (transfer may take 5-15 seconds.)

_____ • 1VLTS-1A
_____ • 1VLTS-1C
_____ • 1VRTS-1A
_____ • 1VRTS-1C
_____ • 1VTTS-1A

_____ 3.3.10 Turn "Press Booster Fan 1A Transfer Switch" to "RETRANSFER" in MG set room. (transfer may take 5-15 seconds)

3.3.11 Place VL Fans in service as follows:

☐ 3.3.11.1 Check NR Containment Pressure gauge in service.

_____ 3.3.11.2 **IF** 1A **AND** 1C VL AHUs running as desired, go to Step 3.3.12.

3.3.11.3 Ensure in "AUTO":

- _____ 1LCVU-D-1 (1A VL AHU Suction Damper Cntrl)
- _____ 1LCVU-D-3 (1C VL AHU Suction Damper Cntrl)

NOTE:	"LOW" speed light should be lit 7 seconds after starting VL AHU.

3.3.11.4 Select "LOW" for the desired VL AHU(s):

- _____ 1A VL AHU Mode Select
- _____ 1C VL AHU Mode Select

3.3.11.5 Check "OPEN" on operating VL AHU(s):

☐ 1LCVU-D-5 (1A VL AHU Disch Damper)
☐ 1LCVU-D-7 (1C VL AHU Disch Damper)

NOTE:	"HIGH" speed light should be lit 1 second after shifting VL AHU to "HIGH".

_____ 3.3.11.6 **IF** desired to operate in high speed **AND** "LOW" light lit for minimum of 1 minute, select "HIGH" for operating VL AHU(s):

- _____ 1A VL AHU Mode Select
- _____ 1C VL AHU Mode Select

CAUTION: IF containment temperature is elevated, containment pressure may decrease rapidly.

_____ 3.3.11.7 IF containment pressure drops to 0.05 psig, perform the following:

_____ A. Remove one VL AHU from service.

☐ B. Monitor containment pressure.

_____ C. IF containment pressure continues to decrease, perform the following to maintain positive pressure in containment.

_____ • Select "LOW" for operating AHUs
OR
_____ • Remove additional VL AHU(s)

☐ 3.3.12 Align Containment Ventilation per applicable enclosures.

3.4 B Train Realignment

_____ (SRO) 3.4.1 IF 1ETB is being supplied by D/G, evaluate loads on the D/G.

3.4.2 Place in "OFF":

_____ • 1B VU AHU Mode Select
_____ • 1D VU AHU Mode Select

3.4.3 Place in "STOP":

_____ • 1B RA Fan Mode Select
_____ • 1D RA Fan Mode Select

3.4.4 Place in "OFF" for any non-operating fans:

_____ • 1B VL AHU Mode Select
_____ • 1D VL AHU Mode Select
_____ • 1B VT AHU Incore Inst Room
_____ • 1B Pipe Tunnel Booster Fan

3.4.5 Ensure "HIGH" selected for operating VL AHUs:

_____ • 1B VL AHU Mode Select
_____ • 1D VL AHU Mode Select

3.4.6 Place in "STOP" for any non-operating fans:

- ____ • 1B CRD Vent Fan
- ____ • 1D CRD Vent Fan

3.4.7 Reset and close:

- ____ (DV) ____ • 1EMXD-1A (1B Lower Containment Cooling Unit Motor)
- ____ (DV) ____ • 1EMXD-2A (1D Lower Containment Cooling Unit Motor)
- ____ (DV) ____ • 1EMXD-3C (1B Control Rod Drive Vent Fan Motor)
- ____ (DV) ____ • 1EMXD-3D (1D Control Rod Drive Vent Fan Motor)
- ____ (DV) ____ • 1EMXD-8B (1B Incore Instrument Room AHU Motor)
- ____ (DV) ____ • 1EMXD-8A (Pzr Compt Fan B Normal Source To Transfer Switch)
- ____ (DV) ____ • 1EMXD-6C (1B Upper Containment AHU Motor)
- ____ (DV) ____ • 1EMXD-6D (1D Upper Containment AHU Motor)
- ____ (DV) ____ • 1EMXD-3B (1B Containment Pipe Tunnel Booster Fan Motor)
- ____ (DV) ____ • 1EMXD-7B (1D Upper Containment Return Air Fan Motor)
- ____ (DV) ____ • 1EMXD-8C (1B Upper Containment Return Air Fan Motor)

NOTE:
- Alternate power for B Train is 1LXB-1C.
- Alternate power must be available to re-transfer to normal power source.

____ 3.4.8 **IF** 1LXB de-energized, go to Step 3.4.11.

3.4.9 Turn the following transfer switches to "RETRANSFER" in switchgear room ETB. (transfer may take 5-15 seconds.)

- ____ • 1VLTS-1B
- ____ • 1VLTS-1D
- ____ • 1VRTS-1B
- ____ • 1VRTS-1D
- ____ • 1VTTS-1B

_____ 3.4.10 Turn "Press Booster Fan 1B Transfer Switch" to "RETRANSFER" in MG set room. (transfer may take 5-15 seconds)

3.4.11 Place VL Fans in service as follows:

☐ 3.4.11.1 Check NR Containment Pressure gauge in service.

_____ 3.4.11.2 **IF** 1B **AND** 1D VL AHUs running as desired, go to Step 3.4.12.

3.4.11.3 Ensure in "AUTO":

- _____ 1LCVU-D-2 (1B VL AHU Suction Damper Cntrl)
- _____ 1LCVU-D-4 (1D VL AHU Suction Damper Cntrl)

NOTE:	"LOW" speed light should be lit 7 seconds after starting VL AHU.

3.4.11.4 Select "LOW" for the desired VL AHU(s):

- _____ 1B VL AHU Mode Select
- _____ 1D VL AHU Mode Select

3.4.11.5 Check "OPEN" on operating VL AHU(s):

- ☐ 1LCVU-D-6 (1B VL AHU Disch Damper)
- ☐ 1LCVU-D-8 (1D VL AHU Disch Damper)

NOTE:	"HIGH" speed light should be lit 1 second after shifting VL AHU to "HIGH".

_____ 3.4.11.6 **IF** desired to operate in high speed **AND** "LOW" light lit for minimum of 1 minute, select "HIGH" for operating VL AHU(s):

- _____ 1B VL AHU Mode Select
- _____ 1D VL AHU Mode Select

CAUTION: **IF** containment temperature is elevated, containment pressure may decrease rapidly.

_____ 3.4.11.7 **IF** containment pressure drops to 0.05 psig, perform the following:

_____ A. Remove one VL AHU from service.

☐ B. Monitor containment pressure.

_____ C. **IF** containment pressure continues to decrease, perform the following to maintain positive pressure in containment.

_____ • Select "LOW" for operating AHUs
OR
_____ • Remove additional VL AHU(s)

☐ 3.4.12 Align Containment Ventilation per applicable enclosures.

End Of Enclosure

Unit 1