A.1.a RO/SRO

TITLE: Critical Safety Function Status Tree Evaluation.
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 15 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☐ TIME CRITICAL ☐ PRA

JPM DIRECTIONS:
1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Requiring the examinee to acquire the required materials may or may not be included as part of the JPM.

TASK STANDARD: Upon successful completion of this JPM, the examinee will:

- Correctly assess and determine the status of ALL CSFs and then determine which FRP is required to be implemented using FNP-2-CSF-0.0.

Examinee:		
Overall JPM Performance:	**Satisfactory** ☐	**Unsatisfactory** ☐
Evaluator Comments (attach additional sheets if necessary)		

EXAMINER: ________________________

Developer	S. Jackson	Date: 4/2/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to MONITOR AND EVALUATE CRITICAL SAFETY FUNCTION STATUS TREES. The conditions under which this task is to be performed are:

a. Unit 2 tripped from 100% power and Safety Injected 30 minutes ago.
b. Plant conditions are given in the attached Table 1.
c. The crew is performing actions in EEP-1, Loss of Reactor or Secondary Coolant.
d. The SPDS computer is **NOT** available for monitoring Critical Safety Functions.
e. You have been directed to manually monitor the Critical Safety Functions using CSF-0.0, Critical Safety Function Status Trees, on Unit 2.

Your Task is to:

1. Document each CSF evaluation on **FNP-2-CSF-0.0** by circling the final colored ball indicating the CSF status.

2. Report the FRP that is required to be implemented, if any.

INITIATING **CUE**: IF you have no questions, you may begin.

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
_____ **START TIME**		
* 1. Evaluate CSF-0.1.	POWER RNG LESS THAN 5% - **YES** BOTH INT RNG SUR ZERO OR NEGATIVE – **NO** Determines that an **Orange** condition exists to go to FRP-S.1.	S / U
* 2. Evaluate CSF-0.2.	FIFTH HOTTEST CORE EXIT TC LESS THAN 1200°F – **YES** RCS SUBCOOLING FROM CORE EXIT TC'S GRTR THAN 16°F{45°F} – **YES** Determines that this CSF is SAT.	S / U

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
* 3. Evaluate CSF-0.3.	NAR RNG LVL IN AT LEAST ONE SG GRTR THAN 31%{48%} - **NO** TOTAL AFW FLOW TO ALL SG'S GRTR THAN 395 GPM – **YES** PRESS IN ALL SG'S LESS THAN 1129 PSIG – **YES** NAR RNG LVL IN ALL SG'S LESS THAN 82% - **YES** PRESS IN ALL SG'S LESS THAN 1075 PSIG – **YES** NAR RNG LVL IN ALL SG'S GRTR THAN 31% - **NO** Determines that a Yellow condition exists to go to FRP-H.5.	S / U
* 4. Evaluate CSF-0.4.	TEMP DECR IN ALL CL IN LAST 60 MIN LESS THAN 100°F – **NO** ALL RCS PRESS CL TEMP (IN LAST 60 MIN) POINTS TO RIGHT OF LIMIT A – **YES** ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 285°F – **NO** Determines that an **<u>Orange</u>** condition exists to go to FRP-P.1.	S / U

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
* 5. Evaluate CSF-0.5.	CTMT PRESS LESS THAN 54 PSIG – **YES** CTMT PRESS LESS THAN 27 PSIG – **YES** CTMT SUMP LVL LESS THAN 7.6 FT. – **YES** BOTH CTMT RAD LESS THAN 2 R/hr. - **YES** Determines that this CSF is SAT.	S / U
* 6. Evaluate CSF-0.6.	PRZR LVL LESS THAN 92% - **YES** PRZR LVL GRTR THAN 15% - **NO** Determines that a Yellow condition exists to go to FRP-I.2.	S / U
* 7. Determines FRP entry requirements.	Determines that FRP-S.1 is required to be implemented.	S / U

_____ **STOP TIME**

Terminate when all elements of the task have been completed.

<u>CRITICAL ELEMENTS</u>: Critical Elements are denoted with an asterisk (∗) before the element number.

GENERAL REFERENCES:

1. FNP-2-CSF-0.0, ver 12.0
2. KA: G2.1.7 - 4.4 / 4.7
   G2.1.20 – 4.6 / 4.6

GENERAL TOOLS AND EQUIPMENT:

1. FNP-2-CSF-0.0, ver 12.0 – on Reference disk
2. FNP-2-CSF-0.0, ver 12.0 – paper copy

Critical ELEMENT justification:

STEP	Evaluation
1	**Critical:** Task completion: required to properly evaluate CSF-0.1 to determine that an Orange path condition exists. This is the highest priority FRP for the conditions given. If this is not evaluated properly, a transition to a lower level procedure could occur, and the highest priority FRP would not be implemented.
2-6	**Critical:** Task completion: Actions are required to evaluate each CSF properly to complete task successfully. This CSF evaluation should determine the CSF color and procedure, if any, that apply.
7	**Critical:** Task completion: required to determine that FRP-S.1 is to be implemented.

COMMENTS:

CONDITIONS

When I tell you to begin, you are to MONITOR AND EVALUATE CRITICAL SAFETY FUNCTION STATUS TREES. The conditions under which this task is to be performed are:

a. Unit 2 tripped from 100% power and Safety Injected 30 minutes ago.
b. Plant conditions are given in the attached Table 1.
c. The crew is performing actions in EEP-1, Loss of Reactor or Secondary Coolant.
d. The SPDS computer is **NOT** available for monitoring Critical Safety Functions.
e. You have been directed to manually monitor the Critical Safety Functions using CSF-0.0, Critical Safety Function Status Trees, on Unit 2.

Your Task is to:

1. Document each CSF evaluation on **FNP-2-CSF-0.0** by circling the final colored ball indicating the CSF status.

2. Report the FRP that is required to be implemented, if any.

Table 1

Parameter	INSTRUMENT			
	Channel I or Train A	Channel II or Train B	Channel III	Channel IV
Power Range NI	0%	0%	0%	0%
Intermediate Range SUR	+0.2 DPM	+0.25 DPM		
Intermediate Range NI	$3.0x10^{-8}$ AMPS	$3.2x10^{-8}$ AMPS		
Source Range SUR	0 DPM	0 DPM		
Source Range NI	0 CPS	0 CPS		
RCS Pressure	1575 psig	1550 psig		
MCB Core Exit T/C Monitor in **TMAX** mode	329°F	325°F		
PRZR level	2%	4%	5%	
CTMT Pressure	0 psig	0 psig	0 psig	0 psig
RCS Subcooling	275°F	278°F		
CTMT Emergency Sump Levels	0 inches	0 inches		
CTMT Radiation	< 1 R / Hr	< 1 R / Hr		

Parameter	RCS Loop 2A	RCS Loop 2B	RCS Loop 2C
SG NR Level (all channels)	20%	0%	20%
AFW flow	325 GPM	0 GPM	340 GPM
SG Pressure (all channels)	800 psig	25 psig	820 psig
RCS WR Cold Leg Temperature	420°F	265°F	425°F
RCP status	Off	Off	Off

FARLEY NUCLEAR PLANT

CRITICAL SAFETY FUNCTION PROCEDURE

FNP-2-CSF-0

CRITICAL SAFETY FUNCTION STATUS TREES

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	**ALL**
Reference Use	
Information Use	

SAFETY RELATED

Approved:

Jim L. Hunter (for)
Operations Manager

Date Issued: 09/14/07

Table of Contents

KEY

A. **Purpose**

This procedure provides actions required to evaluate the status of the Critical Safety Functions.

B. **Symptoms or Entry Conditions**

I. This procedure is entered when monitoring of the Critical Safety Functions is required from FNP-2-EEP-0, REACTOR TRIP OR SAFETY INJECTION, step 23.

II. This procedure is entered when the operator transfers from the guidance of FNP-2-EEP-0, REACTOR TRIP OR SAFETY INJECTION to any other recovery guideline.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**Check at least one control room IPC SPDS console - Operable.**	
1.1	Verify no HOST LINK DOWN message on the IPC title bar.	1.1 Proceed to step 3.2.
___ 2	**Check SPDS TOP LEVEL page.**	
2.1	Click SPDS button on top toolbar.	

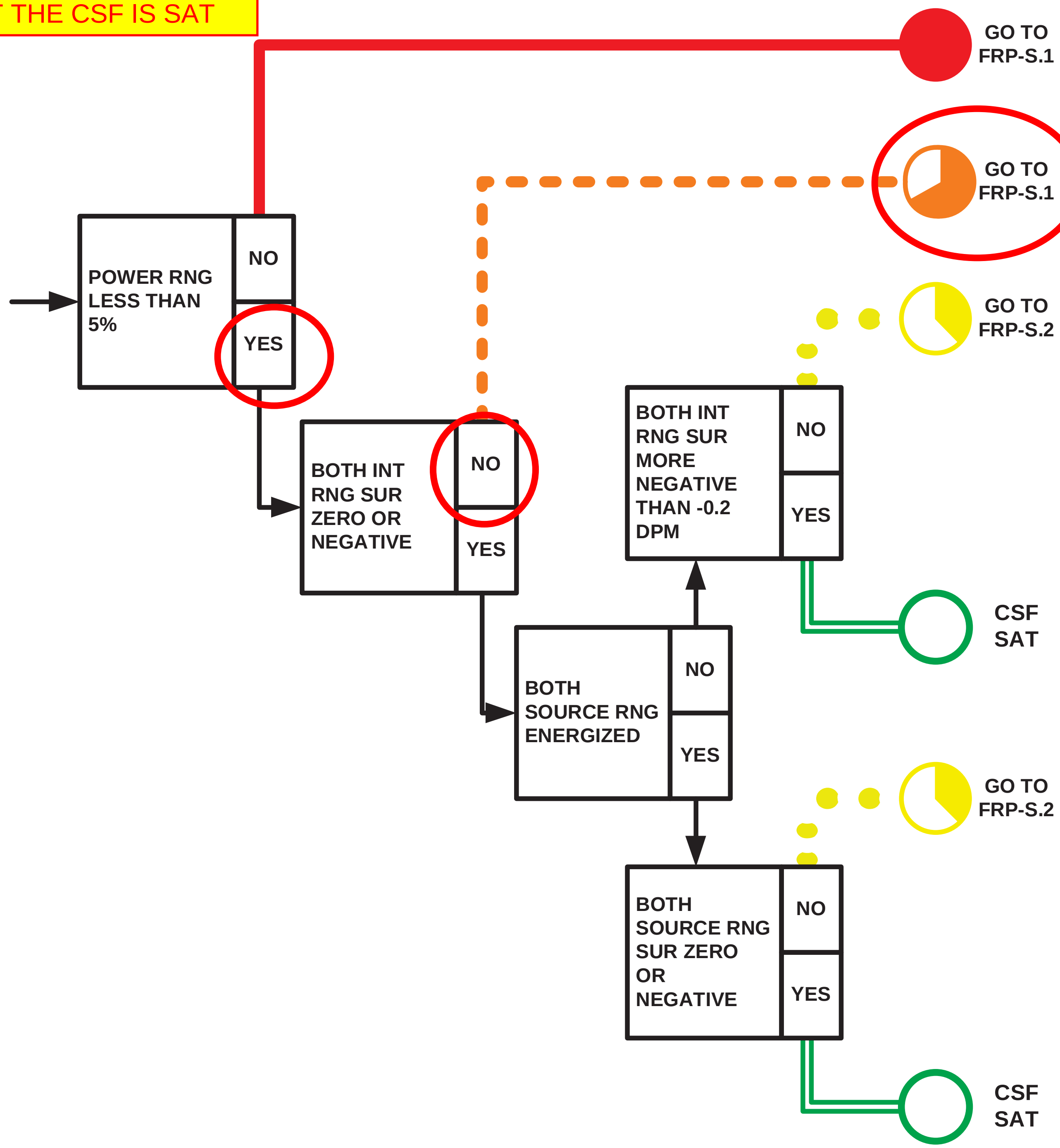
NOTE: Suspect critical safety functions are indicated by the color magenta.

Step	Action/Expected Response	Response NOT Obtained
2.2	Verify no Critical Safety Functions - SUSPECT. [] Subcriticality [] Core Cooling [] Heat Sink [] Integrity [] Containment [] Inventory	2.2 Monitor Critical Safety Function which is SUSPECT using FNP-2-CSF-0.1 through FNP-2-CSF-0.6 as appropriate.
___ 3	**Monitor Critical Safety Functions.**	
3.1	Monitor Critical Safety Functions with SPDS Application on IPC. <u>OR</u>	
3.2	Monitor Critical Safety Functions using FNP-2-CSF-0.1 through FNP-2-CSF-0.6	

-END-

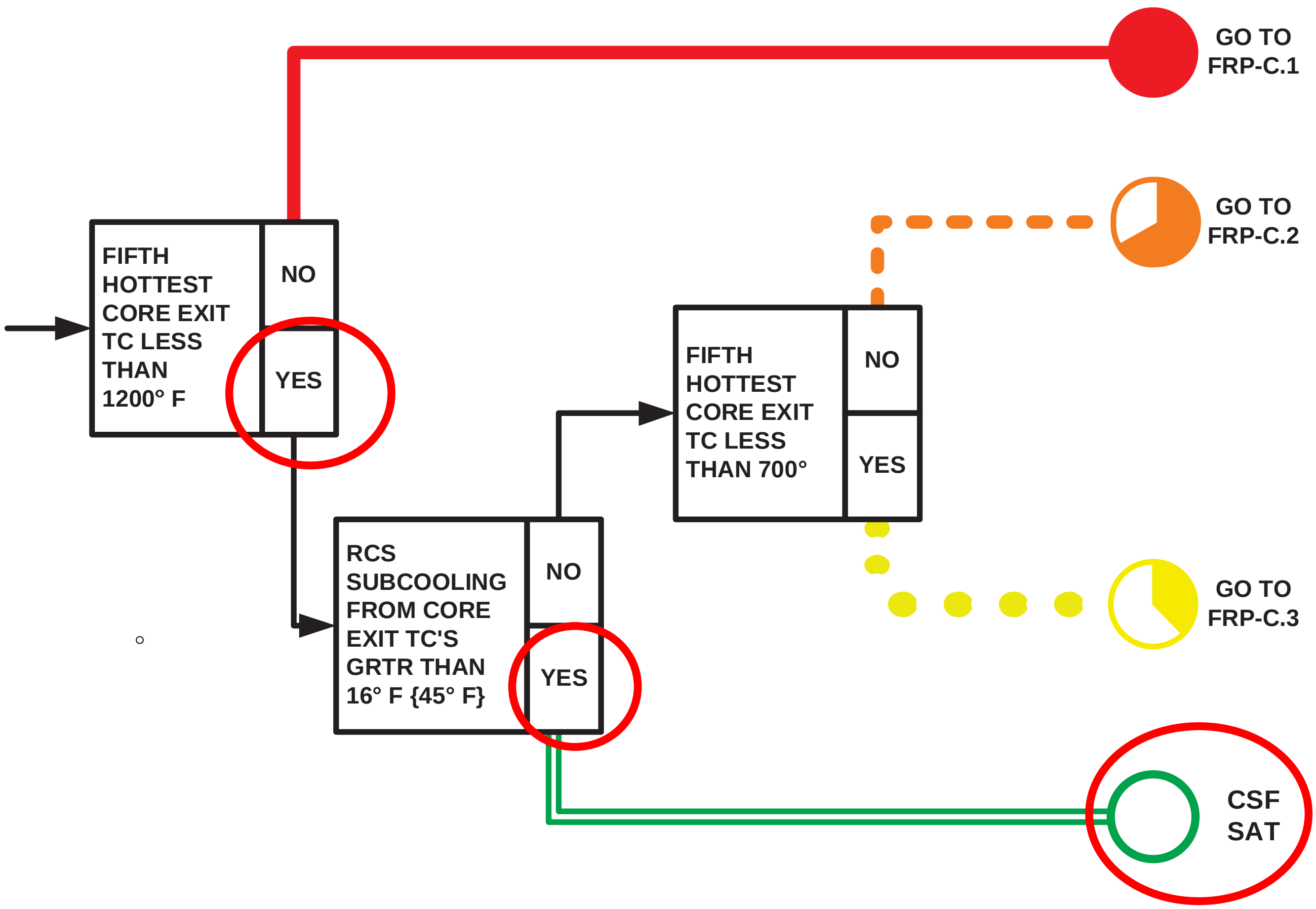
8/29/2007 08:33 **FNP-2-CSF-0.1**	**SUBCRITICALITY**	**Revision 12**

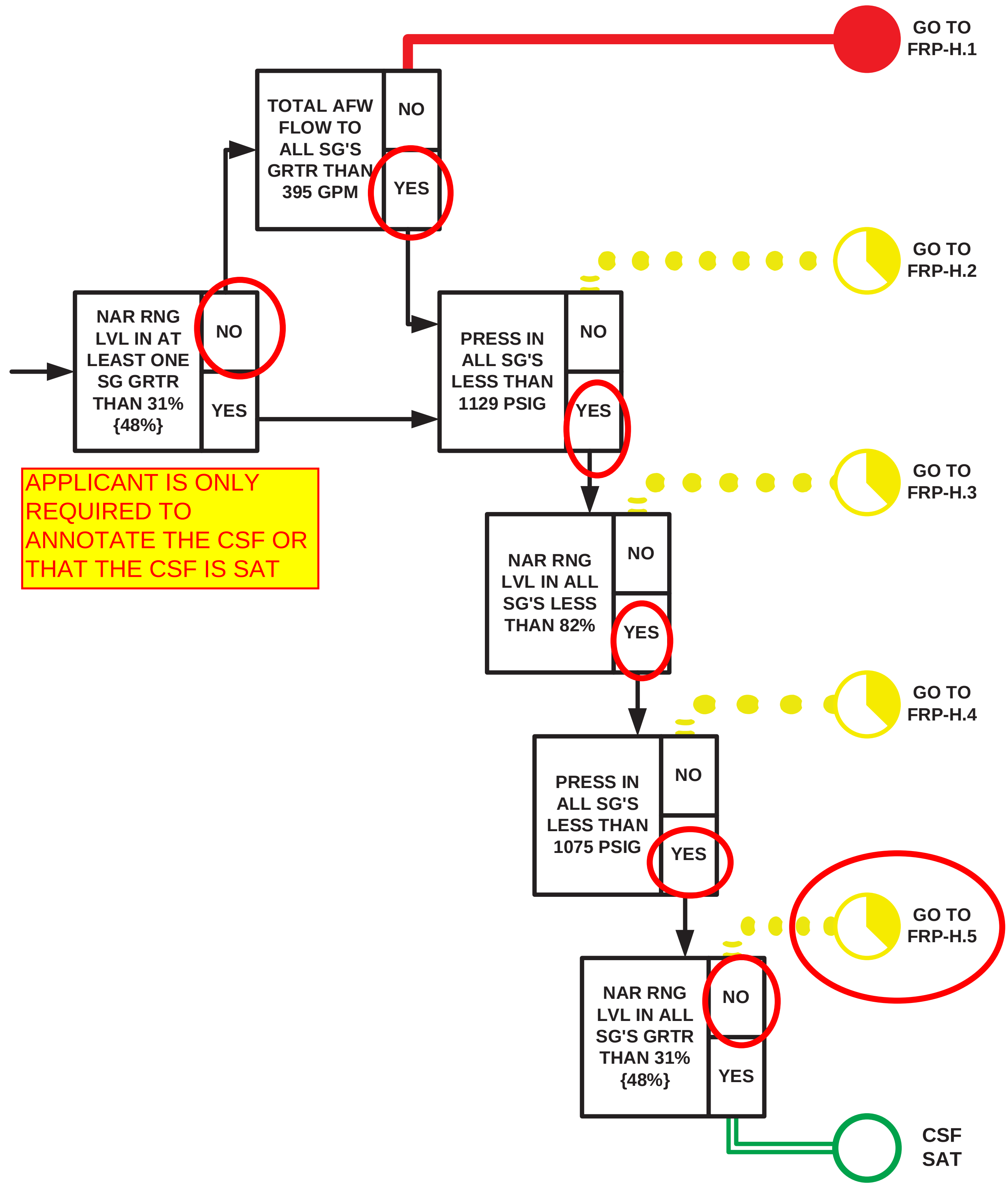
APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT



8/29/2007 08:33 **FNP-2-CSF-0.2**	**CORE COOLING**	**Revision 12**

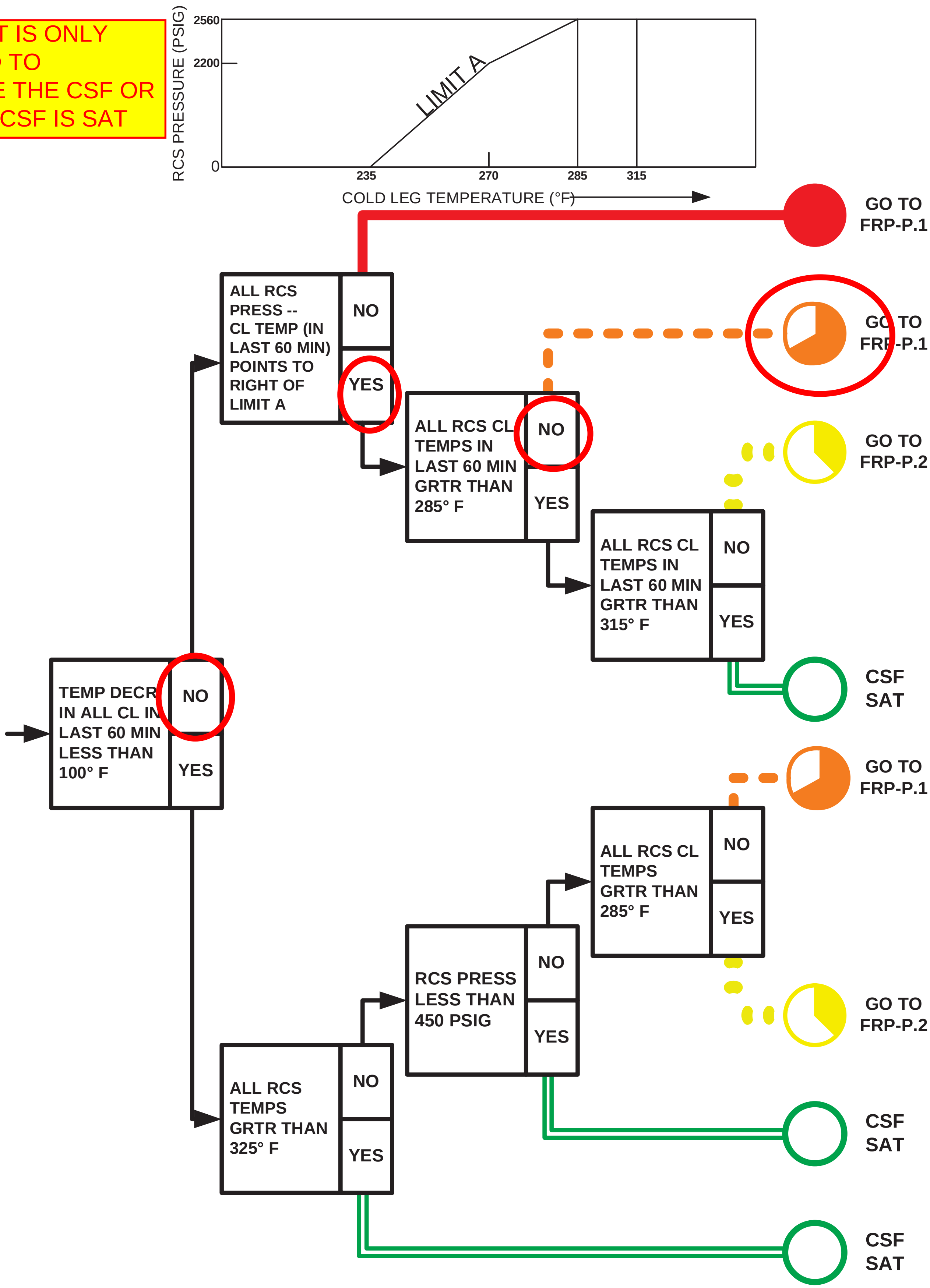
APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT




GO TO FRP-H.1
TOTAL AFW FLOW TO ALL SG'S GRTR THAN 395 GPM
NO
YES
GO TO FRP-H.2
NAR RNG LVL IN AT LEAST ONE SG GRTR THAN 31% {48%}
NO
YES
PRESS IN ALL SG'S LESS THAN 1129 PSIG
NO
YES
APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT
GO TO FRP-H.3
NAR RNG LVL IN ALL SG'S LESS THAN 82%
NO
YES
GO TO FRP-H.4
PRESS IN ALL SG'S LESS THAN 1075 PSIG
NO
YES
GO TO FRP-H.5
NAR RNG LVL IN ALL SG'S GRTR THAN 31% {48%}
NO
YES
CSF SAT

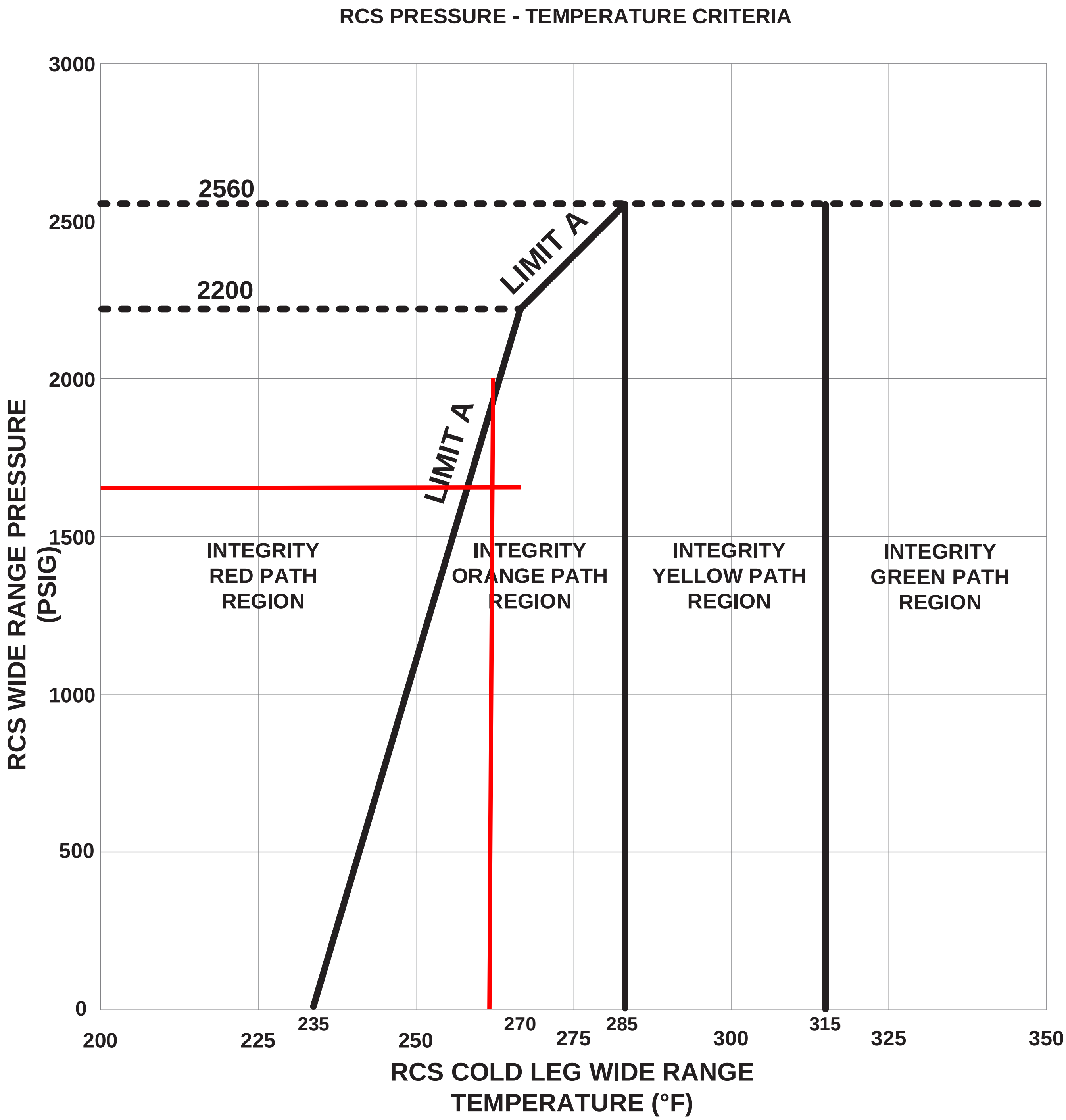
8/29/2007 08:33 **FNP-2-CSF-0.4**	**INTEGRITY**	**Revision 12**

APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT

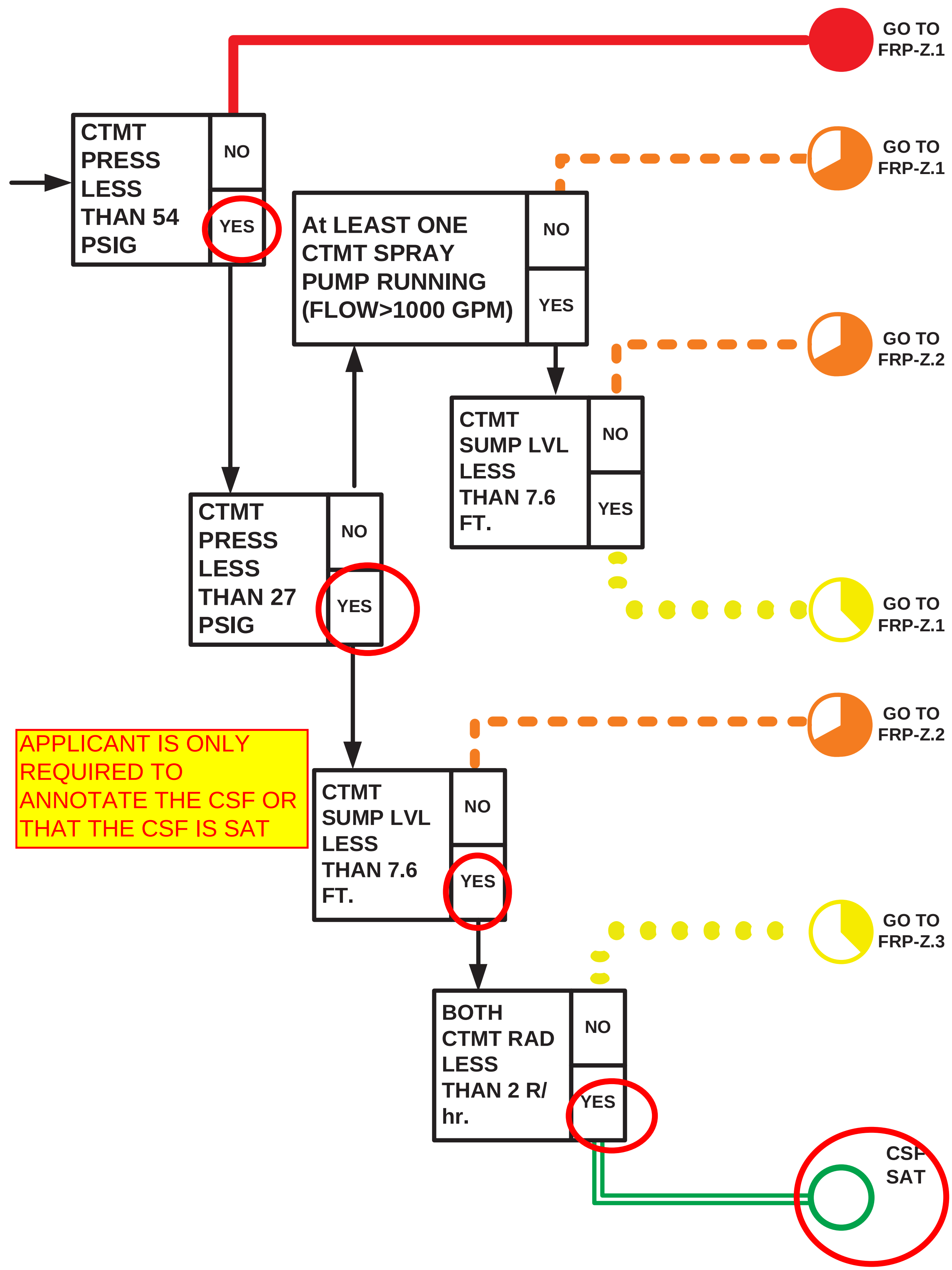


APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT

INTEGRITY



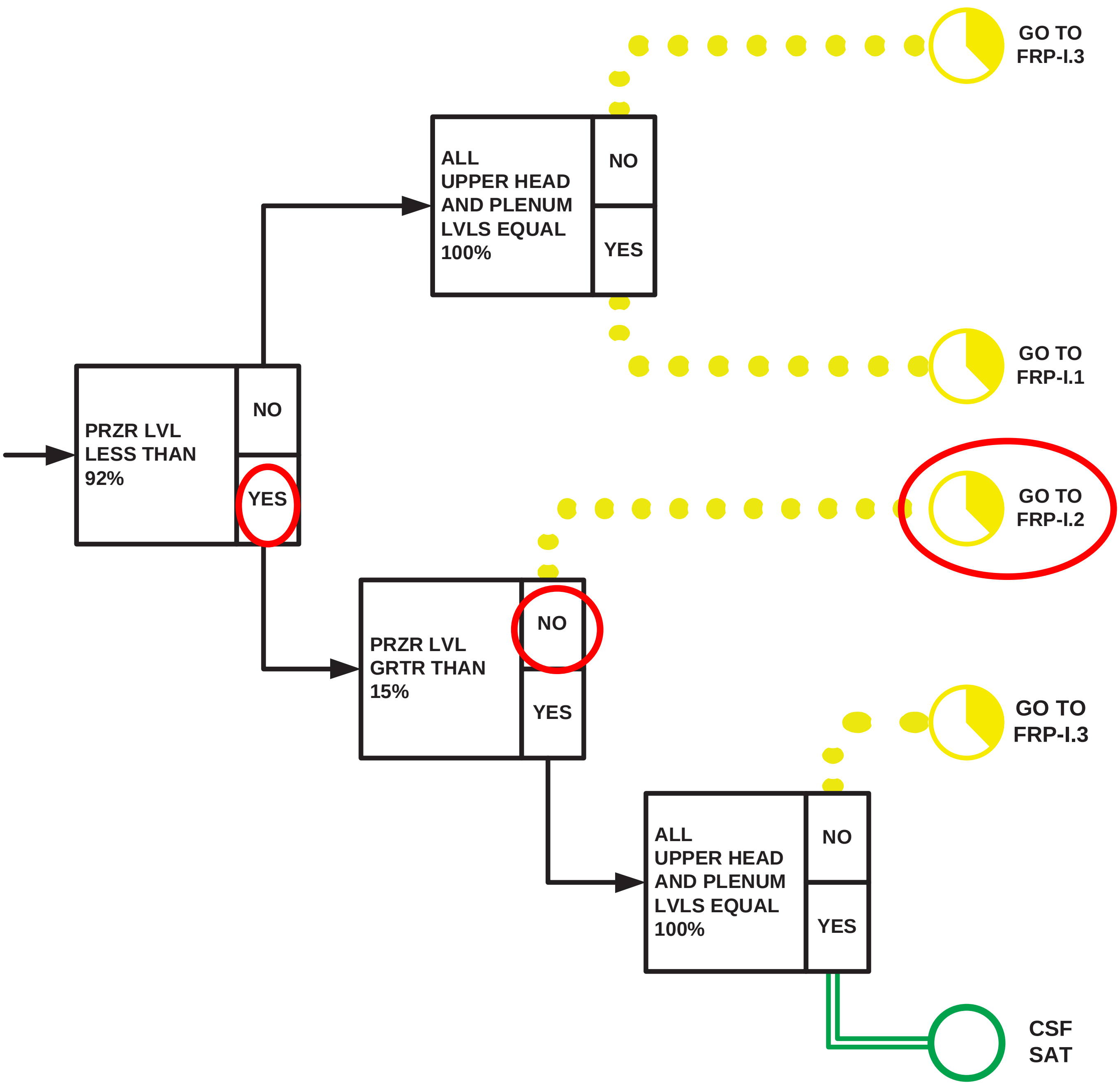
KEY


CTMT PRESS LESS THAN 54 PSIG
NO
YES
GO TO FRP-Z.1
At LEAST ONE CTMT SPRAY PUMP RUNNING (FLOW>1000 GPM)
NO
YES
GO TO FRP-Z.1
CTMT SUMP LVL LESS THAN 7.6 FT.
NO
YES
GO TO FRP-Z.2
CTMT PRESS LESS THAN 27 PSIG
NO
YES
GO TO FRP-Z.1
CTMT SUMP LVL LESS THAN 7.6 FT.
NO
YES
GO TO FRP-Z.2
APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT
BOTH CTMT RAD LESS THAN 2 R/ hr.
NO
YES
GO TO FRP-Z.3
CSF SAT

KEY

8/29/2007 08:33 FNP-2-CSF-0.6	INVENTORY	Revision 12

APPLICANT IS ONLY REQUIRED TO ANNOTATE THE CSF OR THAT THE CSF IS SAT



FARLEY NUCLEAR PLANT

CRITICAL SAFETY FUNCTION PROCEDURE

FNP-2-CSF-0

CRITICAL SAFETY FUNCTION STATUS TREES

PROCEDURE USAGE REQUIREMENTS-per FNP-0-AP-6	SECTIONS
Continuous Use	**ALL**
Reference Use	
Information Use	

SAFETY RELATED

Approved:

Jim L. Hunter (for)
Operations Manager

Date Issued: 09/14/07

Table of Contents

A. Purpose

This procedure provides actions required to evaluate the status of the Critical Safety Functions.

B. Symptoms or Entry Conditions

I. This procedure is entered when monitoring of the Critical Safety Functions is required from FNP-2-EEP-0, REACTOR TRIP OR SAFETY INJECTION, step 23.

II. This procedure is entered when the operator transfers from the guidance of FNP-2-EEP-0, REACTOR TRIP OR SAFETY INJECTION to any other recovery guideline.

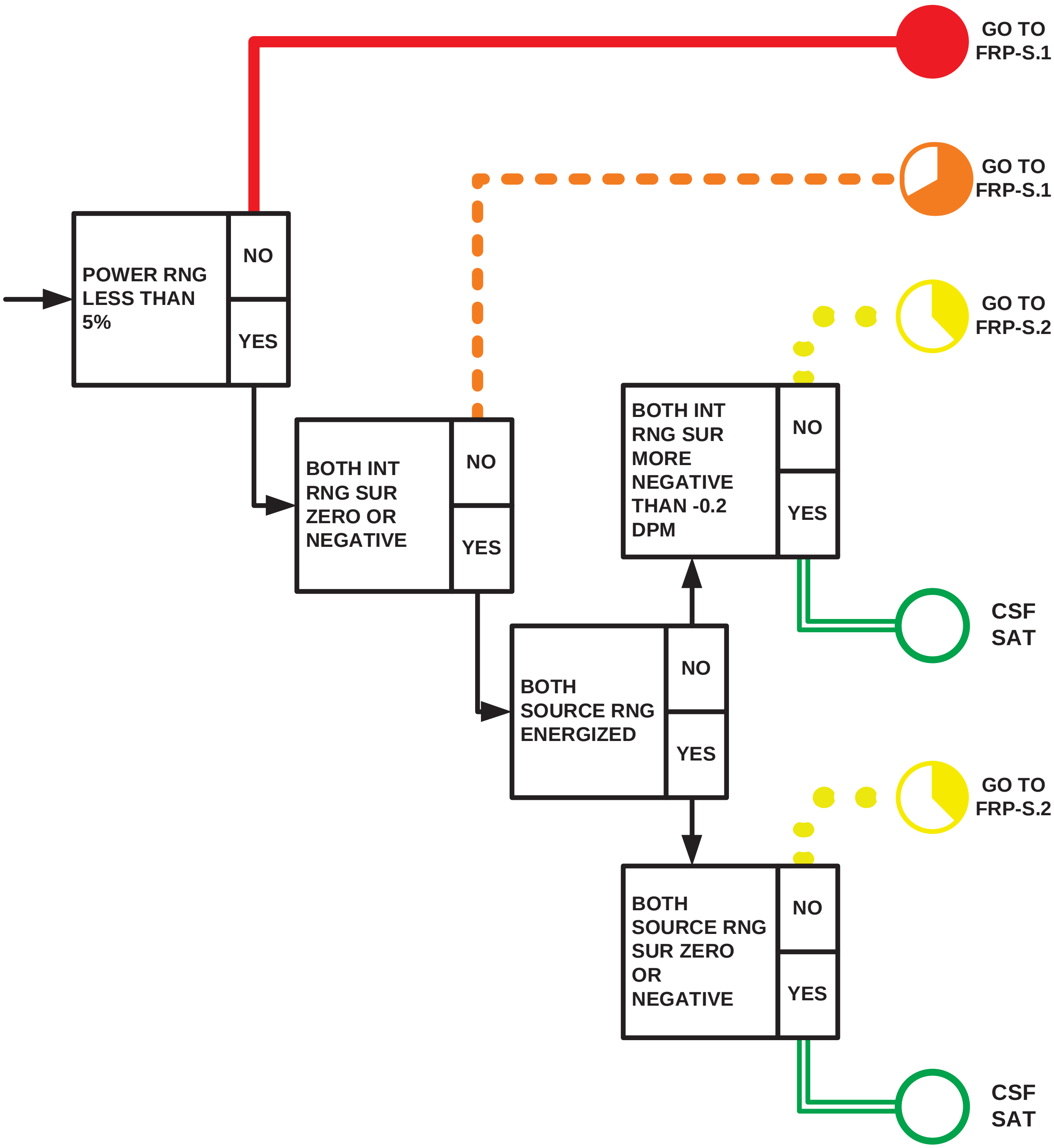
Step	Action/Expected Response	Response NOT Obtained
___ 1	**Check at least one control room IPC SPDS console - Operable.**	
1.1	Verify no HOST LINK DOWN message on the IPC title bar.	1.1 Proceed to step 3.2.
___ 2	**Check SPDS TOP LEVEL page.**	
2.1	Click SPDS button on top toolbar.	

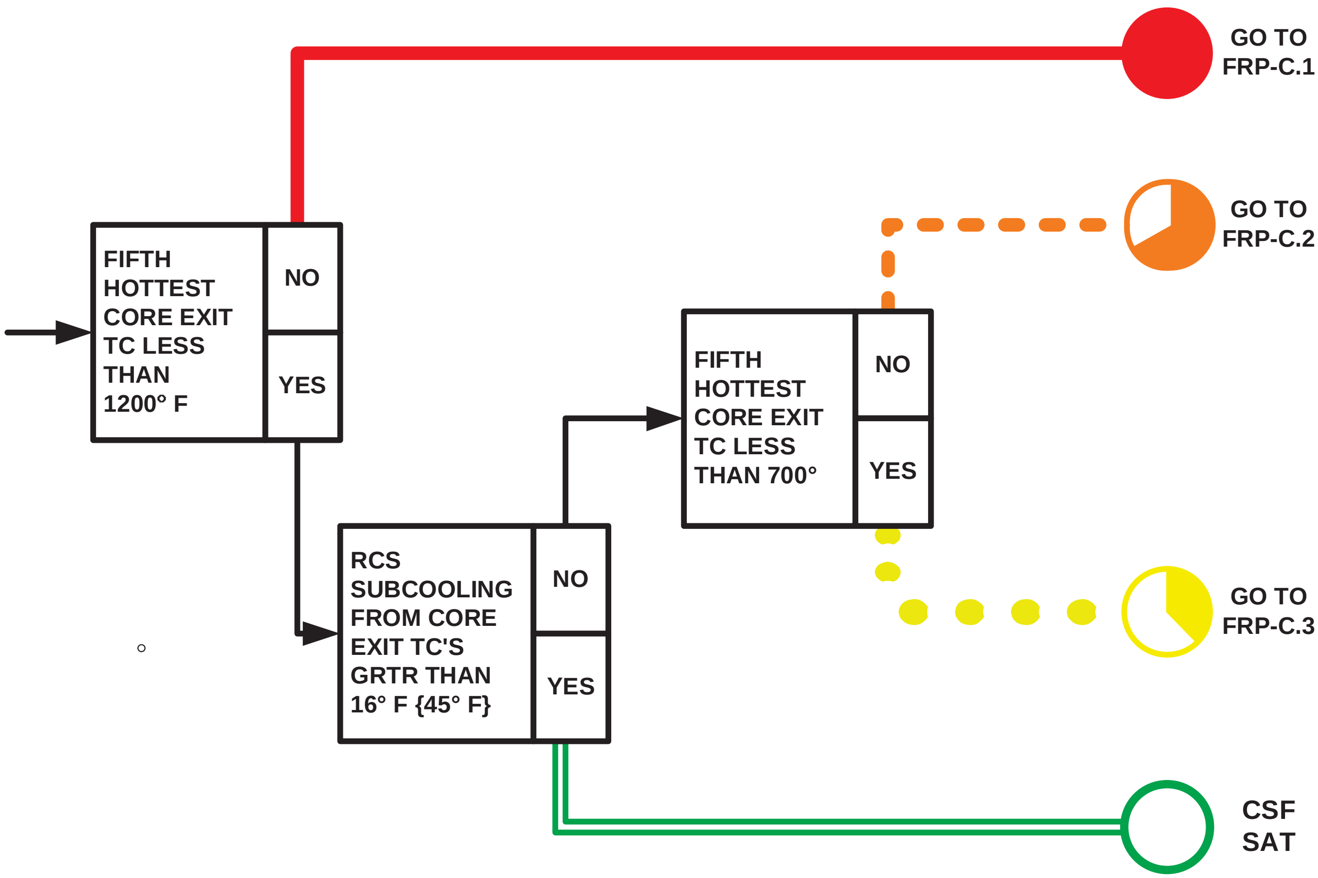
NOTE: Suspect critical safety functions are indicated by the color magenta.

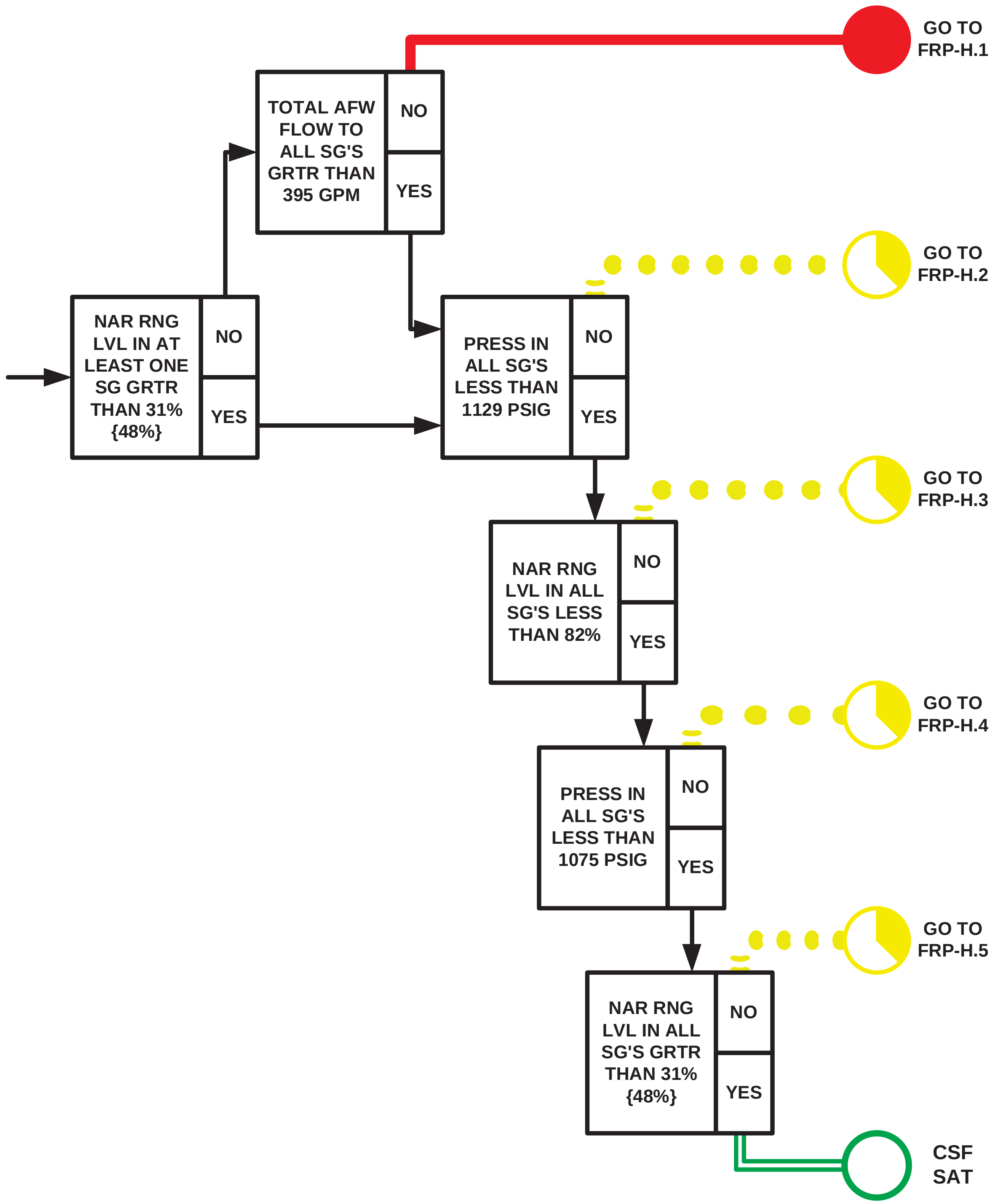
Step	Action/Expected Response	Response NOT Obtained
2.2	Verify no Critical Safety Functions - SUSPECT. [] Subcriticality [] Core Cooling [] Heat Sink [] Integrity [] Containment [] Inventory	2.2 Monitor Critical Safety Function which is SUSPECT using FNP-2-CSF-0.1 through FNP-2-CSF-0.6 as appropriate.
___ 3	**Monitor Critical Safety Functions.**	
3.1	Monitor Critical Safety Functions with SPDS Application on IPC. OR	
3.2	Monitor Critical Safety Functions using FNP-2-CSF-0.1 through FNP-2-CSF-0.6	

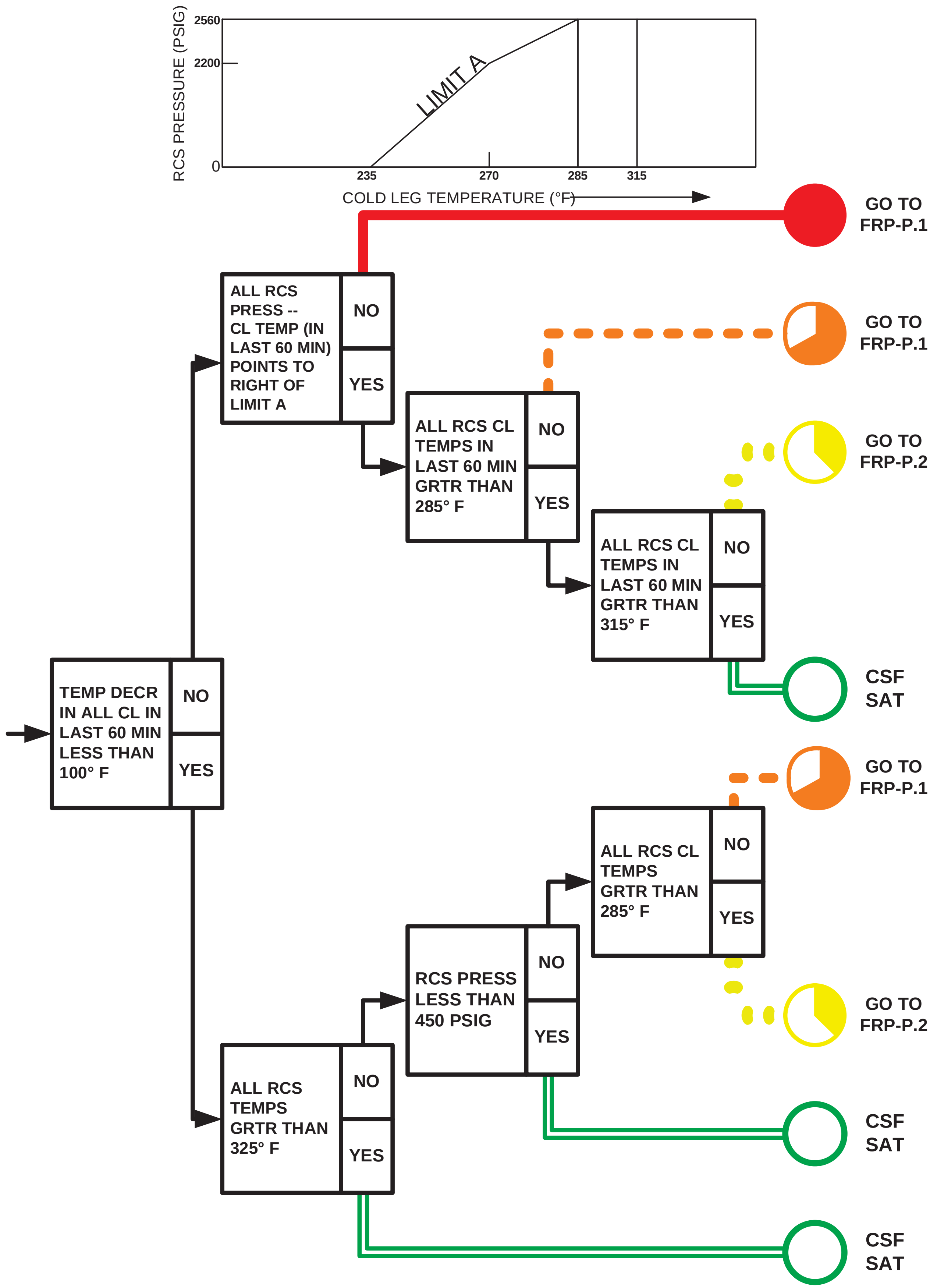
-END-

8/29/2007 08:33 FNP-2-CSF-0.1	SUBCRITICALITY	Revision 12



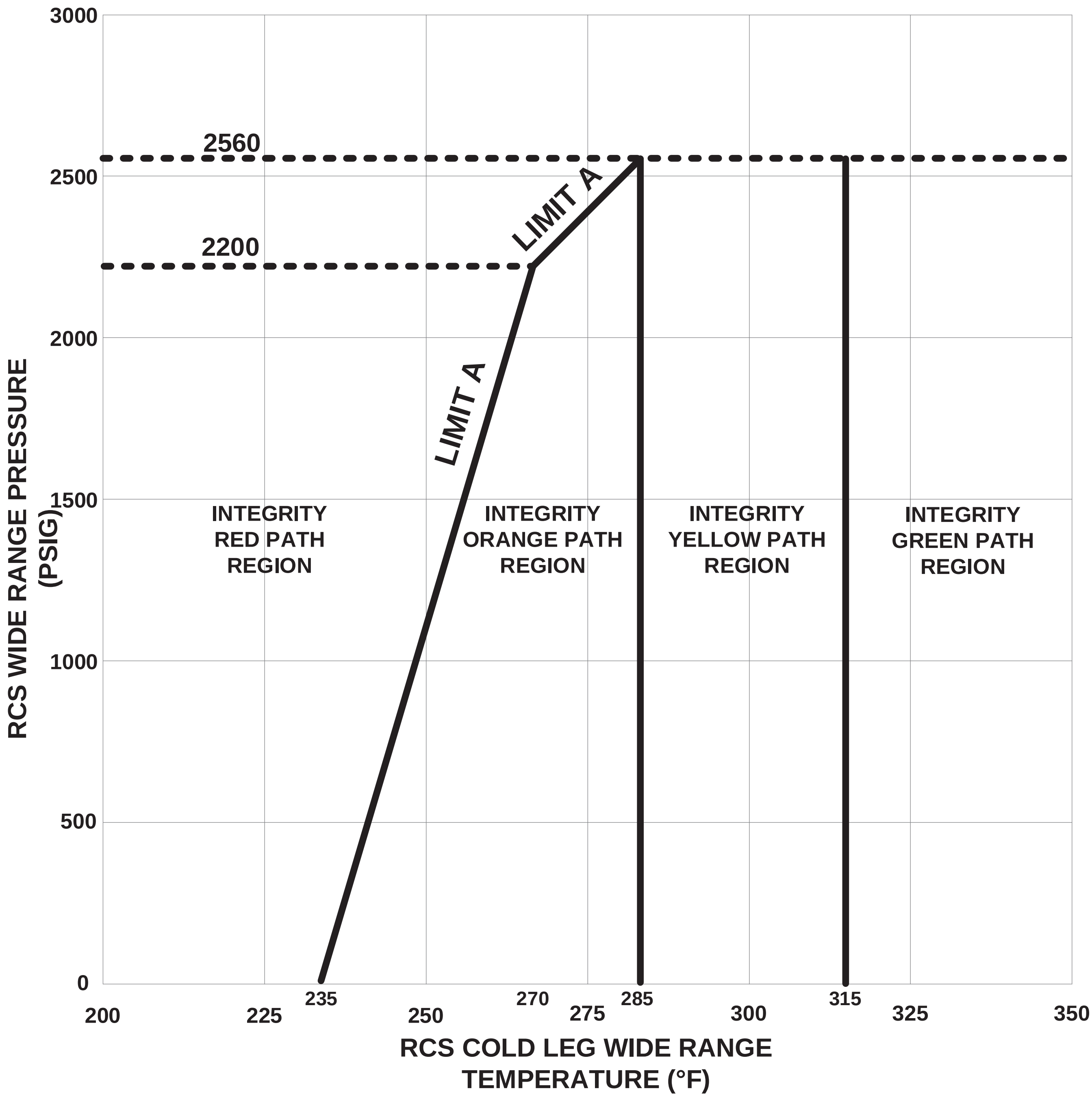

FIFTH HOTTEST CORE EXIT TC LESS THAN 1200° F
NO
YES
GO TO FRP-C.1
RCS SUBCOOLING FROM CORE EXIT TC'S GRTR THAN 16° F {45° F}
NO
YES
FIFTH HOTTEST CORE EXIT TC LESS THAN 700°
NO
YES
GO TO FRP-C.2
GO TO FRP-C.3
CSF SAT

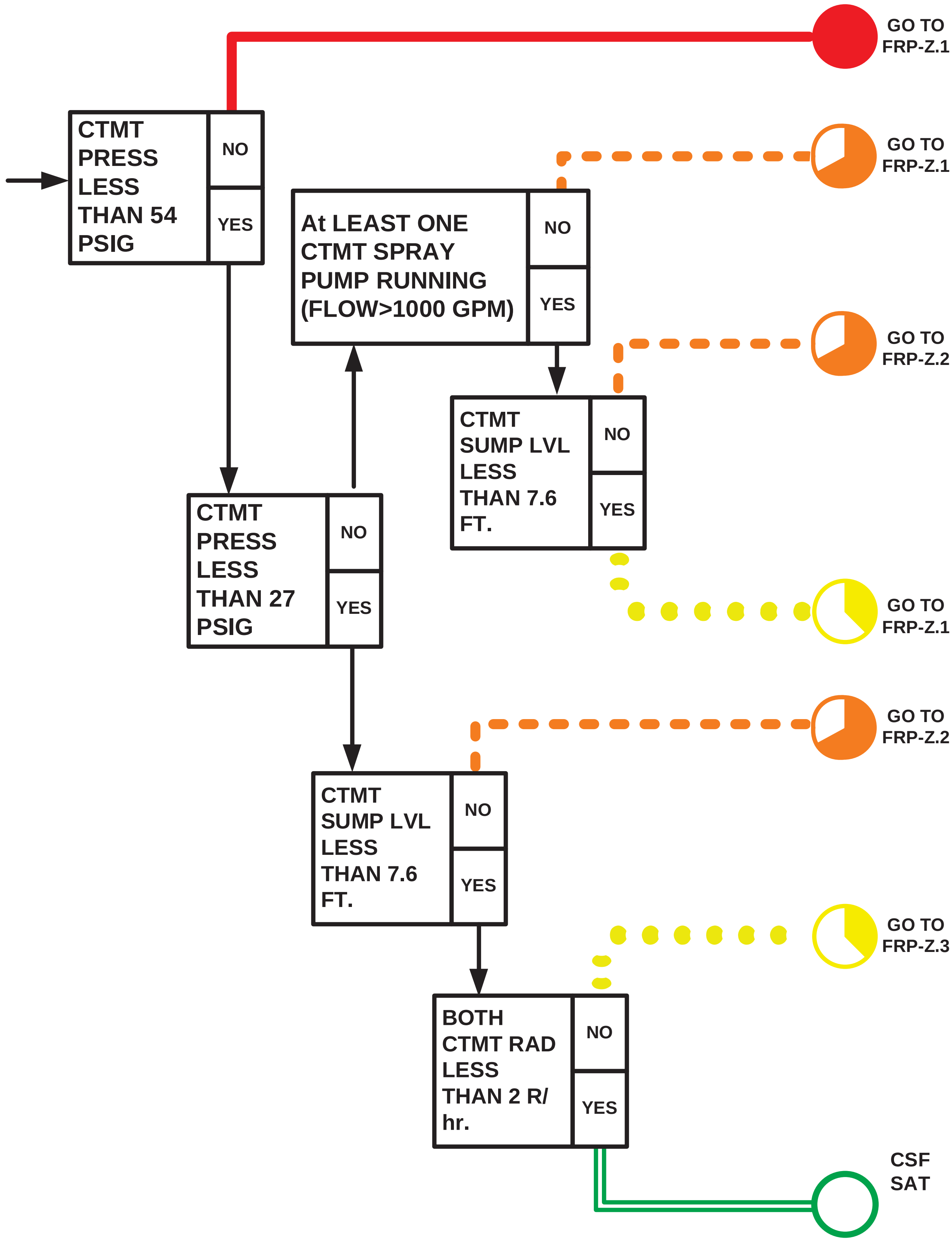

TOTAL AFW FLOW TO ALL SG'S GRTR THAN 395 GPM
NO
YES
GO TO FRP-H.1
NAR RNG LVL IN AT LEAST ONE SG GRTR THAN 31% {48%}
NO
YES
PRESS IN ALL SG'S LESS THAN 1129 PSIG
NO
YES
GO TO FRP-H.2
NAR RNG LVL IN ALL SG'S LESS THAN 82%
NO
YES
GO TO FRP-H.3
PRESS IN ALL SG'S LESS THAN 1075 PSIG
NO
YES
GO TO FRP-H.4
NAR RNG LVL IN ALL SG'S GRTR THAN 31% {48%}
NO
YES
GO TO FRP-H.5
CSF SAT

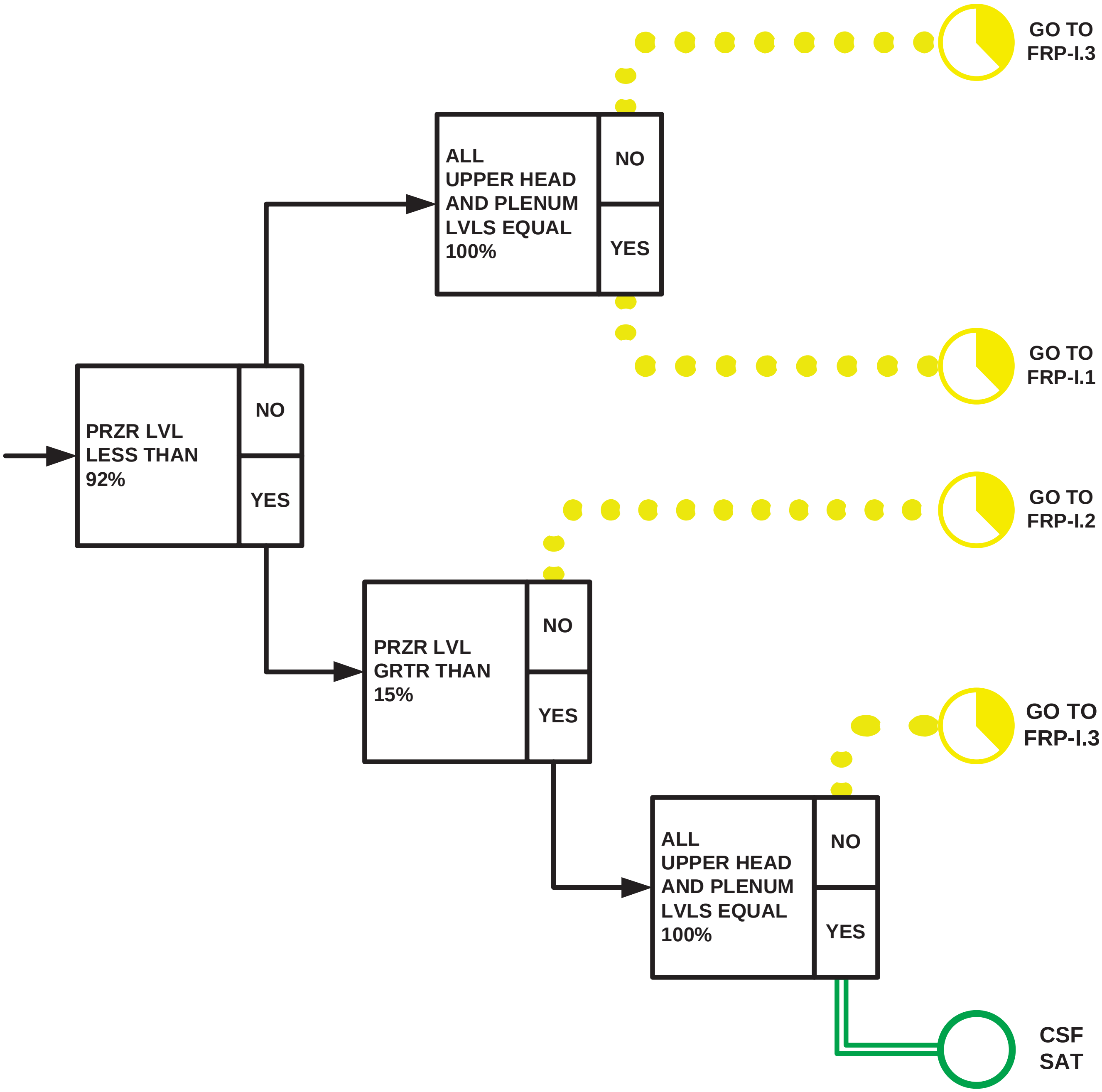

RCS PRESSURE (PSIG)
2560
2200
0
LIMIT A
235
270
285
315
COLD LEG TEMPERATURE (°F)
GO TO FRP-P.1
ALL RCS PRESS -- CL TEMP (IN LAST 60 MIN) POINTS TO RIGHT OF LIMIT A
NO
YES
GO TO FRP-P.1
ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 285° F
NO
YES
GO TO FRP-P.2
ALL RCS CL TEMPS IN LAST 60 MIN GRTR THAN 315° F
NO
YES
CSF SAT
TEMP DECR IN ALL CL IN LAST 60 MIN LESS THAN 100° F
NO
YES
GO TO FRP-P.1
ALL RCS CL TEMPS GRTR THAN 285° F
NO
YES
RCS PRESS LESS THAN 450 PSIG
NO
YES
GO TO FRP-P.2
ALL RCS TEMPS GRTR THAN 325° F
NO
YES
CSF SAT
CSF SAT

INTEGRITY

RCS PRESSURE - TEMPERATURE CRITERIA




GO TO FRP-Z.1
CTMT PRESS LESS THAN 54 PSIG
NO
YES
GO TO FRP-Z.1
At LEAST ONE CTMT SPRAY PUMP RUNNING (FLOW>1000 GPM)
NO
YES
GO TO FRP-Z.2
CTMT SUMP LVL LESS THAN 7.6 FT.
NO
YES
CTMT PRESS LESS THAN 27 PSIG
NO
YES
GO TO FRP-Z.1
GO TO FRP-Z.2
CTMT SUMP LVL LESS THAN 7.6 FT.
NO
YES
GO TO FRP-Z.3
BOTH CTMT RAD LESS THAN 2 R/ hr.
NO
YES
CSF SAT


PRZR LVL
LESS THAN
92%
NO
YES
ALL
UPPER HEAD
AND PLENUM
LVLS EQUAL
100%
NO
YES
GO TO
FRP-I.3
GO TO
FRP-I.1
PRZR LVL
GRTR THAN
15%
NO
YES
GO TO
FRP-I.2
ALL
UPPER HEAD
AND PLENUM
LVLS EQUAL
100%
NO
YES
GO TO
FRP-I.3
CSF
SAT

A.1.b. RO

TITLE: Determine maximum RHR flowrate and time to saturation for a loss of RHR event.
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 20 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☐ TIME CRITICAL ☐ PRA

JPM DIRECTIONS:

1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Requiring the examinee to acquire the required materials may or may not be included as part of the JPM.

TASK STANDARD: Upon successful completion of this JPM, the examinee will:

- Correctly assess and determine the maximum RHR flowrate for the current RCS level.
- Correctly assess and determine the time to core boiling for the current core conditions.

Examinee:		
Overall JPM Performance:	**Satisfactory** ☐	**Unsatisfactory** ☐
Evaluator Comments (attach additional sheets if necessary)		

EXAMINER: ______________________

Developer	S. Jackson	Date: 4/2/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to DETERMINE MAXIMUM RHR FLOWRATE AND TIME TO SATURATION FOR A LOSS OF RHR EVENT. The conditions under which this task is to be performed are:

a. The Unit 1 Reactor has been shutdown for 350 hours.
b. Refueling is complete, with 53 new fuel assemblies loaded into the core.
c. An RCS leak had occurred, but it has now been isolated.
d. 1A RHR pump is the only RHR pump running.
e. RHR flow was lowered during leak isolation and is now at 1300 gpm.
f. RCS level has been restored to 122' 8.5" and is stable.
g. Current RCS temperature is 116°F.
h. A current Shutdown Safety Assessment is not available.

Your task is to perform the following per AOP-12.0:

1) Determine the maximum allowable RHR flowrate.
2) Determine the time to core saturation for a loss of RHR.

INITIATING CUE: IF you have no questions, you may begin.

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
_____ START TIME		
* 1. Evaluate Figure 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing to determine maximum allowable RHR flowrate.	1) Step 7 of AOP-12.0, Maintain RCS level to within the Acceptable Operating Region of Figure 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing for the existing RHR flow. RCS level is 122' 8.5". Determines that maximum RHR flow is ≤ 1750 gpm. Allowable tolerance: ≤ 1600 -1800 gpm.	S / U

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
2. Determine time to core saturation, determine appropriate table of ATTACHMENT 3, TABLE A or TABLE B.	2) ATTACHMENT 3, step 1.1. Determines that Attachment 3, TABLE B is required per ATTACHMENT 3, step 1.1.2, Time to saturation with one third of the spent fuel replaced with new fuel.	S / U
3. Determine time to core saturation, determine appropriate table of ATTACHMENT 3 based on initial RCS temperature : Table for 100°F Table for 120°F Table for 140°F	3) ATTACHMENT 3, step 1.3. Determines that page from Attachment 3, TABLE B for ASSUMED INITIAL TEMPERATURE = 120°F is required.	S / U
4. Determine time to core saturation, determine appropriate column of ATTACHMENT 3, TABLE B , ASSUMED INITIAL TEMPERATURE = 120°F : Time to Saturation at midloop (mins) Time to Saturation 3' below flange (mins) Time to Saturation full Rx cavity (hours)	4) ATTACHMENT 3, step 1.2. Determines that page from Attachment 3, TABLE B for ASSUMED INITIAL TEMPERATURE = 120°F , column for **Time to Saturation at midloop (mins) is required.**	S / U

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
* 5. Determine time to core saturation.	5) Determines that Time After Shutdown (hours) is 350 hours and minutes to boiling is calculated to be 21.35 minutes. 300 hours = 20.2 minutes 400 hours = 22.5 minutes 20.2 + 22.5 = 42.7 42.7/2 = 21.35 minutes After rounding, 21.4 minutes is acceptable. Allowable tolerance: 21.3 -21.4 minutes. **<u>Since the Time After Shutdown chart only shows 300 hours and 400 hours, the candidate may conservatively take the 300 hours after shutdown for time to boil of 20.2 minutes or 20 minutes for rounding. This is acceptable</u>**	S / U

_____ **STOP TIME**

Terminate when all elements of the task have been completed.

<u>CRITICAL ELEMENTS</u>: Critical Elements are denoted with an asterisk (∗) before the element number.

GENERAL REFERENCES:

1. FNP-1-AOP-12.0, v25
2. G2.1.25 – 3.9 / 4.2

GENERAL TOOLS AND EQUIPMENT:

1. FNP-1-AOP-12.0, v25
2. Calculator, ruler or straight edge if requested

Critical ELEMENT justification:

STEP	Evaluation
1.	**Critical:** Task completion: required to properly determine Maximum RHR flowrate.
2-4	Not Critical: Theses step do not have to be performed if the applicant can correctly complete element 5.
5.	**Critical:** Task completion: required to properly determine time to core saturation.

COMMENTS:

CONDITIONS

When I tell you to begin, you are to DETERMINE MAXIMUM RHR FLOWRATE AND TIME TO SATURATION FOR A LOSS OF RHR EVENT. The conditions under which this task is to be performed are:

a. The Unit 1 Reactor has been shutdown for 350 hours.
b. Refueling is complete, with 53 new fuel assemblies loaded into the core.
c. An RCS leak had occurred, but it has now been isolated.
d. 1A RHR pump is the only RHR pump running.
e. RHR flow was lowered during leak isolation and is now at 1300 gpm.
f. RCS level has been restored to 122’ 8.5” and is stable.
g. Current RCS temperature is 116°F.
h. A current Shutdown Safety Assessment is not available.

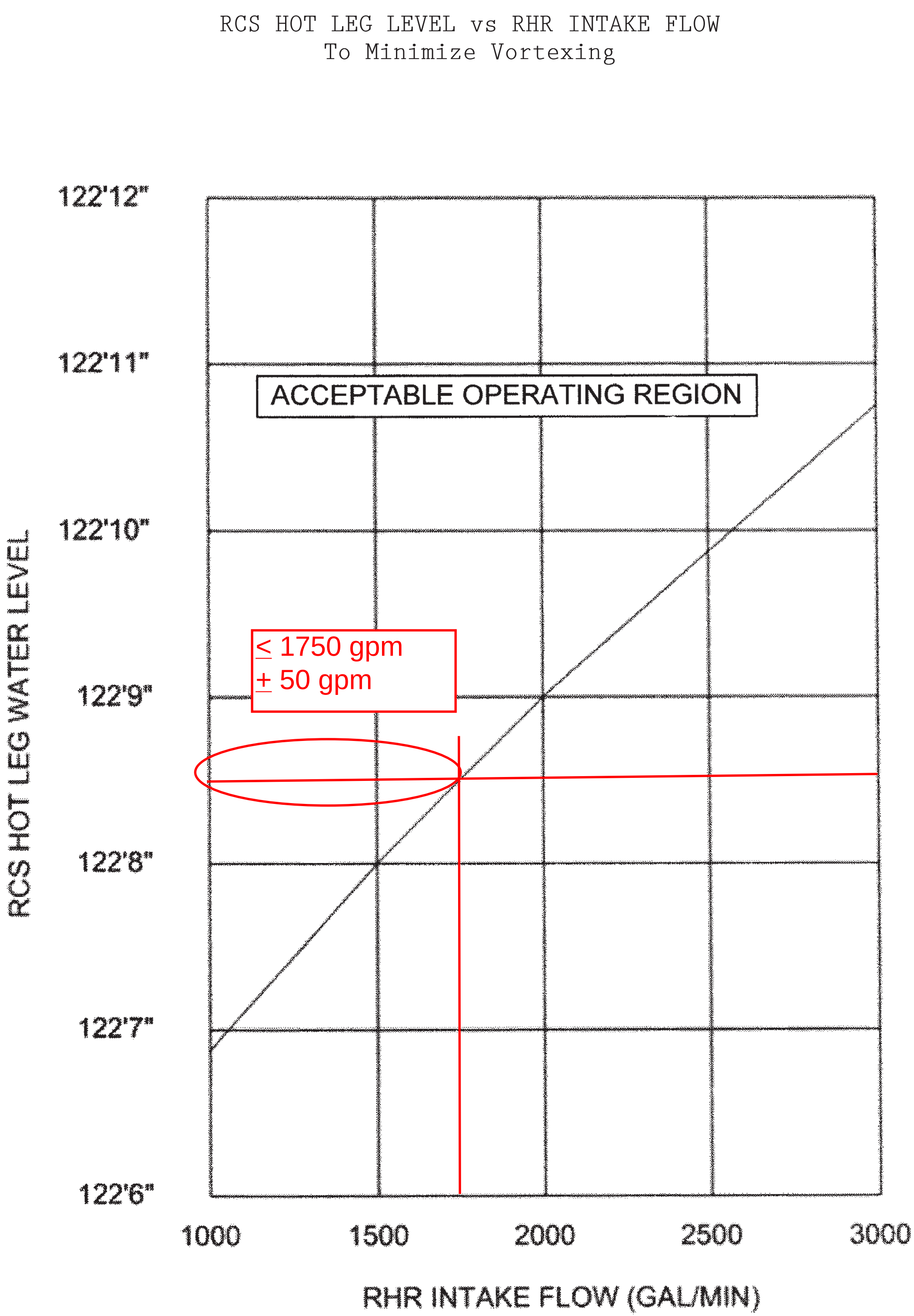
Your task is to perform the following per AOP-12.0:

1) Determine the maximum allowable RHR flowrate.
2) Determine the time to core saturation for a loss of RHR.

AOP-12	
Maximum allowable RHR flowrate	
Time to Core Saturation	

FIGURE 1

RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing



ATTACHMENT 3

Time to Core Saturation

1 **Time to Core Saturation:**

1.1 Tables A and B provide estimates of the time to core boiling following a loss RHR capability for two cases:

1.1.1 **TABLE A** provides a Time to Saturation as a function of time after shutdown for a full core immediately after shutdown for a refueling.

1.1.2 **TABLE B** provides a Time to Saturation as a function of time after shutdown for a core in which one third of the spent fuel has been replaced with new fuel.

1.2 Both cases are evaluated for conditions when RCS level is at mid loop (122'9"), at three feet below the reactor flange (126'7"), and when the reactor cavity is full.

1.3 Both cases are also evaluated for three assumed initial temperatures: 100°F, 120°F, and 140°F.

1.4 These figures can be used to estimate the amount of time available for operator action to restore RHR before additional protective measures must be taken.

ATTACHMENT 3

Time to Core Saturation

TABLE B---POWER UPRATED UNIT

TIME TO SATURATION: ONE THIRD NEW FUEL

ASSUMED INITIAL TEMPERATURE=120°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
100	12.8	17.5	9.2
200	17.1	23.4	12.4
300	20.2	[illegible]	14.6
400	22.5	[illegible]	16.3
500	25.4	[illegible]	18.4
600	28.3	[illegible]	20.5
700	30.5	41.7	22.1
800	33.0	45.2	23.9

If using interpolation - 21.3 - 21.4 min.
May use 20 minutes since 20.2 rounds to 20 and 0.2 minutes is 12 sec

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

FNP-1-AOP-12.0
1-02-2013
Revision 25.0

FARLEY NUCLEAR PLANT

ABNORMAL OPERATING PROCEDURE

FNP-1-AOP-12.0

RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION

SAFETY RELATED

PROCEDURE USAGE REQUIREMENTS per NMP-AP-003	SECTIONS
Continuous Use	ALL
Reference Use	
Information Use	

Approved:

David L Reed (for)

Operations Manager

Date Issued: 01/28/13

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A. Purpose

This procedure provides actions for response to a malfunction of the RHR system.

Actions in this procedure for restoring RHR PUMPs assume electrical power is available. During loss of electrical power conditions, FNP-1-AOP-5.0, LOSS OF A OR B TRAIN ELECTRICAL POWER, provides actions for restoration of electrical power which should be performed in addition to continuing with this procedure.

The first part of this procedure deals with the protection of any running RHR pump and isolation of any leakage. If a running train is maintained the procedure is exited. Credit may be taken for RCS Loops providing core cooling in place of a running train of RHR. The next portion deals with restoring a train of RHR while monitoring core temperatures. If a train cannot be restored actions are taken for protection of personnel, establishing containment closure, and provides alternate methods of decay heat removal while trying to restore a train of RHR. Alternate cooling methods include: establishing a secondary heat sink if steam generators are available; feed and bleed cooling and feed and spill cooling.

The intent of feed and bleed cooling is to regain pressurizer level and allow steaming through a bleed path to provide core cooling. This requires that the RCS be in a configuration that will allow a level in the pressurizer.

The intent of feed and spill cooling is to allow spillage from the RCS and locally throttle injection flow to provide core cooling. This method is used when the reactor vessel head is blocked or RCS loop openings exist.

This procedure is applicable in modes 4, 5 and 6.

Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS <u>AND</u> the RCS temperature is below 180°F.

B. Symptoms or Entry Conditions

1 This procedure is entered when a malfunction of the RHR system is indicated by any of the following:

1.1 Trip of any operating RHR pump

1.2 Excessive RHR system leakage

1.3 Evidence of running RHR pump cavitation

1.4 Closure of loop suction valve

1.5 High RCS or core exit T/C temperature

1.6 Procedure could be entered from various annunciator response procedures.

CF3 1A OR 1B RHR PUMP OVERLOAD TRIP

CF4 1A RHR HX OUTLET FLOW LO

CF5 1B RHR HX OUTLET FLOW LO

CG3 1A OR 1B RHR HX CCW DISCH FLOW HI

EA5 1A OR 1B RHR PUMP CAVITATION

EB5 MID-LOOP CORE EXIT TEMP HI

EC5 RCS LVL HI-LO

Step	Action/Expected Response	Response NOT Obtained

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CAUTION: Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS AND the RCS temperature is below 180°F.

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CAUTION: Filling the pressurizer to 100% will cause a loss of nozzle dams due to the head of water.

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NOTE: RCS to RHR loop suction valves will be deenergized if RCS TAVG is less than 180°F.

___ 1 **Check RHR loop suction valves - OPEN.**

RHR PUMP	1A	1B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSED(IF REQUIRED)	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3

1 Stop any RHR PUMP with closed loop suction valve(s).

1.1 IF required, THEN adjust charging flow to maintain RCS level.

___ 2 **IF the standby RHR train is NOT affected AND plant conditions permit operation, THEN place the standby RHR train in service per FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM.**

2 IF core cooling provided by the SGs, THEN proceed to step 8.

Step	Action/Expected Response	Response NOT Obtained

NOTE: Rapid flow adjustments may cause more severe pump cavitation.

Step	Action/Expected Response	Response NOT Obtained
___ **3**	**Check RHR PUMPs - NOT CAVITATING.** The following parameters should be stable and within normal ranges. [] RHR flow rate within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing. [] Discharge pressure [] Suction pressure [] RHR motor ammeter readings [] No unusual pump noise	3 Perform the following: 3.1 Slowly reduce RHR flow rate to eliminate cavitation. 3.2 <u>IF</u> cavitation CANNOT be eliminated, <u>THEN</u> stop the affected RHR pump(s).
___ **4**	**Check any RHR PUMP - RUNNING**	4 Proceed to step 13.
___ **5**	**Verify RHR flow > 3000 gpm.** 1A(1B) RHR HDR FLOW [] FI 605A [] FI 605B	5 Refer to Technical Specifications 3.9.4 and 3.9.5 for applicability.

Step	Action/Expected Response	Response NOT Obtained

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CAUTION: Indicated RCS level will rise approximately 1 ft for every 0.5 psi rise in RCS pressure if the indication is not pressure compensated.

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CAUTION: Only borated water should be added to the RCS to maintain adequate shutdown margin.

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Step	Action/Expected Response	Response NOT Obtained
___ 6	**Check RCS level ADEQUATE**	
6.1	Compare any available level indications. [] LT 2965A&B/level hose [] LI-2384 1B LOOP RCS NR LVL [] LI-2385 1C LOOP RCS NR LVL [] Temporary remote level indicator off of a RCS FT on A or C loop	
6.2	Check RCS level within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing.	6.2 Raise RCS level. 6.2.1 Notify personnel in containment that RCS level will be raised. 6.2.2 Align Technical Requirements Manual boration flow path. 6.2.3 Raise RCS level to within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing for the existing RHR flow.

Step	Action/Expected Response	Response NOT Obtained
___ 7	**Maintain RCS level within the following limits:** [] Maintain RCS level to within the Acceptable Operating Region of FIGURE 1, RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing for the existing RHR flow. [] Maintain RCS level less than 123 ft 4 in if personnel are in the channel heads without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if primary manways are removed without nozzle dams installed. [] Maintain RCS level less than 123 ft 9 in if seal injection is not established and RCPs are not backseated. [] Maintain RCS level less than 124 ft if safety injection check valves are disassembled.	7 Verify RHR PUMP(s) stopped AND proceed to step 13.

Step	Action/Expected Response	Response NOT Obtained

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CAUTION: IF the leaking RHR train can NOT be identified, THEN both trains should be assumed leaking.

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___ 8 **Check RHR system - INTACT**

- [] Stable RCS level.
- [] No unexpected rise in containment sump level.
- [] No RHR HX room sump level rising.
- [] No RHR pump room sump level rising.
- [] No waste gas processing room sump level rising
- [] No rising area radiation monitor
- [] No unexplained rise in PRT level or temperature.

8 Isolate RHR leakage.

8.1 Isolate affected RHR train(s) from RCS.

8.1.1 Stop affected RHR pump(s).

8.1.2 Verify closed affected RHR train valves.

Affected RHR Train	A	B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSED	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3
1A(1B) RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B
1A(1B) RHR TO RCS HOT LEGS XCON Q1E11MOV	[] 8887A	[] 8887B

8.2 Isolate source of any RHR/RCS leakage.

___ 9 **Check core cooling provided by RHR or SGs.**

9 Proceed to step 13.

___10 **Check RCS temperature stable or lowering.**

10 Proceed to step 13.

Step	Action/Expected Response	Response NOT Obtained
___11	**Verify low pressure letdown aligned to operating RHR train:**	
11.1	Determine RHR train that low pressure letdown is aligned.	
11.2	<u>IF</u> required, <u>THEN</u> align low pressure letdown to the operating RHR train using FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM	
___12	**Go to procedure and step in effect.**	

****

<u>CAUTION</u>: Containment closure is required to be completed within 2 hours of the initiating event unless an operable RHR pump is placed in service cooling the RCS and the RCS temperature is below 180 F.

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Step	Action/Expected Response	Response NOT Obtained
___13	**Begin establishing containment closure using FNP-1-STP-18.4, CONTAINMENT MID-LOOP <u>AND</u>/OR REFUELING INTEGRITY VERIFICATION <u>AND</u> CONTAINMENT CLOSURE.**	13 <u>IF</u> in mode 6, <u>THEN</u> refer to Technical Specifications 3.9.4 and 3.9.5 for other containment isolation requirements.

Step	Action/Expected Response	Response NOT Obtained
___14	**Monitor time to core saturation.**	
14.1	Check time to core saturation from the current Shutdown Safety Assessment.	14.1 Determine time to core saturation: • Use ATTACHMENT 3, Time to Core Saturation OR • Monitor any available core exit thermocouples for a heat up trend.
14.2	Monitor RCS temperature trend during the performance of this procedure.	
14.2.1	Check vacuum degas system NOT in service.	14.2.1 IF vacuum refill in progress maintaining a vacuum on the RCS, THEN break vacuum on the RCS using FNP-0-SOP-74.0, OPERATION OF THE RCVRS SKID. (155' CTMT)

NOTE: Step 14.2.2 is a continuing action step.

Step	Action/Expected Response	Response NOT Obtained
14.2.2	IF RCS level decreases to less than 121 ft 11 in AND core exit T/Cs are greater than 200°F, THEN proceed to step 21.	
14.3	IF applicable, THEN review the current shutdown safety assessment of FNP-0-UOP-4.0 for applicability of other outage Abnormal Operating Procedures.	
___15	**Begin venting any RHR trains which have experienced evidence of cavitation using ATTACHMENT 1, RHR PUMP VENTING.**	

Step	Action/Expected Response	Response NOT Obtained
___16	**Suspend any boron dilution in progress. (IN 91-54)**	
___17	**IF the charging system is still in service, THEN align the RWST to the running Charging pump.**	

Operable CHG PUMP	1A	1B(A TRN)	1B(B TRN)	1C
RWST TO CHG PUMP Q1E21LCV	[] 115B	[] 115B	[] 115D	[] 115D

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CAUTION: The RCS tygon level hose and LT 2965A&B utilize the same level tap. These are not independent indications.

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Step	Action/Expected Response	Response NOT Obtained
18	**Check for two independent RCS level indications.**	
___ 18.1	Compare available level indications. [] LT 2965A&B/level hose [] LI-2384 1B LOOP RCS NR LVL [] LI-2385 1C LOOP RCS NR LVL [] Temporary remote level indicator off of a RCS FT on A or C loop	
18.2	Check RCS level greater than 123 ft 3 in.	18.2 Raise RCS level. 18.2.1 Notify personnel in containment that RCS level will be raised. 18.2.2 Align Technical Requirements Manual boration flow path. 18.2.3 Raise RCS level to greater than 123 ft 3 in.

Step 18 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

18.3 Maintain RCS level within the following limits:

- [] Maintain RCS level less than 123 ft 4 in if personnel are in the channel heads without nozzle dams installed.
- [] Maintain RCS level less than 123 ft 9 in if primary manways are removed without nozzle dams installed.
- [] Maintain RCS level less than 123 ft 9 in if seal injection is not established and RCPs are not backseated.
- [] Maintain RCS level less than 124 ft if safety injection check valves are disassembled.

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CAUTION: The standby RHR train may be lost due to cavitation if it is placed in service without adequate RCS level.

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CAUTION: Starting an RHR PUMP may cause RCS level to fall due to shrink or void collapse.

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NOTE: The term "standby RHR train" refers to the train most readily available to restore RHR cooling.

Step	Action/Expected Response	Step	Response NOT Obtained
19	**WHEN RCS level greater than 123 ft 3 in, THEN place standby RHR train in service.**	19	IF unable to establish at least one train of RHR, THEN proceed to step 21 while continuing efforts to restore at least one train of RHR.
19.1	Verify CCW PUMP in standby train - STARTED.		

Step 19 continued on next page.

Step | **Action/Expected Response** | **Response NOT Obtained**

19.2 Verify CCW - ALIGNED TO STANDBY RHR HEAT EXCHANGER.

Standby RHR Train	A	B
CCW TO 1A(1B) RHR HX Q1P17MOV	[] 3185A	[] 3185B

19.3 Verify the following conditions satisfied.

19.3.1 RWST TO 1A(1B) RHR PUMP Q1E11MOV8809A and B closed.

19.3.2 1A(1B) RHR HX TO CHG PUMP SUCT Q1E11MOV8706A and B closed.

19.3.3 RCS pressure less than 402.5 psig.

19.3.4 PRZR vapor space temperature less than 475°F.

Step 19 continued on next page.

Step | **Action/Expected Response** | **Response NOT Obtained**

NOTE: RCS to RHR loop suction valves will be deenergized if RCS TAVG is less than 180°F.

19.4 Verify standby RHR train loop suction valves - OPEN.

Standby RHR Train	A	B
1C(1A) RCS LOOP to 1A(1B) RHR PUMP Q1E11MOV	[] 8701A [] 8701B	[] 8702A [] 8702B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP LOOP SUCTION POWER SUPPLY BREAKERS CLOSE(IF REQUIRED)	[] FU-T5 [] FV-V2	[] FU-G2 [] FV-V3

Step 19 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

19.5 Check standby RHR train discharge flow path available.

19.5.1 Verify standby RHR train - ALIGNED TO RCS COLD LEGS.

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV——OPEN	[] 8888A	[] 8888B

NOTE: The RHR HX bypass valves will fail closed and the RHR HX discharge valves will fail open upon loss of air to the AUX BLDG.

19.5.2 Verify standby RHR train HX BYP FLOW - ADJUSTED TO 15% OPEN.

Standby RHR Train	A	B
1A(1B) RHR HX BYP FLOW FK	[] 605A	[] 605B

19.5.3 Verify standby RHR train HX discharge valve - ADJUSTED CLOSED.

Standby RHR Train	A	B
1A(1B) RHR HX TO RCS DISCH VLV HIK	[] 603A	[] 603B

Response NOT Obtained:

19.5.3 Close standby RHR train - TO RCS COLD LEGS ISO valves. (121 ft, AUX BLDG piping penetration room)

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B

Step 19 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

19.6 Verify standby RHR train pump miniflow valve - OPEN.

Standby RHR Train	A	B
1A(1B) RHR PUMP MINIFLOW Q1E11FCV	[] 602A	[] 602B

19.7 Start RHR PUMP in standby train.

19.8 Control standby RHR train RHR HX bypass valve to obtain desired flow.

Standby RHR Train	A	B
1A(1B) RHR HX BYP FLOW FK	[] 605A	[] 605B

19.8 IF unable to control standby RHR train flow with RHR HX bypass valve, THEN locally control RHR HX TO RCS COLD LEGS ISO valves. (121 ft, AUX BLDG piping penetration room)

RHR Train	A	B
RHR HX TO RCS COLD LEGS ISO Q1E11MOV	[] 8888A	[] 8888B

___20 **IF RHR restored, THEN go to procedure and step in effect.**

20 Continue efforts to restore at least one RHR train while continuing with this procedure.

Step	Action/Expected Response	Response NOT Obtained

___21 **Initiate protective measures for personnel in containment.**

21.1 Evacuate all nonessential personnel from containment.

21.2 Ensure HP monitors essential personnel remaining in containment for the following:

[] Changing containment conditions which could require evacuation of all personnel.
[] Use of extra protective clothing if needed.
[] Use of respirators if needed.

21.3 Monitor containment radiation monitors for changing conditions.

[] R-2 CTMT 155 ft
[] R-7 SEAL TABLE
[] R-27A CTMT HIGH RANGE (BOP)
[] R-27B CTMT HIGH RANGE (BOP)

Step	Action/Expected Response	Response NOT Obtained
___22	**Start all available containment coolers**	
___ 22.1	Determine which containment coolers have Service Water aligned. [] Q1E12H001A [] Q1E12H001B [] Q1E12H001C [] Q1E12H001D	
22.2	Start Containment coolers with service water aligned and with power available in FAST speed. [] 1A CTMT CLR FAN FAST SPEED Q1E12H001A to START (BKR EA10) [] 1B CTMT CLR FAN FAST SPEED Q1E12H001B to START (BKR EB05) [] 1C CTMT CLR FAN FAST SPEED Q1E12H001C to START (BKR EB06) [] 1D CTMT CLR FAN FAST SPEED Q1E12H001C to START (BKR EC12)	22.2 Start Containment coolers with service water aligned and with power available in SLOW speed. [] 1A CTMT CLR FAN SLOW SPEED Q1E12H001A to START (BKR ED15) [] 1B CTMT CLR FAN SLOW SPEED Q1E12H001B to START (BKR ED16) [] 1C CTMT CLR FAN SLOW SPEED Q1E12H001C to START (BKR EE08) [] 1D CTMT CLR FAN SLOW SPEED Q1E12H001D to START (BKR EE16)
22.3	Check discharge damper open on any started containment cooler. [] CTMT CLR 1A DISCH 3186A indicates OPEN. [] CTMT CLR 1B DISCH 3186B indicates OPEN. [] CTMT CLR 1C DISCH 3186C indicates OPEN. [] CTMT CLR 1D DISCH 3186d indicates OPEN.	22.3 STOP any containment cooler whose discharge damper fails to indicate OPEN.
___23	**<u>IF</u> not previously started, <u>THEN</u> begin venting any RHR train(s) which have experienced evidence of cavitation using ATTACHMENT 1, RHR PUMP VENTING.**	

Step	Action/Expected Response		Response NOT Obtained

NOTE: Steps 24 and 25 should be performed in conjunction with the remainder of this procedure.

Step	Action/Expected Response		Response NOT Obtained
___24	**Check SGs available.** • Check SG primary nozzle dams - REMOVED. • Check SG primary manways - INSTALLED. • Check SG secondary handhole covers - INSTALLED.	24	Proceed to step 26.

NOTE: Establishing a secondary heat sink will reduce RCS heat up and pressurization rate to provide more time for recovery actions.

Step	Action/Expected Response		Response NOT Obtained
___25	**Verify secondary heat sink established.**		
25.1	Maintain wide range level in all available SGs greater than 75% using FNP-1-SOP-22.0, AUXILIARY FEEDWATER SYSTEM.		
25.2	IF SG steam space intact, THEN open atmospheric relief valves to prevent SG pressurization. 1A(1B,1C) MS ATMOS REL VLV [] PC 3371A adjusted [] PC 3371B adjusted [] PC 3371C adjusted		
25.3	IF SGBD system available, AND AFW system available, THEN establish blowdown from available SGs using FNP-1-SOP-16.3, STEAM GENERATOR FILLING AND DRAINING.		

Step	Action/Expected Response	Response NOT Obtained
___26	**Evaluate event classification and notification requirements using NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION AND INITIAL ACTION, NMP-EP-111, EMERGENCY NOTIFICATIONS, and FNP-0-EIP-8, NON-EMERGENCY NOTIFICATIONS.**	
___27	**Verify RCS isolated.**	
27.1	Close RHR TO LTDN HX HIK 142.	
27.2	Close LTDN LINE ISO Q1E21LCV459 and Q1E21LCV460.	
27.3	Close EXC LTDN LINE ISO VLV Q1E21HV8153 and Q1E21HV8154.	
27.4	Dispatch personnel to isolate all known RCS drain paths.	
27.5	Dispatch personnel to isolate any RCS leakage.	
___28	**Dispatch personnel to close hot leg recirculation valve disconnects. (139 ft, AUX BLDG rad-side)** CHG PUMP TO RCS HOT LEGS Q1E21MOV8886(8884) [] Q1R18B029-A (Master Z key) [] Q1R18B033-B (Master Z key)	
___29	**Check core cooling.**	
29.1	Check RCS level LESS than 121 ft 11 in <u>AND</u> core exit T/Cs GREATER than 200°F.	29.1 Return to step 1.0.

Step	Action/Expected Response	Response NOT Obtained

NOTE:
- Maintaining RCS level is the primary concern. RCS makeup should be restored as soon as possible through any available makeup path.
- RCS makeup flow requirements can exceed 90 gpm due to boil off if an adequate hot leg vent is established.

___30 **<u>WHEN</u> RHR flow restored,**
<u>THEN</u> proceed to step 40.

___31 **Check any CHG PUMP - AVAILABLE.**

31 Establish RWST gravity drain using ATTACHMENT 2, RWST TO RCS GRAVITY FEED.

31.1 <u>WHEN</u> gravity drain established,
<u>THEN</u> proceed to step 37.

___32 **Verify operable CHG PUMP miniflow valves - OPEN.**

1A(1B,1C) CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8109A
[] Q1E21MOV8109B
[] Q1E21MOV8109C

___33 **Verify CHG PUMP miniflow isolation valve - OPEN.**

CHG PUMP
MINIFLOW ISO
[] Q1E21MOV8106

___34 **Verify RWST to CHG PUMP valve for operable CHG PUMP - OPEN.**

Operable CHG PUMP	1A	1B(A TRN)	1B(B TRN)	1C
RWST TO CHG PUMP Q1E21LCV	[] 115B	[] 115B	[] 115D	[] 115D

___35 **Verify operable CHG PUMP - STARTED.**

Step | **Action/Expected Response** | **Response NOT Obtained**

___36 **Verify required injection path isolation valve - OPEN.**

Q1E21MOV8803A	HHSI TO RCS CL ISO
Q1E21MOV8803B	HHSI TO RCS CL ISO
Q1E21MOV8885	CHG PUMP RECIRC TO RCS COLD LEGS
Q1E21MOV8884	CHG PUMP RECIRC TO RCS HOT LEGS
Q1E21MOV8886	CHG PUMP RECIRC TO RCS HOT LEGS

Step	Action/Expected Response	Response NOT Obtained

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CAUTION: Reactor vessel level may be much lower than indicated if no hot leg vent path is available.

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CAUTION: RCS pressurization may cause SG nozzle dam failure. This will cause a rapid loss of RCS inventory and the creation of a RCS spill pathway.

****

37 IF RCS configuration will allow a level in the pressurizer, THEN establish feed and bleed cooling.

37.1 Verify RCS bleed path available as follows.

- Verify all pressurizer safety valves - REMOVED.

OR

- Verify pressurizer manway - REMOVED.

OR

- Verify both PRZR PORVs and PRZR PORV ISOs - OPEN.

37 IF RCS configuration will NOT allow a level in the pressurizer, THEN establish feed and spill cooling as follows.

a) Locally control required injection path isolation valve to maintain core exit T/Cs less than 200°F.

b) Proceed to step 38.

Step 37 continued on next page.

Step	Action/Expected Response	Response NOT Obtained
37.2	WHEN pressurizer level greater than 7% (136 ft 9 in), THEN establish normal charging.	37.2 Locally control required injection path isolation valve to maintain pressurizer level greater than 7% (136 ft 9 in).
37.2.1	Verify charging pump miniflow valves - OPEN. 1A(1B,1C) CHG PUMP MINIFLOW ISO [] Q1E21MOV8109A [] Q1E21MOV8109B [] Q1E21MOV8109C CHG PUMP MINIFLOW ISO [] Q1E21MOV8106	
37.2.2	Manually close charging flow control valve. CHG FLOW [] FK 122	
37.2.3	Verify charging pump discharge flow path - ALIGNED. CHG PUMP DISCH HDR ISO [] Q1E21MOV8132A open [] Q1E21MOV8132B open [] Q1E21MOV8133A open [] Q1E21MOV8133B open CHG PUMPS TO REGENERATIVE HX [] Q1E21MOV8107 open [] Q1E21MOV8108 open	

Step 37 continued on next page.

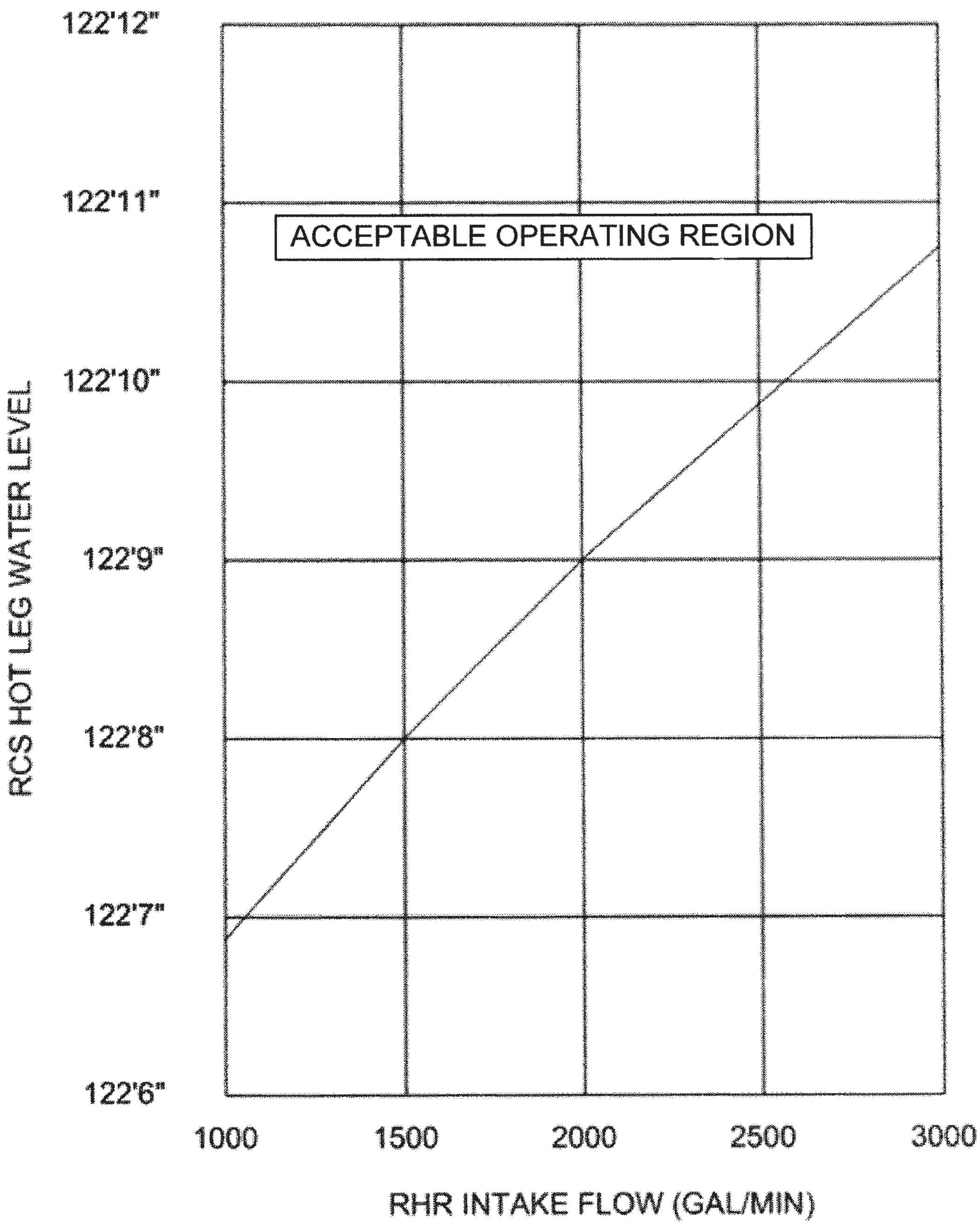
Step	Action/Expected Response	Response NOT Obtained
37.2.4	Verify only one charging line valve - OPEN. RCS NORMAL CHG LINE [] Q1E21HV8146 RCS ALT CHG LINE [] Q1E21HV8147	
37.2.5	Maintain pressurizer level greater than 7% (136 ft 9 in). CHG FLOW [] FK 122 adjusted	
37.2.6	Close required injection path isolation valve.	
___38	**Maintain RCS feed and bleed cooling until at least one RHR train restored.**	38 Maintain RCS feed and spill cooling until at least one RHR train restored.
___39	**Check RHR - RESTORED.**	39 Return to step 37.
___40	**Maintain RCS at desired level.**	
___41	**Begin RCS cooldown using FNP-1-SOP-7.0, RESIDUAL HEAT REMOVAL SYSTEM.**	
___42	**<u>WHEN</u> core exit T/Cs stable at desired temperature, <u>THEN</u> go to procedure and step in effect.**	

-END-

FIGURE 1

RCS HOT LEG LEVEL vs RHR INTAKE FLOW To Minimize Vortexing

RCS HOT LEG LEVEL vs RHR INTAKE FLOW
To Minimize Vortexing



Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

RHR PUMP VENTING

****

CAUTION: Installation of vent rigs must not delay venting operations if only the air bound train is available for service. Contamination should be minimized but contamination control must not interfere with venting.

****

___ 1 **IF both trains of RHR are air bound OR unavailable, THEN proceed to step 4**

NOTE: Vent rigs may be routed to either floor drains or poly bottles.

___ 2 **IF 1A RHR PUMP AIR bound, THEN install vent rigs on A train RHR system.**

2.1 Install vent rig at 1A RHR PUMP SEAL COOLER OUTLET VENT ISO Q1E11V080C. (83 ft, AUX BLDG 1A RHR PUMP room)

2.2 Install vent rig at 1A RHR HX OUTLET VENT ISO Q1E11V068C. (83 ft, AUX BLDG RHR HX room)

2.3 Install vent rig at 1C RCS LOOP TO 1A RHR PUMP HDR VENT ISO Q1E11V064C. (100 ft, AUX BLDG piping penetration room, PEN #16)

2.4 Install vent rig at 1A RHR HX TO RCS COLD LEGS HDR VENT ISO Q1E11V055B. (121 ft, AUX BLDG piping penetration room, PEN #15)

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

___ 3 **<u>IF</u> 1B RHR PUMP air bound, <u>THEN</u> install vent rigs on B train RHR system.**

3.1 Install vent rig at 1B RHR PUMP SEAL COOLER OUTLET VENT ISO Q1E11V080D. (83 ft, AUX BLDG 1B RHR PUMP room)

3.2 Install vent rig at 1B RHR HX OUTLET VENT ISO Q1E11V068D. (83 ft, AUX BLDG RHR HX room)

3.3 Install vent rig at 1A RCS LOOP TO 1B RHR PUMP HDR VENT ISO Q1E11V064D. (100 ft, AUX BLDG piping penetration room, PEN #18)

3.4 Install vent rig at 1B RHR HX TO RCS COLD LEGS HDR VENT ISO Q1E11V058B. (121 ft, AUX BLDG piping penetration room, PEN #17)

Step | **Action/Expected Response** | **Response NOT Obtained**

ATTACHMENT 1

****

CAUTION: Using the RCS as a makeup source for RHR system inventory lost during venting (per RNO), will result in a loss of RCS inventory and therefore a lowering of RCS level. This could jeopardize the other train of RHR, if it is in operation.

****

NOTE: The intent of aligning the RWST to the air bound train when the RCS loop suctions are open is to make up for inventory lost when venting, however, this action also initiates gravity flow from the RWST. Close coordination will be required between the control room operator monitoring RCS level and the operator controlling the RWST supply locally.

___ **4 Align a source of make up to the air bound train.**

4.1 Locally, throttle open RWST supply to air bound train until it is just off the closed seat. (83 ft el, RHR PUMP Rm)

Air Bound Train	A	B
RWST TO 1A(1B) RHR PUMP Q1E11MOV	[]8809A	[]8809B

RNO:

4.1 Open RCS supply to air bound train.

Air Bound Train	A	B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[]8701A []8701B	[]8702A []8702B

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

___ 5 **<u>IF</u> 1A RHR PUMP air bound,**
<u>THEN</u> perform the following.

5.1 Open 1A RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080C and Q1E11V080A. (83 ft, AUX BLDG 1A RHR PUMP room)

5.2 <u>WHEN</u> air free water is seen, <u>THEN</u> close 1A RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080C and Q1E11V080A. (83 ft, AUX BLDG 1A RHR PUMP room)

5.3 Open 1A RHR HX OUTLET VENTS Q1E11V068C and Q1E11V068A. (83 ft, AUX BLDG RHR HX room)

5.4 <u>WHEN</u> air free water is seen, <u>THEN</u> close 1A RHR HX OUTLET VENTS Q1E11V068C and Q1E11V068A. (83 ft, AUX BLDG RHR HX room)

5.5 Open 1C RCS LOOP TO 1A RHR PUMP HDR VENTS Q1E11V064C and Q1E11V064A. (100 ft, AUX BLDG piping penetration room, PEN #16)

5.6 <u>WHEN</u> air free water is seen, <u>THEN</u> close 1C RCS LOOP TO 1A RHR PUMP HDR VENTS Q1E11V064C and Q1E11V064A. (100 ft, AUX BLDG piping penetration room)

5.7 Open 1A RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V055B and Q1E11VO55A. (121 ft, AUX BLDG piping penetration room, PEN #15)

5.8 <u>WHEN</u> air free water is seen, <u>THEN</u> close 1A RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V055B and Q1E11VO55A. (121 ft, AUX BLDG piping penetration room)

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

___ 6 **IF 1B RHR PUMP air bound, THEN perform the following.**

6.1 Open 1B RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080D and Q1E11V080B. (83 ft, AUX BLDG 1B RHR PUMP room)

6.2 WHEN air free water is seen, THEN close 1B RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080D and Q1E11V080B. (83 ft, AUX BLDG 1B RHR PUMP room)

6.3 Open 1B RHR HX OUTLET VENTS Q1E11V068D and Q1E11V068B. (83 ft, AUX BLDG RHR HX room)

6.4 WHEN air free water is seen, THEN close 1B RHR HX OUTLET VENTS Q1E11V068D and Q1E11V068B. (83 ft, AUX BLDG RHR HX room)

6.5 Open 1A RCS LOOP TO 1B RHR PUMP HDR VENTS Q1E11V064D and Q1E11V064B. (100 ft, AUX BLDG piping penetration room, PEN #18)

6.6 WHEN air free water is seen, THEN close 1A RCS LOOP TO 1B RHR PUMP HDR VENTS Q1E11V064D and Q1E11V064B. (100 ft, AUX BLDG piping penetration room)

6.7 Open 1B RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V058B and Q1E11VO58A. (121 ft, AUX BLDG piping penetration room, PEN #17)

6.8 WHEN air free water is seen, THEN close 1B RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V058B and Q1E11VO58A. (121 ft, AUX BLDG piping penetration room)

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

___ 7 **IF RWST aligned to air bound train, THEN prepare the air bound pump for starting as follows.**

7.1 Verify closed RCS supply to air bound train.

Air Bound Train	A	B
1C(1A) RCS LOOP TO 1A(1B) RHR PUMP Q1E11MOV	[]8701A []8701B	[]8702A []8702B

7.2 Verify air bound train RHR HX BYP FLOW - ADJUSTED TO 15% OPEN.

Air Bound Train	A	B
1A(1B) RHR HX BYP FLOW FK	[]605A	[]605B

7.3 Verify air bound train RHR HX discharge valve - ADJUSTED CLOSED.

Air Bound Train	A	B
1A(1B) RHR HX TO RCS DISCH VLV HIK	[]603A	[]603B

7.4 Open fully RWST supply to air bound train.

Air Bound Train	A	B
RWST TO 1A(1B) RHR PUMP Q1E11MOV	[]8809A	[]8809B

Response NOT Obtained:

7 IF RCS aligned to air bound train, THEN prepare the air bound pump for starting as follows.

a) Verify air bound train RHR HX BYP FLOW - ADJUSTED TO 15% OPEN.

Air Bound Train	A	B
1A(1B) RHR HX BYP FLOW FK	[]605A	[]605B

b) Verify air bound train RHR HX discharge valve - ADJUSTED CLOSED.

Air Bound Train	A	B
1A(1B) RHR HX TO RCS DISCH VLV HIK	[]603A	[]603B

c) Proceed to step 8.

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

****

CAUTION: Excessive start/stop cycling of RHR PUMPs may cause motor damage.

****

___ 8 **Run air bound RHR PUMP for 10 seconds.**

___ 9 **IF 1A RHR PUMP was run for 10 seconds, THEN perform the following.**

9.1 Open 1A RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080C and Q1E11V080A. (83 ft, AUX BLDG 1A RHR PUMP room)

9.2 WHEN air free water is seen, THEN close 1A RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080C and Q1E11V080A. (83 ft, AUX BLDG 1A RHR PUMP room)

9.3 Open 1A RHR HX OUTLET VENTS Q1E11V068C and Q1E11V068A. (83 ft, AUX BLDG RHR HX room)

9.4 WHEN air free water is seen, THEN close 1A RHR HX OUTLET VENTS Q1E11V068C and Q1E11V068A. (83 ft, AUX BLDG RHR HX room)

9.5 Open 1C RCS LOOP TO 1A RHR PUMP HDR VENTS Q1E11V064C and Q1E11V064A. (100 ft, AUX BLDG piping penetration room, PEN #16)

9.6 WHEN air free water is seen, THEN close 1C RCS LOOP TO 1A RHR PUMP HDR VENTS Q1E11V064C and Q1E11V064A. (100 ft, AUX BLDG piping penetration room)

Step 9 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

9.7 Open 1A RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V055B and Q1E11V055A. (121 ft, AUX BLDG piping penetration room, PEN #15)

9.8 WHEN air free water is seen, THEN close 1A RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V055B and Q1E11V055A. (121 ft, AUX BLDG piping penetration room)

___10 IF 1B RHR PUMP was run for 10 seconds, THEN perform the following.

10.1 Open 1B RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080D and Q1E11V080B. (83 ft, AUX BLDG 1B RHR PUMP room)

10.2 WHEN air free water is seen, THEN close 1B RHR PUMP SEAL COOLER OUTLET VENTS Q1E11V080D and Q1E11V080B. (83 ft, AUX BLDG 1B RHR PUMP room)

10.3 Open 1B RHR HX OUTLET VENTS Q1E11V068D and Q1E11V068B. (83 ft, AUX BLDG RHR HX room)

10.4 WHEN air free water is seen, THEN close 1B RHR HX OUTLET VENTS Q1E11V068D and Q1E11V068B. (83 ft, AUX BLDG RHR HX room)

10.5 Open 1A RCS LOOP TO 1B RHR PUMP HDR VENTS Q1E11V064D and Q1E11V064B. (100 ft, AUX BLDG piping penetration room, PEN #18)

Step 10 continued on next page.

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 1

10.6	<u>WHEN</u> air free water is seen, <u>THEN</u> close 1A RCS LOOP TO 1B RHR PUMP HDR VENTS Q1E11V064D and Q1E11V064B. (100 ft, AUX BLDG piping penetration room)	
10.7	Open 1B RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V058B and Q1E11VO58A. (121 ft, AUX BLDG piping penetration room, PEN #17)	
10.8	<u>WHEN</u> air free water is seen, <u>THEN</u> close 1B RHR HX TO RCS COLD LEGS HDR VENTS Q1E11V058B and Q1E11VO58A. (121 ft, AUX BLDG piping penetration room)	
___11	**<u>IF</u> no air seen, <u>THEN</u> notify control room that venting is complete.**	11 Return to step 8.
___12	**<u>WHEN</u> desired, <u>THEN</u> remove RHR vent rigs.**	
___13	**<u>WHEN</u> desired, <u>THEN</u> verify vent lines capped.**	
___14	**Notify control room that ATTACHMENT 1 is complete.**	

-END-

Step	Action/Expected Response	Response NOT Obtained

ATTACHMENT 2

RWST TO RCS GRAVITY FEED

****

CAUTION: Gravity feed may not be sufficient to prevent core uncovery if a secondary heat sink or a hot leg vent path is not available.

****

NOTE:
- ATTACHMENT 2, FIGURE 1 and ATTACHMENT 2, FIGURE 2 provide expected gravity feed flow rates.
- RWST TO 1A(1B) RHR PUMP Q1E11MOV8809A and Q1E11MOV8809B may be locally adjusted to control gravity feed flow at the Shift Supervisor's discretion.

Step	Action/Expected Response	Response NOT Obtained
___ 1	**IF A train RHR to RCS hot leg flow path available, THEN perform the following.**	1 IF B train RHR to RCS hot leg flow path available, THEN perform the following.
	1.1 Open 1C RCS LOOP TO 1A RHR PUMP Q1E11MOV8701A and Q1E11MOV8701B.	a) Open 1A RCS LOOP TO 1B RHR PUMP Q1E11MOV8702A and Q1E11MOV8702B.
	1.2 Open RWST TO 1A RHR PUMP Q1E11MOV8809A to establish gravity feed.	b) Open RWST TO 1B RHR PUMP Q1E11MOV8809B to establish gravity feed.
___ 2	**IF gravity feed established, THEN proceed to step 4.**	

Step	Action/Expected Response	Response NOT Obtained

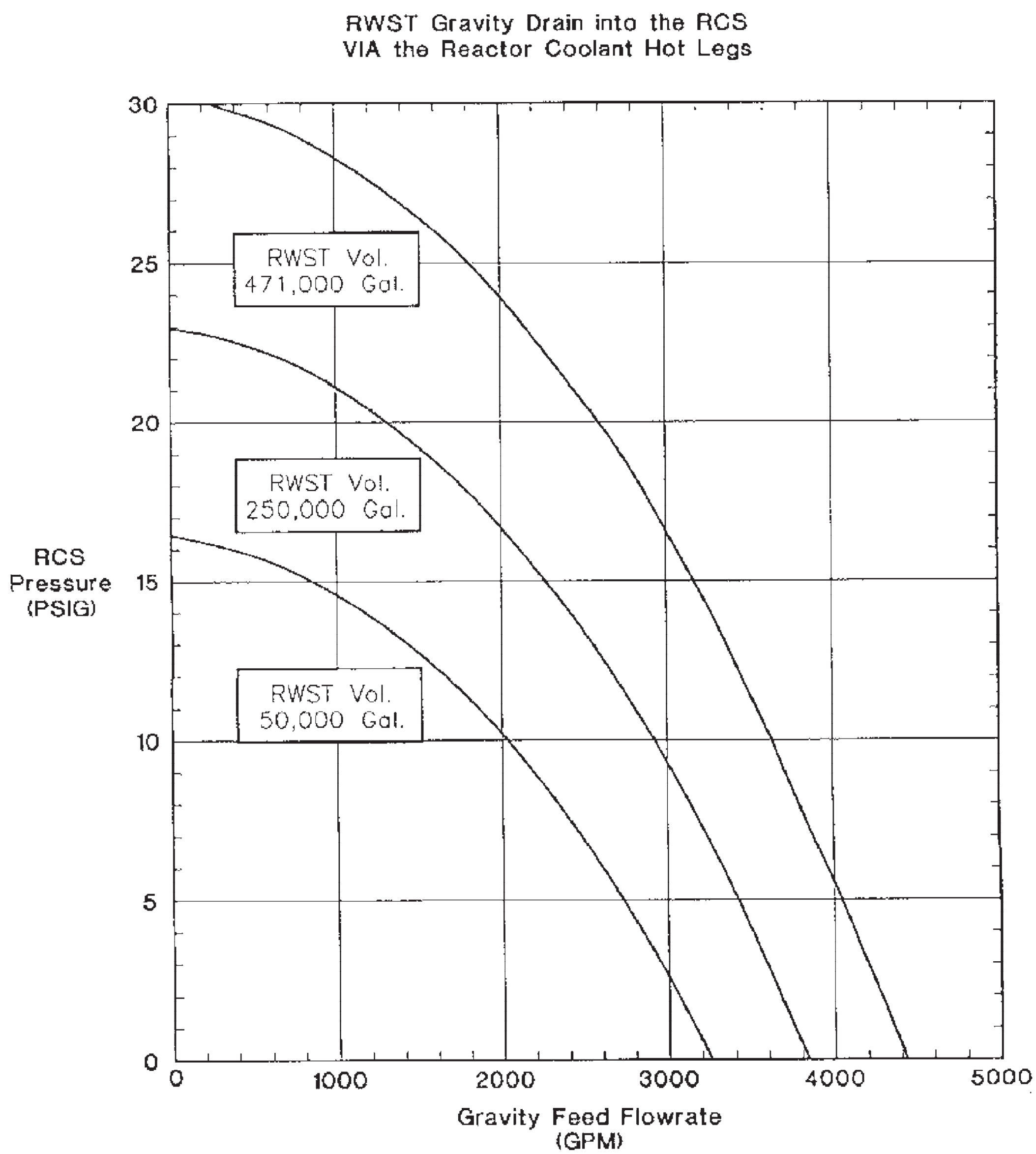
ATTACHMENT 2

___ 3	**IF A train RHR to RCS cold leg flow path available, THEN perform the following.**	3 IF B train RHR to RCS cold leg flow path available, THEN perform the following.
	3.1 Verify 1C RCS LOOP TO 1A RHR PUMP Q1E11MOV8701A and Q1E11MOV8701B - CLOSED.	a) Verify 1A RCS LOOP TO 1B RHR PUMP Q1E11MOV8702A and Q1E11MOV8702B - CLOSED.
	3.2 Verify 1A RHR PUMP MINIFLOW Q1E11FCV602A - OPEN.	b) Verify 1B RHR PUMP MINIFLOW Q1E11FCV602B - OPEN.
	3.3 Verify 1A RHR HX TO RCS COLD LEGS ISO Q1E11MOV8888A - OPEN.	c) Verify 1B RHR HX TO RCS COLD LEGS ISO Q1E11MOV8888B - OPEN.
	3.4 Open RWST TO 1A RHR PUMP Q1E11MOV8809A to establish gravity feed.	d) Open RWST TO 1B RHR PUMP Q1E11MOV8809B to establish gravity feed.
___ 4	**Notify control room that ATTACHMENT 2 is complete.**	

-END-

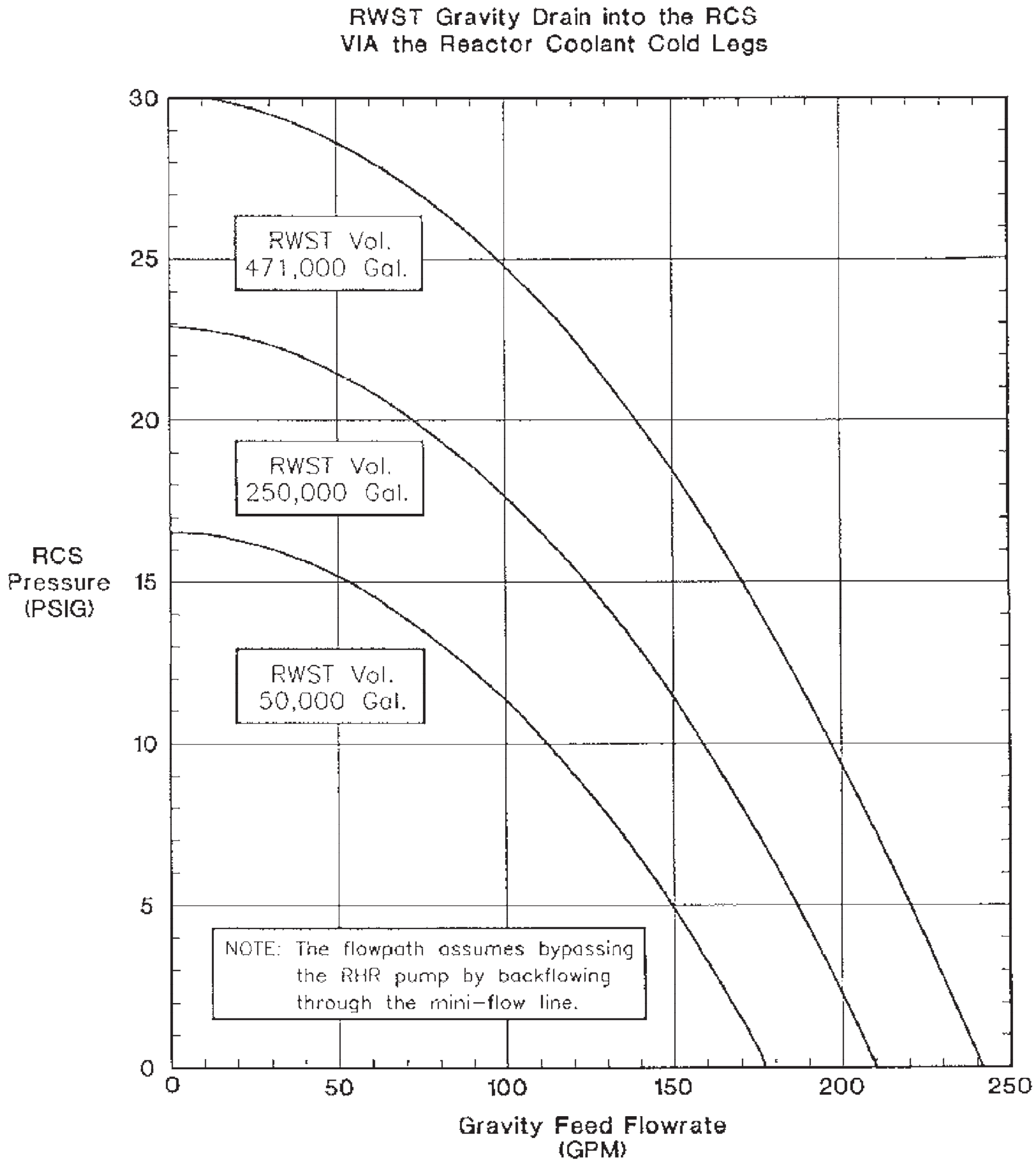
ATTACHMENT 2

FIGURE 1



ATTACHMENT 2

FIGURE 2



ATTACHMENT 3

Time to Core Saturation

1 Time to Core Saturation:

1.1 Tables A and B provide estimates of the time to core boiling following a loss RHR capability for two cases:

1.1.1 **TABLE A** provides a Time to Saturation as a function of time after shutdown for a full core immediately after shutdown for a refueling.

1.1.2 **TABLE B** provides a Time to Saturation as a function of time after shutdown for a core in which one third of the spent fuel has been replaced with new fuel.

1.2 Both cases are evaluated for conditions when RCS level is at mid loop (122'9"), at three feet below the reactor flange (126'7"), and when the reactor cavity is full.

1.3 Both cases are also evaluated for three assumed initial temperatures: 100°F, 120°F, and 140°F.

1.4 These figures can be used to estimate the amount of time available for operator action to restore RHR before additional protective measures must be taken.

ATTACHMENT 3

Time to Core Saturation

TABLE A---POWER UPRATED UNIT

TIME TO SATURATION: FULL CORE

ASSUMED INITIAL TEMPERATURE=100°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
40	7.7	10.5	5.6
60	8.7	11.9	6.3
80	9.5	13.0	6.9
100	10.4	14.2	7.5
120	11.3	15.4	8.2
140	11.9	16.3	8.6
160	12.7	17.4	9.2
180	13.3	18.2	9.6
200	13.9	19.0	10.1
336	17.1	23.4	12.4
504	20.8	28.5	15.1

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

ATTACHMENT 3

Time to Core Saturation

TABLE A---POWER UPRATED UNIT

TIME TO SATURATION: FULL CORE

ASSUMED INITIAL TEMPERATURE=120°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
40	6.3	8.6	4.5
60	7.1	9.8	5.2
80	7.8	10.6	5.6
100	8.5	11.7	6.2
120	9.2	12.6	6.7
140	9.8	13.4	7.1
160	10.4	14.2	7.5
180	10.9	14.9	7.9
200	11.4	15.6	8.2
336	14.0	19.1	10.1
504	17.0	23.3	12.3

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

ATTACHMENT 3

Time to Core Saturation

TABLE A---POWER UPRATED UNIT

TIME TO SATURATION: FULL CORE

ASSUMED INITIAL TEMPERATURE=140°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
40	4.9	6.7	3.5
60	5.6	7.6	4.0
80	6.1	8.3	4.4
100	6.6	9.1	4.8
120	7.2	9.8	5.2
140	7.6	10.4	5.5
160	8.1	11.1	5.9
180	8.5	11.6	6.1
200	8.9	12.1	6.4
336	10.9	14.9	7.9
504	13.3	18.2	9.6

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

ATTACHMENT 3

Time to Core Saturation

TABLE B---POWER UPRATED UNIT

TIME TO SATURATION: ONE THIRD NEW FUEL

ASSUMED INITIAL TEMPERATURE=100°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
100	15.6	21.4	11.3
200	20.9	28.5	15.1
300	24.7	33.7	17.8
400	27.5	37.6	19.9
500	31.1	42.5	22.5
600	34.5	47.3	25.0
700	37.2	51.0	27.0
800	40.4	55.3	29.2

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

ATTACHMENT 3

Time to Core Saturation

TABLE B---POWER UPRATED UNIT

TIME TO SATURATION: ONE THIRD NEW FUEL

ASSUMED INITIAL TEMPERATURE=120°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
100	12.8	17.5	9.2
200	17.1	23.4	12.4
300	20.2	27.6	14.6
400	22.5	30.8	16.3
500	25.4	34.8	18.4
600	28.3	38.7	20.5
700	30.5	41.7	22.1
800	33.0	45.2	23.9

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

ATTACHMENT 3

Time to Core Saturation

TABLE B---POWER UPRATED UNIT

TIME TO SATURATION: ONE THIRD NEW FUEL

ASSUMED INITIAL TEMPERATURE=140°F

Time After Shutdown (hours)	Time to Saturation at midloop (mins)	Time to Saturation 3' below flange (mins)	Time to Saturation full Rx cavity (hours)
100	10.0	13.6	7.2
200	13.3	18.2	9.6
300	15.7	21.5	11.4
400	17.5	24.0	12.7
500	19.8	27.1	14.3
600	22.0	30.1	15.9
700	23.7	32.5	17.2
800	25.7	35.2	18.6

VOLUME REFERENCE TABLE

MIDLOOP VOLUME(FT^3)	945		
VOLUME 3FT BELOW FLANGE(FT^3)	348	TOTAL=	1293
VOLUME FULL REACTOR CAVITY(FT^3)	39750	TOTAL=	41043

-END-

ATTACHMENT 4

REFERENCES/COMMITMENTS

1 0007011 Commmitment completed by Rev 1&2 of this procedure

2 0007012 PROCEDURE STEPS, step 19 Caution prior to the step

3 0007013 PROCEDURE STEPS, step 15

4 0007230, 0007236 Entire procedure fulfills these commitments

5 0007569 PROCEDURE STEPS, step 21.1

6 0007570 PROCEDURE STEPS, step 22

7 0007583 PROCEDURE STEPS, step 31

8 0007584, 0007594, 0009103 Entire procedure fulfills these commitments

-END-

A.1.b SRO

TITLE: Determine Active License Status.
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 30 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☐ TIME CRITICAL ☐ PRA

JPM DIRECTIONS:

1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Requiring the examinee to acquire the required references may or may not be included as part of the JPM.

TASK STANDARD: Upon successful completion of this JPM, the examinee will:

- Correctly assess and determine the Active or Inactive License status of Plant Operators.

Examinee:				
Overall JPM Performance:	**Satisfactory**	☐	**Unsatisfactory**	☐
Evaluator Comments (attach additional sheets if necessary)				

EXAMINER: ______________________

Developer	S Jackson	Date: 4/2/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to DETERMINE THE ACTIVE OR INACTIVE STATUS OF LICENSED OPERATORS. The conditions under which this task is to be performed are:

a. An RO is required to fill the OATC position on October 31, 2015.
b. Three off shift RO's are available.
c. All three are current in LOCT (Licensed Operator Continuing Training) and have had a medical exam as required to maintain an active license.
d. None of the three have worked any shifts since September 30, 2015.
e. The three operators' work history are as follows:

- Operator A - <u>License was **active** on July 1, 2015.</u>
  07/02/15 worked 1900-0700 as Unit 2 OATC
  07/04/15 worked 1900-0700 as Unit 1 UO
  07/05/15 worked 1900-0700 as Unit 1 OATC
  08/14/15 worked 0700-1500 as Unit 2 UO
  08/17/15 worked 0700-1500 as Unit 2 UO
  08/18/15 worked 0700-1100 as Unit 2 UO

- Operator B - <u>License was **active** on July 1, 2015.</u>
  07/28/15 worked 0700-1900 as Unit 1 UO
  08/03/15 worked 0700-1900 as Unit 1 UO
  08/05/15 worked 0700-1900 as an on shift Extra
  08/14/15 worked 1900-0700 as Unit 1 OATC
  09/05/15 worked 0700-1900 as Unit 1 UO

- Operator C - <u>License was **inactive** on July 1, 2015.</u>
  From 07/12/2015 thru 07/16/2015 worked 40 hours under the direction of the Unit 1 OATC and completed all requirements for license reactivation.
  08/15/15 worked 0700-1900 as Unit 2 OATC
  09/04/15 worked 0700-1900 as Unit 2 OATC
  09/16/15 worked 0700-1900 as Unit 1 UO
  09/17/15 worked 0700-1900 as Unit 1 OATC

f. You have been directed to determine the Active or Inactive status of the three off shift RO's on October 31, 2015, in accordance with NMP-TR-406, Active License Maintenance.

INITIATING CUE: IF you have no questions, you may begin.

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS (CIRCLE)
_____ **START TIME**		
* 1. Evaluate the status of Operator A.	Operator A is determined to have **INACTIVE** license status based on the 8/18/15 shift is less than 8 or 12 hours so it does not count toward the 56 hour total. 52 hours count. Step 5.5.2.2 of NMP -TR-406.	S / U
* 2. Evaluate the status of Operator B.	Operator B is determined to have **INACTIVE** license status. This operator worked 5 - 12 hour shifts during the calendar quarter July 1 – September 30, but one of those shifts was **NOT** in a position required by Tech Specs (08/05/2015 working as an on shift Extra). Step 5.5.2.1 of NMP -TR-406.	S / U
* 3. Evaluate the status of Operator C.	Operator C is determined to have **ACTIVE** license status. This operator reactivated his license during the calendar quarter of July 1-September 30, 2015. When a license is reactivated, it is considered active for that quarter without working any additional shifts. When a licensed operator has met the requirements for an active license in a quarter he is available and considered active for the next quarter. Step 5.6.1 and 5.6.8 and 5.6.9 of NMP -TR-406.	S / U
_____ **STOP TIME**		

Terminate when all elements of the task have been completed.

CRITICAL ELEMENTS: Critical Elements are denoted with an asterisk (*) before the element number.

GENERAL REFERENCES:

1. NMP-TR-406, ver 6.2
2. KA: G2.1.4 – 3.3 / 3.8

GENERAL TOOLS AND EQUIPMENT:

1. NMP-TR-406, ver 6.2 – on Reference Disk
2. Scratch paper, calculator as requested.

Critical ELEMENT justification:

STEP	Evaluation
1	**Critical:** Task completion: required to properly evaluate the active or inactive status of Operator A.
2	**Critical:** Task completion: required to properly evaluate the active or inactive status of Operator B.
3	**Critical:** Task completion: required to properly evaluate the active or inactive status of Operator C.

COMMENTS:

KEY

Operator A status - ____**INACTIVE**___________. (Active / Inactive)

Operator B status - ____**INACTIVE** _____. (Active / Inactive)

Operator C status - ____**ACTIVE** ____. (Active / Inactive)

CONDITIONS

When I tell you to begin, you are to DETERMINE THE ACTIVE OR INACTIVE STATUS OF LICENSED OPERATORS. The conditions under which this task is to be performed are:

a. An RO is required to fill the OATC position on October 31, 2015.
b. Three off shift RO's are available.
c. All three are current in LOCT (Licensed Operator Continuing Training) and have had a medical exam as required to maintain an active license.
d. None of the three have worked any shifts since September 30, 2015.
e. The three operators' work history are as follows:

- Operator A - <u>License was **active** on July 1, 2015.</u>
  07/02/15 worked 1900-0700 as Unit 2 OATC
  07/04/15 worked 1900-0700 as Unit 1 UO
  07/05/15 worked 1900-0700 as Unit 1 OATC
  08/14/15 worked 0700-1500 as Unit 2 UO
  08/17/15 worked 0700-1500 as Unit 2 UO
  08/18/15 worked 0700-1100 as Unit 2 UO

- Operator B - <u>License was **active** on July 1, 2015.</u>
  07/28/15 worked 0700-1900 as Unit 1 UO
  08/03/15 worked 0700-1900 as Unit 1 UO
  08/05/15 worked 0700-1900 as an on shift Extra
  08/14/15 worked 1900-0700 as Unit 1 OATC
  09/05/15 worked 0700-1900 as Unit 1 UO

- Operator C - <u>License was **inactive** on July 1, 2015.</u>
  From 07/12/2015 thru 07/16/2015 worked 40 hours under the direction of the Unit 1 OATC and completed all requirements for license reactivation.
  08/15/15 worked 0700-1900 as Unit 2 OATC
  09/04/15 worked 0700-1900 as Unit 2 OATC
  09/16/15 worked 0700-1900 as Unit 1 UO
  09/17/15 worked 0700-1900 as Unit 1 OATC

a. You have been directed to determine the Active or Inactive status of the three off shift RO's on October 31, 2015, in accordance with NMP-TR-406, Active License Maintenance.

Operator A status - _______________. (Active / Inactive)

Operator B status - _______________. (Active / Inactive)

Operator C status - _______________. (Active / Inactive)

5.4.3 After the Licensed Operator signs the NRC Form 398, they shall return it to the Operations Training Group who will then route the NRC Form 398 and the individual's most recent NRC Form 396 to the Training Manager for review and approval.

5.4.3.1 The Training Manager shall sign the NRC Form 398 after the individual and then send the signed form to the Site Vice President or Senior Management Representative.

5.4.4 The Operation Training Group shall mail the original NRC Form 398 and the NRC Form 396, along with a cover letter to the NRC, per 10 CFR 55, at least 30 days (i.e., 25 Administrative day limit) prior to expiration of the Licensed Operator's license. Refer to Attachment 5 for a sample Cover Letter.

NOTE

IF an Operator or Senior Operator applies for a renewal at least 30 days (i.e., 25 Administrative day limit) before the Expiration Date of the existing license, THEN the license does NOT expire until the NRC determines the final disposition of the renewal application.

5.4.5 When the license renewal has been signed and mailed to the NRC in a timely manner (i.e., at least 25 Administrative days prior to expiration) the Operations Training Group shall create a Learning Event for the Licensed Operator for the appropriate item in the LMS for the date the license renewal was submitted.

5.4.6 Upon receipt of the license, renewal from the NRC the Operations Training Group shall edit the Learning Event for the Licensed Operator for the appropriate item in the LMS for the date the license was effective.

5.4.7 The Operations Training Group shall transmit a copy of NRC Form 396 to Medical Services for record processing. The Operations Training Group shall process the cover letter, the NRC Form 398, and the license for records retention.

5.5 Maintenance of an **ACTIVE** License

Only a Licensed Operator with an **ACTIVE** license shall be allowed to manipulate the controls or supervise the manipulation of the controls of the reactor.

5.5.1 Per NUREG 1262 Q. 293, a newly Licensed Operator is considered to have met the proficiency requirements for an active license for the initial calendar quarter in which the license was issued.

5.5.1.1 Upon receipt of a new license, a Learning Event shall be created for the Licensed Operator to give credit for proficiency.

5.5.2 NUREG 1021 states:

"In accordance with 10 CFR 55.53 (e), licensed operators are required to maintain their proficiency by actively performing the functions of an operator or senior operator on at least seven 8-hour or five 12-hour shifts per calendar quarter. This requirement may be completed with a combination of complete 8- and 12-hour shifts (in a position required by the plant's technical specifications) at sites having a mixed shift schedule, and watches shall not be truncated with the minimum quarterly requirement (56 hours) is satisfied. Overtime may be credited if the overtime work is in a position required by the plant's technical specifications. Overtime as an extra "helper" after the official watch has been turned over to another watch stander does not count toward proficiency time."

5.5.2.1 Maintenance of an active license requires that an individual spend seven (7) eight-hour shifts or five (5) twelve-hour shifts in a position that requires the license per the Technical Specifications (i.e., OATC, UO, SS, SM, or SRO during Core Alterations as defined in Technical Specifications) in a calendar quarter.

5.5.2.2 IF an individual stands a combination of complete 12 or eight-hour shifts that total 56 hours in the quarter, THEN this requirement is satisfied. A shift of less than eight (8) hours does NOT count toward the 56-hour total. IF an individual spends this shift time in a position that only requires an RO license (i.e., UO, OATC), THEN they are an active RO only. If they spend this time in an SRO position (i.e., SS, SM) they are an active SRO. IF they spend this time as a SRO in charge of fuel handling during Core Alterations, THEN they are an active SRO only for supervising Core Alterations.

5.5.2.2.1 It is permissible for an individual with an SRO license to maintain only the RO portion of their license in an active state by performing the functions of an RO for a minimum of seven (7) 8-hour OR five (5) 12-hour shifts per calendar quarter pursuant to 10 CFR 55.53(e).

5.5.2.3 In order to maintain the Supervisory portion of the SRO license active, a SRO must stand at least one (1) complete watch per calendar quarter in an SRO-only supervisory position. The remainder of complete watches required in a calendar quarter may be performed in either a credited SRO or RO position. These shifts must be on a unit that has fuel in the vessel. IF a Licensed SRO stands all of their required proficiency watches in an SRO position, THEN the RO portion of the license is still considered active. Performing the required number of shifts per calendar quarter on a single unit maintains the license active for all similar units on an individual's license.

5.5.3 The active Licensed Operator shall complete NMP-TR-406-F01 once per quarter to document these proficiency hours and forward the form to the Operation Training Coordinator.

5.5.4 The Operations Department Training Coordinator or designee shall maintain a record of these hours and create a Learning Event for each Licensed Operator who meets the SRO requirement or for each Licensed Operator who meets the RO requirement. Failure to meet the time requirements for hour's on-shift places that level of license (i.e., RO, SRO) in an "Inactive" status. The Licensed Operator shall NOT be allowed to stand shift in a position that requires that level of license until they have completed reactivation per this procedure. Operations Supervision and the Licensed Operator shall be notified by the Operations Training Group OR the Operations Training Coordinator if the Licensed Operator's license is placed in an 'Inactive' condition.

5.5.5 An **ACTIVE** license shall require a Licensed Operator to either:

- Maintain NMP-TR-406-F01 OR
- Complete NMP-TR-406-F02 OR NMP-TR-406-F03 OR
- Receive a Nuclear Regulatory Commission (NRC) license within the current calendar quarter.

5.5.5.1 Additionally, an **ACTIVE** license shall require a Licensed Operator to:

- Maintain Medical Certification.
- Maintain Medical Certification for respirator use per the applicable Medical Services procedures.
- Have Dosimetry available.
- Have contacts OR respirator glasses readily available to correct vision to within the limits of ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable.
- Be current in Licensed Operator Continuing Training (LOCT) as demonstrated by showing qualification complete in the Learning Management System (LMS).
- Be current in respirator medical per the LMS qualifications “S-MEDRES49” OR “S-MEDRES50”.
- Be current in Respirator Training per the LMS Qualification.
- Be current in Self-Contained Breathing Apparatus (SCBA) Training per the LMS.

5.5.5.2 IF a Licensed Operator fails to meet the Medical OR Training Requirements above, THEN they may be removed from a shift position that requires an active license until the requirement is met. The Operations Training Group shall notify Operations Management of the required removal from active licensed duties via a telephone call to Line Management followed by a written memo.

5.6 License Reactivation

NOTE
All items shall be completed within the same calendar quarter.

In order to reactivate an RO or SRO license, 10 CFR 55 paragraph 55.53(f) requires:

5.6.1 Before resumption of functions authorized by a license issued under this part, an authorized Representative of the Facility shall certify the following:

That the licensee has completed a minimum of 40 hours of shift functions (i.e., UO or OATC for RO; SS or SM for SRO) under the direction of an Operator or Senior

Operator (i.e., SS or SM) as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the Plant and all required shift turnover procedures with an Operator or Senior Operator. The 40 hours must be on a unit that has fuel in the vessel and be performed in the same calendar quarter. Refer to 2.14 Page 78, Question 277 of NUREG 1262.

5.6.2 The above means that the individual will stand shift with the person in the stated position. The individual reactivating may only be separate from the person who signs for the time credited for infrequent (i.e., 1-2 times in a shift) brief periods OR during plant tours. The Plant Tour is part of the 40 hours of shift functions. At least one (1) shift turnover at the beginning of shift and one (1) at the end of shift must be observed.

5.6.3 Only one (1) individual per licensed position may reactivate under the direction and in the presence of a Licensed Operator or Senior Operator.

5.6.4 The Licensee reactivating shall ensure that entries are made in the Control Room Operator Log for the time period involved in reactivation; including each shift, turnover, and Plant Tour.

5.6.5 Complete NMP-TR-406-F03 of this procedure and return it to the Lead Instructor – Operations Continuing Training OR the Nuclear Operations Training Manager (OTM).

5.6.6 Operations Training Supervision shall forward the form to the Operations Director for approval.

5.6.7 After the Operations Director or designee approves the reactivation form, it shall be returned to the Operations Training Group. Training shall then create a learning event in the LMS for Reactivation. Training shall transmit the original to Document Control.

5.6.8 The Licensed Operator does NOT have to stand any more shifts through the end of the calendar quarter in which they reactivated.

5.6.9 The license will remain active until the Licensed Operator fails to meet the requirements of this procedure to maintain an active license.

5.6.10 All items of NMP-TR-406-F03, up to and including the Operations Director's signature for reactivation approval, shall be completed within the same calendar quarter.

5.7 Reactivation of a Senior Reactor Operator for Supervising Core Alterations

NOTE
Reactivation of the Core Alterations license is only good for one refueling outage and the license shall be de-activated in the LMS at the end of the refueling outage.

In order to reactivate a SRO license for supervising Core Alterations only, NUREG 1021 states:

> The NRCs requirements regarding the conduct of under-instruction or training watches are reflected in 10 CFR 55.13, which allows trainees to manipulate the controls of a facility "under the direction and in the presence of a licensed operator or senior operator…" This position is also evident in the responses to Questions 252 and 276 in NUREG 1262, "Answers to Questions at Public Meetings Regarding Implementation of

Procedure Owner: Robert W. Fletcher / Fleet Operations Training Manager / Corporate
(Print: Name / Title / Site)

Approved By: Original signed by Robert W. Fletcher / 6/30/2014
(Procedure Owner's Signature) (Approval Date)

Special Considerations:

Applicable to Corporate, FNP, HNP, VEGP 1-2, VEGP 3-4

PRB Not Required

This Standardization process control NMP is under the oversight of the Training Peer Team.

Writer(s): Eric Snell

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous:	**NONE**
Reference:	**NONE**
Information:	**ALL**

Procedure Version Description

Version Number	Version Description
1.0	Initial issue of procedure
2.0	Added the Effective Date for Vogtle 3-4 to the cover sheet and to Section 8.0 of the procedure and corrected the typo in Step 6.4.3.1. Reworded Step 6.7.1 to allow the use of a Checklist or JPM.
3.0	This revision added a NOTE prior to Step 6.4.5 and reformatted the procedure to meet NMP-AP-002 updates.
4.0	This revision addresses Plant Hatch Level 2 TEs 516405 and 516395 to provide clarification to the license activation/reactivation process that was discovered during the Causal Analysis.
4.1	This editorial revision adds Plant Vogtle 1-2 to Step 5.7.1.1 which was inadvertently omitted from Version 4.0.
5.0	This revision amends the following: • Adds new reference to NMP-TR-406-F04, "Notification to Supervisor and Medical Services of Changes in Medical Condition for a Licensed Operator" • Adds the requirement for the Licensed Operator to complete NMP-TR-406-F04 for any change in medical conditions
6.0	This revision addresses a Corrective Action for NRC License Medical Condition Reporting issues at Plant Vogtle 1-2 for TE 681782
6.1	This editorial revision provides clarification to Step 5.2.6 based on a review of Frequently Asked Questions for NUREG 1021
6.2	Editorial revision to add the definition of "Personally Identifiable Information (PII)" and add a note to assist with the processing of submittals which contain PII along with sample PII language to be considered for submittal coversheets. Section 8.0 was updated to add PII references. Additionally, an editorial change was made to Attachment 8 to align the attachment with the guidance from NMP-GM-002-001 for initiating a new CR due to changes in medical conditions.

Table of Contents

1.0 PURPOSE

This procedure establishes the processes for obtaining and maintaining Nuclear Regulatory Commission (NRC) licenses including the following:

- Initial License Application
- Active Licensed Duties
- Biennial Licensed Operator Medical Examination
- Evaluation of Changes in the Physical Conditions of Licensed Operators
- License Renewal
- Maintenance of an Active License
- License Reactivation
- Reactivation of a Senior Reactor Operator for Supervising Core Alterations
- Maintenance of a Senior Reactor Operator Licenses Reactivated to Supervise Core Alterations
- Respirator Glasses
- No Solo Operation

2.0 APPLICABILITY

This procedure is applicable to all individuals holding <u>OR</u> pursuing a Reactor Operator (RO) or Senior Reactor Operator (SRO) license.

3.0 DEFINITIONS

3.1 ACTIVE License

An **ACTIVE** license shall require a Licensed Operator to either:

- Maintain NMP-TR-406-F01 <u>OR</u>
- Complete NMP-TR-406-F02 <u>OR</u> NMP-TR-406-F03 <u>OR</u>
- Receive a Nuclear Regulatory Commission (NRC) license within the current calendar quarter.

3.1.1 Additionally, an **ACTIVE** license shall require a Licensed Operator to:

- Maintain Medical Certification.
- Maintain Medical Certification for respirator use per the applicable Medical Services procedures.
- Have Dosimetry available.
- Have contacts OR respirator glasses readily available to correct vision to within the limits of ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable.
- Be current in Licensed Operator Continuing Training (LOCT) as demonstrated by showing qualification complete in the Learning Management System (LMS).
- Be current in respirator medical per the LMS qualifications “S-MEDRES49” OR “S-MEDRES50”.
- Be current in Respirator Training per the LMS Qualification.
- Be current in Self-Contained Breathing Apparatus (SCBA) Training per the LMS.

3.2 Learning Management System (LMS)

A Learning Management System refers to a database maintained by the Training Department and is used to administer computer-based instruction and to determine individual qualification status.

3.3 No Solo

No Solo refers to a licensee with a special NRC license restriction. For example, a typical restriction requires that another person, capable of summoning assistance, be present when the licensee is manipulating controls. Specific requirements may vary and are described in NUREG 1021 as follows:

“An RO who is at risk of sudden incapacitation may have a no-solo restriction that requires another licensed operator to be in view when the restricted operator is performing control manipulations, and someone capable of summoning assistance to be present at all other times while the restricted operator is performing licensed duties. The analogous SRO restriction would require another licensed operator to be in view when the restricted operator is performing control manipulations, and another senior operator to be present on site at all other times while the restricted operator is performing SRO licensed duties or someone capable of summoning assistance to be present at all other times while the restricted operator is performing RO licensed duties.”

3.4 Personally Identifiable Information (PII)

Personally Identifiable Information refers to information that can be used to identify or contact a person uniquely and reliably or can be traced back to a specific individual (i.e., a person’s name in combination with any of the following information, such as relatives’ names, postal address, personal e-mail address, home or cellular telephone number, personal characteristics, Social Security number (SSN), date or place of birth, mother’s maiden name, driver’s license number, bank account information, credit card information, or any information that would make the individual’s identity easily discernible or traceable).

4.0 <u>RESPONSIBILITIES</u>

4.1 The Southern Nuclear Fleet Training Manager is the Corporate Functional Area Manager (CFAM) who shall be accountable for the governance and oversight of this procedure.

4.2 The Site Vice President or Senior Management Representative Onsite shall be responsible for approving NRC Forms and Letters.

4.3 The Training Department Manager is the Site Functional Area Manager (SFAM) who shall be accountable for performance in the use of this procedure.

4.4 The Operations Director shall be accountable to maintain oversight and ensure that License Operators are properly qualified prior to performing Licensed Operator duties.

4.5 The Medical Service Group shall be responsible for the following:

- The completion of the License Operator physical examinations.
- Completing appropriate sections of NRC Form 396.
- Maintaining NRC Form 396 records.
- Determining if a medical condition requires a restriction to be added to a license.
- Updating Medical Items in the Learning Management System (LMS).
- Notifying the Operations Training Group and supervision of any discovered or reported change in medical condition that impacts a License Operator's License.
- Initiating a Condition Report to evaluate a medical condition that may result in a License restriction change. Refer to NMP-TR-406-F04.

4.6 The Operations Training Group/Operations Training Coordinator shall be responsible for the following:

- The processing of NRC Forms.
- Updating the LMS when there is a change to a Licensed Operator's status or training has been completed.
- Tracking of license maintenance hours.

4.7 The Licensed Operator shall be responsible for the following:

- Contacting Medical Services to schedule physical examinations, as applicable.
- Notifying their Supervisor and Medical Services of any change in medical condition, by completing NMP-TR-406-F04.
- Ensuring they are qualified prior to performing Licensed Operator duties.

5.0 PROCEDURE

> NOTE
>
> Documents submitted to the NRC may contain Personally Identifiable Information (PII). In accordance with 10 CFR 2.390, "the submitter shall request withholding at the time the document is submitted."
>
> The cover letter for each submittal containing PII should clearly state that the attached documents contain PII. Additionally, the top of every page of the submittal that contains PII should include the marking "Personally Identifiable Information - Withhold Under 10 CFR 2.390."
>
> Ref. 10 CFR 2.390, NRC Information Notice 2013-22, and Regulatory Issue Summary (RIS) 2007-04.

5.1 Active Licensed Duties

10 CFR 55.4 defines "actively performing the functions of an Operator or Senior Operator" to mean that a Licensed Operator has a position on the Shift Crew that requires the individual to be licensed as defined in the Facility's Technical Specifications, and that the Licensed Operator carries out and is responsible for the duties covered by that position.

For purposes of this procedure, actively performing the functions of an Operator or Senior Operator (i.e., active licensed duties) is defined as follows:

- On-shift as Operator at the Controls (OATC)
- On-shift as Unit Operator (UO)
- On-shift as Shift Supervisor (SS)
- On-shift as Shift Manager (SM)
- Supervision of Core Alterations as SRO

5.2 Biennial Licensed Operator Medical Examination (Refer to Attachment 2)

Biennial is a time period equal to 730 days with the flexibility that the examination can be any time during the anniversary month and therefore slightly longer than 730 days. Every licensed individual shall have a Biennial Medical Examination as follows:

5.2.1 The Medical Services Group shall determine who is due for a physical by running a Trainee Learning Needs Requirement Report in the LMS. The LMS or Medical Service Group shall notify the Licensed Operators of the required Medical Examination using a memo or electronic mail.

5.2.2 The Licensed Operator shall contact the Medical Services Group and schedule their medical examination to ensure that the exam is completed by the last day of the month in which the medical examination is due.

5.2.3 The Medical Services Coordinator or designee shall ensure that the physician completes the medical examination. The physician shall document the findings on the Medical Examination Report (MER).

5.2.4 The Medical Services Coordinator or designee shall complete the appropriate sections of the Licensed Operator's NRC Form 396 and forward it to the Operations Training Group. If the Medical Services Coordinator or designee marks "restriction change from previous submittal" on the NRC Form 396, the Medical Services Coordinator or designee shall notify the Operations Training Group within five (5) working days of the medical examination. Upon completion of the NRC Form 396, the Medical Services Group shall create a Learning Event for the individual for item S-MEDNRC in the LMS for the date the exam was completed.

5.2.5 The Operations Training Group shall review the completed NRC Form 396 for restriction changes from any previous submittal. If a restriction has changed, the Site Vice President or Senior Management Representative shall sign the NRC Form 396. The Operations Training Group shall mail the NRC Form 396 and a letter explaining the changes to the NRC within 30 days (i.e., 25 Administrative day limit) per 10 CFR 55.25.

5.2.5.1 Training shall transmit a copy of the Cover Letter and the completed NRC Form 396 to Medical Services for record processing. Notification of a restriction change shall be via a telephone call to Line Management followed by a written memo to the affected Licensed Operator and to Operations Management. Attachment 1 is provided as an example.

5.2.6 If there is no restriction change, then the NRC Form 396 is NOT required to be signed by Site Vice President or Senior Management Representative.

5.3 Evaluation of Changes in the Physical Conditions of Licensed Operators (Refer to Attachment 3)

5.3.1 Any Licensed Operator shall complete NMP-TR-406-F04 to notify their Supervisor and the Medical Services Group of any change in their physical OR mental condition. The following changes, refer to Attachment 4 for a list of examples, include, but are not limited to:

- Wearing corrective lenses
- Taking prescription medicine
- High blood pressure
- Surgery
- Any condition that restricts the range of motion of an extremity (e.g., back, neck, shoulder, etc.)
- Any mental health condition (i.e., depression, anxiety, etc.) or treatment for drug or alcohol abuse.
- Any condition evaluated by ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable

5.3.2 The Supervisor shall notify the Operations Training Group and the Medical Services Coordinator or designee of any change in the Licensed Operator's physical condition.

5.3.3 The Medical Services Coordinator or designee shall initiate an evaluation of the reported condition for changes in the Licensed Operator's license by utilizing Attachment 8 or for removal from active licensed duties, if necessary. In addition, the Medical Services Coordinator or designee shall generate a Condition Report to ensure timely notification made to NRC.

5.3.4 If the condition results in the need for a permanent change in the Licensed Operator's license, the Medical Services Coordinator or designee shall complete NRC Form 396 and forward it to the Operations Training Group.

5.3.5 IF a restriction has changed, THEN the Operations Training Group shall forward the NRC Form 396 to the Site Vice President or Senior Management Representative for signature. After the Site Vice President or Senior Management Representative approves the NRC Form 396, the approved (i.e., signed) Form and accompanying letter explaining the changes shall be mailed to the NRC. The 10 CFR 55.25 requires this notification be complete within 30 days (i.e., 25 Administrative day limit) of the determination by the physician that the restriction has changed. Training shall transmit a copy of the cover letter and the completed NRC Form 396 to Medical Services for record processing. Notification of a restriction change to the appropriate department will be via a memo or electronic mail.

5.3.6 The Operations Training Group shall notify Operations Management and the individual, via an electronic or hard copy memo, if the condition requires removal from active licensed duties as defined in this procedure. In addition, a learning event shall be entered in the LMS. In the Comment section, record that the block is due to medical status hold.

5.3.7 The Medical Services Coordinator or designee shall notify the Licensed Operator and their Supervisor, per the applicable Medical Services Group procedures, if the condition results in a temporary restriction that does not require removal from active licensed duties.

5.3.8 When the physician restores the Licensed Operator to active licensed duties, then the Medical Services Coordinator or designee shall notify the Operations Training Group. The Operations Training Group shall notify the Licensed Operator and their Supervisor via a memo or electronic mail that the Licensed Operator is restored to active licensed duties. A Learning Event shall be entered in the LMS. In the Comment section, record that the medical hold is removed.

5.4 License Renewal

The Operations Training Group shall initiate renewal of a Licensed Operator's license as required by 10 CFR 55 as follows:

5.4.1 The Operations Training Group shall determine whose license is due for renewal by running a Trainee Learning Needs Requirement Report in the LMS.

5.4.2 The Operations Training Group shall complete the renewal application per the instructions on NRC Form 398 and NUREG 1021 and then forward the application to the Licensed Operator for their review and signature.

5.4.3 After the Licensed Operator signs the NRC Form 398, they shall return it to the Operations Training Group who will then route the NRC Form 398 and the individual's most recent NRC Form 396 to the Training Manager for review and approval.

5.4.3.1 The Training Manager shall sign the NRC Form 398 after the individual and then send the signed form to the Site Vice President or Senior Management Representative.

5.4.4 The Operation Training Group shall mail the original NRC Form 398 and the NRC Form 396, along with a cover letter to the NRC, per 10 CFR 55, at least 30 days (i.e., 25 Administrative day limit) prior to expiration of the Licensed Operator's license. Refer to Attachment 5 for a sample Cover Letter.

NOTE
<u>IF</u> an Operator or Senior Operator applies for a renewal at least 30 days (i.e., 25 Administrative day limit) before the Expiration Date of the existing license, <u>THEN</u> the license does <u>NOT</u> expire until the NRC determines the final disposition of the renewal application.

5.4.5 When the license renewal has been signed and mailed to the NRC in a timely manner (i.e., at least 25 Administrative days prior to expiration) the Operations Training Group shall create a Learning Event for the Licensed Operator for the appropriate item in the LMS for the date the license renewal was submitted.

5.4.6 Upon receipt of the license, renewal from the NRC the Operations Training Group shall edit the Learning Event for the Licensed Operator for the appropriate item in the LMS for the date the license was effective.

5.4.7 The Operations Training Group shall transmit a copy of NRC Form 396 to Medical Services for record processing. The Operations Training Group shall process the cover letter, the NRC Form 398, and the license for records retention.

5.5 Maintenance of an **ACTIVE** License

Only a Licensed Operator with an **ACTIVE** license shall be allowed to manipulate the controls or supervise the manipulation of the controls of the reactor.

5.5.1 Per NUREG 1262 Q. 293, a newly Licensed Operator is considered to have met the proficiency requirements for an active license for the initial calendar quarter in which the license was issued.

5.5.1.1 Upon receipt of a new license, a Learning Event shall be created for the Licensed Operator to give credit for proficiency.

5.5.2 NUREG 1021 states:

"In accordance with 10 CFR 55.53 (e), licensed operators are required to maintain their proficiency by actively performing the functions of an operator or senior operator on at least seven 8-hour or five 12-hour shifts per calendar quarter. This requirement may be completed with a combination of complete 8- and 12-hour shifts (in a position required by the plant's technical specifications) at sites having a mixed shift schedule, and watches shall not be truncated with the minimum quarterly requirement (56 hours) is satisfied. Overtime may be credited if the overtime work is in a position required by the plant's technical specifications. Overtime as an extra "helper" after the official watch has been turned over to another watch stander does not count toward proficiency time."

5.5.2.1 Maintenance of an active license requires that an individual spend seven (7) eight-hour shifts or five (5) twelve-hour shifts in a position that requires the license per the Technical Specifications (i.e., OATC, UO, SS, SM, or SRO during Core Alterations as defined in Technical Specifications) in a calendar quarter.

5.5.2.2 IF an individual stands a combination of complete 12 or eight-hour shifts that total 56 hours in the quarter, THEN this requirement is satisfied. A shift of less than eight (8) hours does NOT count toward the 56-hour total. IF an individual spends this shift time in a position that only requires an RO license (i.e., UO, OATC), THEN they are an active RO only. If they spend this time in an SRO position (i.e., SS, SM) they are an active SRO. IF they spend this time as a SRO in charge of fuel handling during Core Alterations, THEN they are an active SRO only for supervising Core Alterations.

5.5.2.2.1 It is permissible for an individual with an SRO license to maintain only the RO portion of their license in an active state by performing the functions of an RO for a minimum of seven (7) 8-hour OR five (5) 12-hour shifts per calendar quarter pursuant to 10 CFR 55.53(e).

5.5.2.3 In order to maintain the Supervisory portion of the SRO license active, a SRO must stand at least one (1) complete watch per calendar quarter in an SRO-only supervisory position. The remainder of complete watches required in a calendar quarter may be performed in either a credited SRO or RO position. These shifts must be on a unit that has fuel in the vessel. IF a Licensed SRO stands all of their required proficiency watches in an SRO position, THEN the RO portion of the license is still considered active. Performing the required number of shifts per calendar quarter on a single unit maintains the license active for all similar units on an individual's license.

5.5.3 The active Licensed Operator shall complete NMP-TR-406-F01 once per quarter to document these proficiency hours and forward the form to the Operation Training Coordinator.

5.5.4 The Operations Department Training Coordinator or designee shall maintain a record of these hours and create a Learning Event for each Licensed Operator who meets the SRO requirement or for each Licensed Operator who meets the RO requirement. Failure to meet the time requirements for hour's on-shift places that level of license (i.e., RO, SRO) in an "Inactive" status. The Licensed Operator shall NOT be allowed to stand shift in a position that requires that level of license until they have completed reactivation per this procedure. Operations Supervision and the Licensed Operator shall be notified by the Operations Training Group OR the Operations Training Coordinator if the Licensed Operator's license is placed in an 'Inactive' condition.

5.5.5 An **ACTIVE** license shall require a Licensed Operator to either:

- Maintain NMP-TR-406-F01 <u>OR</u>
- Complete NMP-TR-406-F02 <u>OR</u> NMP-TR-406-F03 <u>OR</u>
- Receive a Nuclear Regulatory Commission (NRC) license within the current calendar quarter.

5.5.5.1 Additionally, an **ACTIVE** license shall require a Licensed Operator to:

- Maintain Medical Certification.
- Maintain Medical Certification for respirator use per the applicable Medical Services procedures.
- Have Dosimetry available.
- Have contacts <u>OR</u> respirator glasses readily available to correct vision to within the limits of ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable.
- Be current in Licensed Operator Continuing Training (LOCT) as demonstrated by showing qualification complete in the Learning Management System (LMS).
- Be current in respirator medical per the LMS qualifications “S-MEDRES49” OR “S-MEDRES50”.
- Be current in Respirator Training per the LMS Qualification.
- Be current in Self-Contained Breathing Apparatus (SCBA) Training per the LMS.

5.5.5.2 <u>IF</u> a Licensed Operator fails to meet the Medical <u>OR</u> Training Requirements above, <u>THEN</u> they may be removed from a shift position that requires an active license until the requirement is met. The Operations Training Group shall notify Operations Management of the required removal from active licensed duties via a telephone call to Line Management followed by a written memo.

5.6 License Reactivation

NOTE
All items shall be completed within the same calendar quarter.

In order to reactivate an RO or SRO license, 10 CFR 55 paragraph 55.53(f) requires:

5.6.1 Before resumption of functions authorized by a license issued under this part, an authorized Representative of the Facility shall certify the following:

That the licensee has completed a minimum of 40 hours of shift functions (i.e., UO or OATC for RO; SS or SM for SRO) under the direction of an Operator or Senior

Operator (i.e., SS or SM) as appropriate and in the position to which the individual will be assigned. The 40 hours must have included a complete tour of the Plant and all required shift turnover procedures with an Operator or Senior Operator. The 40 hours must be on a unit that has fuel in the vessel and be performed in the same calendar quarter. Refer to 2.14 Page 78, Question 277 of NUREG 1262.

5.6.2 The above means that the individual will stand shift with the person in the stated position. The individual reactivating may only be separate from the person who signs for the time credited for infrequent (i.e., 1-2 times in a shift) brief periods OR during plant tours. The Plant Tour is part of the 40 hours of shift functions. At least one (1) shift turnover at the beginning of shift and one (1) at the end of shift must be observed.

5.6.3 Only one (1) individual per licensed position may reactivate under the direction and in the presence of a Licensed Operator or Senior Operator.

5.6.4 The Licensee reactivating shall ensure that entries are made in the Control Room Operator Log for the time period involved in reactivation; including each shift, turnover, and Plant Tour.

5.6.5 Complete NMP-TR-406-F03 of this procedure and return it to the Lead Instructor – Operations Continuing Training OR the Nuclear Operations Training Manager (OTM).

5.6.6 Operations Training Supervision shall forward the form to the Operations Director for approval.

5.6.7 After the Operations Director or designee approves the reactivation form, it shall be returned to the Operations Training Group. Training shall then create a learning event in the LMS for Reactivation. Training shall transmit the original to Document Control.

5.6.8 The Licensed Operator does NOT have to stand any more shifts through the end of the calendar quarter in which they reactivated.

5.6.9 The license will remain active until the Licensed Operator fails to meet the requirements of this procedure to maintain an active license.

5.6.10 All items of NMP-TR-406-F03, up to and including the Operations Director's signature for reactivation approval, shall be completed within the same calendar quarter.

5.7 Reactivation of a Senior Reactor Operator for Supervising Core Alterations

NOTE
Reactivation of the Core Alterations license is only good for one refueling outage and the license shall be de-activated in the LMS at the end of the refueling outage.

In order to reactivate a SRO license for supervising Core Alterations only, NUREG 1021 states:

> The NRCs requirements regarding the conduct of under-instruction or training watches are reflected in 10 CFR 55.13, which allows trainees to manipulate the controls of a facility "under the direction and in the presence of a licensed operator or senior operator…" This position is also evident in the responses to Questions 252 and 276 in NUREG 1262, "Answers to Questions at Public Meetings Regarding Implementation of

Title 10, Code of Federal Regulations, Part 55 on Operators' Licenses," which indicate that a trainee's activities are to be closely monitored by the responsible person.

5.7.1 If a facility licensee needs to reactivate a regular SRO license for the purpose of supervising refueling activities, the Operator must complete one of the following two (2) options:

5.7.1.1 Option 1 (Plants Hatch and Vogtle 1-2 Only) – Review Core Alterations training material, or receive Instructor lead training, and completion of Site-specific requirement(s) (i.e. JPM or Checklist). This option will ideally be completed no more than one (1) week prior to a refueling outage.

5.7.1.2 Option 2 – The Operator must complete one (1) shift under direction on the refueling floor, as discussed above, and the facility licensee must ensure that the Operator is administratively restricted from performing full SRO duties. Only one (1) individual at a time may reactivate under the direction and in the presence of the Senior Operator in charge of refueling.

5.7.2 The individual reactivating shall ensure entries are made in the Control Room Operator Log for the time period involved in reactivation.

5.7.3 Complete NMP-TR-406-F02 of this procedure and return it to the Operations Training Group.

5.7.4 After the requirements of NMP-TR-406-F02 are completed, Training shall transmit the original to Document Control.

5.8 Respirator Glasses

5.8.1 Any Licensed Operator whose license is restricted to corrective lenses shall have contacts <u>OR</u> respirator glasses readily available to correct their vision to within the limits of ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable, while performing licensed duties.

5.8.2 When a Licensed Operator is initially restricted to corrective lenses, the Operations Training Group shall be notified by Operations via a memo or electronic mail, and this memo will state the need for respirator lenses <u>OR</u> contacts.

5.8.3 It shall be the responsibility of the Licensed Operator to ensure that the respirator glasses <u>OR</u> contacts will bring them into compliance with the requirements of ANSI 3.4, 1983 or ANSI 3.4, 1996, as applicable.

5.9 No Solo Operation

5.9.1 <u>IF</u> a Licensed Operator's license is amended to "No Solo Operation", <u>THEN</u> the No-Solo requirements shall be met for them to perform active license duties.

5.9.2 A No Solo SRO may be Fueling Handling Supervisor as long as the crew is made aware of the content and implication of the No Solo restriction. However, another SRO shall be onsite and someone capable of summoning the SRO shall be in view of the No Solo SRO.

5.10 Nuclear Regulatory Commission Correspondence

5.10.1 Correspondence to the NRC related to licensee medical condition change, recommended license restriction change, or license renewal applications shall be dispatched in a manner that provides verification of receipt. The verification documentation shall be included in the disposition of Condition Report or Technical Evaluation.

5.10.2 When verification is received that correspondence has been delivered, it is recommended that a follow-up phone call be made to confirm.

5.11 License Termination

5.11.1 When an individual no longer requires a license, the NRC shall be requested in 30 (i.e., 25 Administrative day limit) days to terminate the individual's license. Refer to Attachment 6 for a sample letter.

5.11.2 The Operations Training Supervisor or designee shall initiate a Condition Report once notified to terminate an Operator License to track the 30-day notification.

5.11.3 The Operations Training Supervisor or designee shall ensure that the LMS is updated.

5.12 Initial License Application

5.12.1 Operations Training Supervision shall ensure that the NRC License Operator Medical Exam is scheduled to occur within approximately six (6) months of the anticipated licensing date (i.e., typically one (1) month after the license exam).

5.12.2 Operations Training Supervision shall furnish the information to complete the NRC Form 398, referring to NRC Form Instructions approximately 45 days before the scheduled license exam.

5.12.3 Medical Services shall complete NRC Form 396 approximately 45 days before the scheduled license exam.

5.12.4 Operations Training Supervision shall develop NRC Form 396s (e.g., name, etc.) and transmit them to Medical Services for completion.

5.12.5 Operations Training Supervision shall develop NRC Form 398s when information is received.

5.12.6 The Initial License Training (ILT) Lead Instructor <u>AND</u> OTM shall review NRC Form 396s and 398s for accuracy.

5.12.7 Operations Training Supervision shall submit the preliminary unsigned NRC Form 396s and 398s to the NRC Examiner at least 30 days (i.e., 25 Administrative day limit) prior to the scheduled examination date. (Required NRC Date)

5.12.8 Operations Training Supervision shall make any corrections and submit the NRC Form 398s to the license holder for signature.

5.12.9 Operations Training Supervision shall obtain the Training Manager's signature on NRC Form 398s.

5.12.10 Operations Training Supervision shall develop a cover letter, and then submit cover letter, NRC Form 398s and 396s to the Site Vice President or Senior Management Representative for signature.

5.12.11 The Operations Training Group shall transmit a copy of NRC Form 396 to Medical Services for record processing and shall process the cover letter and the NRC Form 398 for records retention.

5.12.12 Operations Training Supervision shall mail the package by certified mail with return receipt to the NRC at least 14 days prior to NRC license exam date. (Required NRC Date)

5.12.13 When the NRC License is received from the NRC, Operations Training Supervision shall ensure that the LMS is updated with the information (i.e., expiration date and license number).

6.0 RECORDS

6.1 All records shall be maintained per appropriate site procedures.

QA Records (X)	Non-QA Records (X)	Record Generated	R-Type
X		Active License Maintenance	TR0.001
X		Senior Reactor Operator Reactivation for Core Alterations	TR0.001
X		License Reactivation Documentation	TR0.001
X		Notification to Supervisor and Medical Services of Changes in Medical Condition for a Licensed Operator	TR0.001
	X	Medical Conditions Routing Checklist	N/A

7.0 COMMITMENTS

7.1 Farley

None

7.2 Hatch

None

7.3 Vogtle 1-2

None

7.4 Vogtle 3-4

None

8.0 REFERENCES

8.1 ANSI 3.4, 1983, “Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants”

8.2 ANSI 3.4, 1996, “Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants”

8.3 10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding”

8.4 10 CFR 50, “Domestic Licensing of Production and Utilization Facilities”

8.5 10 CFR 55, “Operator’s Licenses”

8.6 10 CFR 55.4, “Operator’s Licenses – Definitions”

8.7 10 CFR 55.25, “Incapacitation because of Disability or Illness”

8.8 Meeting Summary – NRC Region II Annual Training Managers’ Conference, Tom Peebles to George Hairston, June 2, 1995

8.9 MS-MED-004, “NRC Licensed Operator Physical Examination”

8.10 MS-MED-005, “NRC Licensed Operator Physical Examination – ANSI/ANS 3.4 1996”

8.11 NEI 99-02, “Regulatory Assessment Performance Indicator Guideline”

8.12 NMP-TR-406-F01, “Active License Maintenance”

8.13 NMP-TR-406-F02, “Senior Reactor Operator Reactivation for Core Alterations”

8.14 NMP-TR-406-F03, “License Activation Documentation”

8.15 NMP-TR-406-F04, “Notification to Supervisor and Medical Services of Changes in Medical Condition for a Licensed Operator”

8.16 NRC Information Notice IN 91-08, “Medical Examinations for Licensed Operators”

8.17 NRC Information Notice IN 94-14 including Supplement 1, “Failure to Implement Requirements for Biennial Medical Examinations and Notification to the NRC of Changes in Licensed Operator Medical Conditions”

8.18 NRC Information Notice 97-66, “Failure to Provide Special Lenses for Operators Using Respirator or Self-Contained Breathing Apparatus During Emergency Operations”

8.19 NRC Information Notice 99-05, “Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration”

8.20 NRC Information Notice 2013-22, “Recent Licensing Submittals Containing Personally Identifiable Information”

8.21 NRC Form 396, “Certification of Medical Examination by Facility Licensee”

8.22 NRC Form 398, “Personal Qualification Statement – Licensee”

8.23 NUREG 1021, “Operator Licensing Examiner Standards for Power Reactors”

8.24 NUREG 1262, “Answers to Questions at Public Meetings Regarding Implementation of Title 10, Code of Federal Regulations, Part 55 on Operators’ License”

8.25 Regulatory Issue Summary (RIS) 2007-04, “Personally Identifiable Information Submitted to the U.S. Nuclear Regulatory Commission”

8.26 Regulatory Issue Summary (RIS) 2007-29, “Clarified Guidance for Licensed Operator Watch-Standing Proficiency”

8.27 SH-GEN-039, “NRC Licensed Operator Physical Examination”

8.28 U.S. Nuclear Regulatory Commission Regulatory Guide 1.134, “Medical Evaluation of Licensed Personnel at Nuclear Power Plants”

Attachment 1

Nuclear Regulatory Commission Medical Restriction Change Letter Sample

XXXXX (insert name of Regional Administrator, Region II)
The Regional Administrator, Region II
United States Nuclear Regulatory Commission
245 Peachtree Center Avenue, NE Suite 1200
Atlanta, Georgia 30303-1257

Attention: Mr. XXXX XXXXX :

Dear XXXXX: (Insert name of Regional Administrator, Region II)

This letter is to inform you that NAME, docket number 55-XXXXX and license number OP-XXXXX-X, has had a change in his medical condition as indicated on the attached per ANSI/ANS 3.4, 1983.

This information is regarded as Privacy Act Information and should be treated as such.

Sincerely,

X X. XXXXXX
Site Vice President

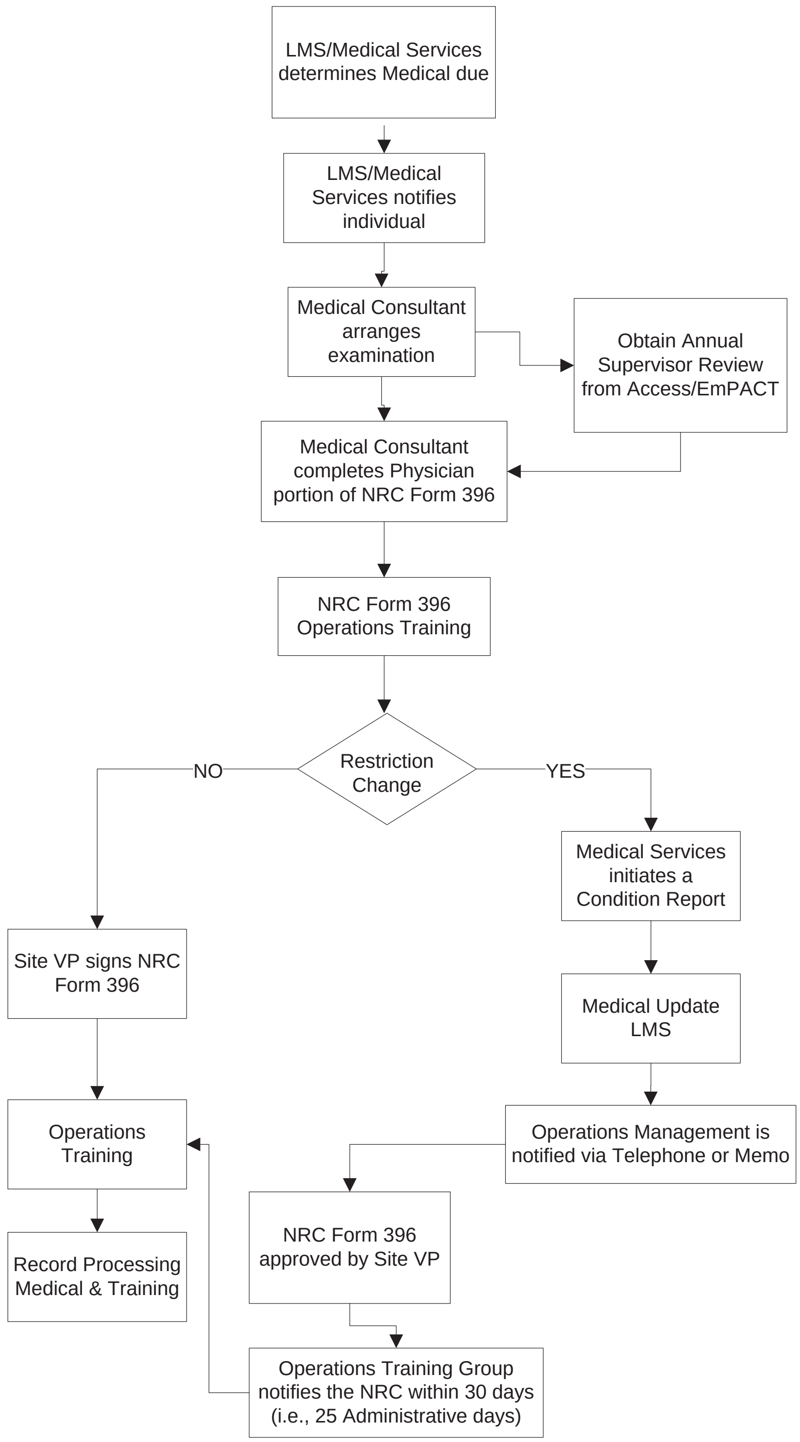
JRJ/DDH:las

cc: Site Training Manager (without attachments)
Site Occupational Health Svcs. Lead Spc. (without attachments)
INDIVIDUAL (with attachments)

Attached documents contain Personally Identifiable Information (PII)

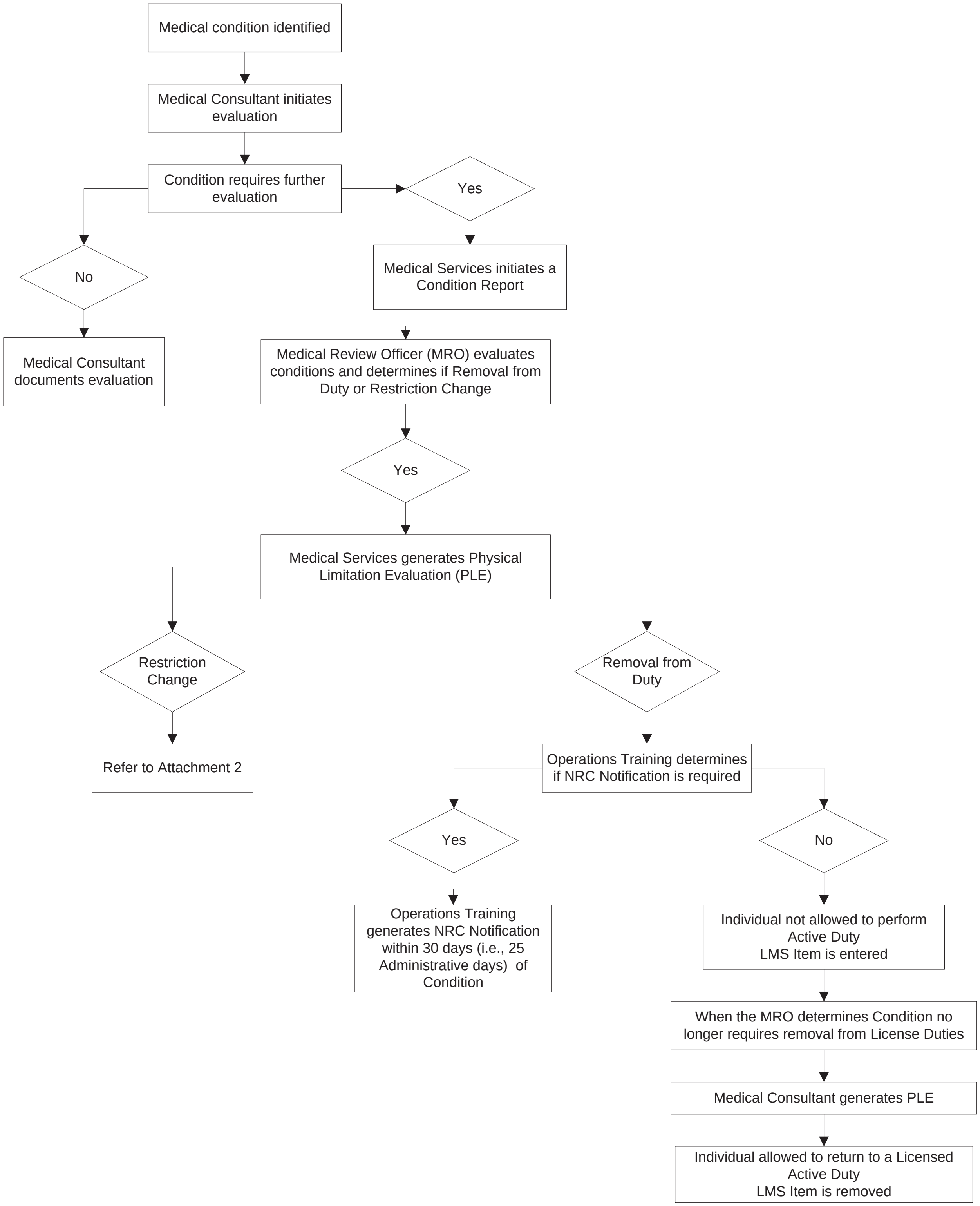
Attachment 2

Routine Nuclear Regulatory Commission License Medical Examination Flowpath



Attachment 3

Medical Condition Flowpath



Attachment 4

Medical Conditions List

Intra-Company Correspondence

TO: Licensed Operators

Listed below are some of the Medical Conditions that Licensed Operators shall be aware of that can or may have an affect on their license status and require evaluation by Medical Services:

> NOTE
>
> These guidelines are NOT all-inclusive and are based on ANSI/ANS 3.4-1983. There are many situations and conditions that could affect a person and the licensee should remember that they are the one ultimately responsible for the "Safe Operation of the Nuclear Power Plant". Any change in your medical condition shall be reported to the Medical Department.

> NOTE
>
> Please contact the Medical Services Coordinator/designee or the Lead Instructor – Operations Continuing Training at any time if you have questions concerning the above.

1. Asthma
2. Emphysema
3. Heart Attack or any other heart condition including High Blood Pressure
4. Diabetes (Use of insulin shall disqualify Operator for Solo Operation)
5. Medical conditions requiring frequent or continued use of steroids (i.e., Lupus, Arthritis, or any other Collagen disorder)
6. Severe recurrent dermatitis (e.g., skin rashes) which could interfere with the wearing of personal protective equipment or decontamination procedures
7. Any abnormal blood work (i.e., blood studies that might indicate Anemia, infection, etc.)
8. Any form of cancer
9. Seizures (A history of Epilepsy shall disqualify the Operator for Solo Operation)
10. Depression, anxiety attacks, etc.
11. Any prescription medication

> NOTE
>
> Medications that can be taken safely and without adverse side affects by one person may have the opposite effect on another, thus evaluating the effect of medication must be done on an individual basis.

12. Any medication taken that could have an effect on an Operator's ability to perform (i.e., cold medication such as Drixoral, Actifed, etc. that could cause drowsiness, pain medication such as Tylenol #3, Demerol, Darvocet, etc. tranquilizers and anti-depressants such as Elavil, Sinequan, Prozac, Xanax, etc.; antihistamines such as Benadryl, Atarax, etc., and narcotics.
13. Any change in vision (i.e., new corrective lenses, prescription, being told by a doctor that corrective lenses are needed, Radial Keratotomy, etc.)
14. Any loss of mobility (i.e., sprained ankle, which may cause an Operator to need crutches, a hand or wrist splint/cast, etc.)
15. Sleep Apnea
16. Any other condition that could affect an Operator's ability to perform "under normal, abnormal, and emergency conditions".
17. Any allergic reaction that requires the use of an Epi Pen.
18. Any medical condition that requires regular follow up with a health care professional.

Attachment 5

License Renewal Letter
Sample

May 2, 2005

XXXX (insert name of Regional Administrator, Region II)
The Regional Administrator, Region II
United States Nuclear Regulatory Commission
245 Peachtree Center Avenue, NE Suite 1200
Atlanta, Georgia 30303-1257

Attention: Mr. XXXX XXXXX: (insert the name of Chief, Operations Branch)

Dear XXXXX: (insert the name of Regional Administrator, Region II)

Please find attached the Personal Qualification Statement--Licensee, NRC Form 398 and the Certification of Medical Examination by Facility Licensee, NRC Form 396, for Mr. Xxx Xxxxxxx Docket Number 55-XXXXX, License Number SOP-XXXXX-X, and Mr. Xxxx X. Xxxxx, Docket Number 55-XXXXX, and License Number OP-XXXXX-X.

If you have any questions about these documents, please contact Training Supervision at XXX-XXX-XXXX.

Sincerely,

X.X. XXXXXX
Site Vice President

JRJ/DDH:las

Enclosure

cc: Site Training Manager (without attachments)
File

Attached documents contain Personally Identifiable Information (PII)

Attachment 6

License Termination Letter
Sample

June 28, 2010

Mr. XXXXX XXXXXXX
Regional Administrator, Region II
United States Nuclear Regulatory Commission
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, GA 30303-1257
Attention: Mr. XXXXX XXXXXXX

Dear Mr. XXXXXX:

This letter is to notify you that Southern Nuclear has determined that the individual listed below no longer has a need to maintain an operating license and therefore request that their license be terminated.

Mr. XXXX XXXX	Docket 55-XXXXX	License OP-XXXXX

If you have any questions regarding this matter, please contact Training Supervision at XXX-XXX-XXXX.

Sincerely,

____
XXXXXXX
Site Vice President

cc: Lead Instructor – Continuing
File

Attachment 7

Initial License Application Letter
Sample

May 29, 2012

XXXX (insert the name of Regional Administrator, Region II)
The Regional Administrator, Region II
United States Nuclear Regulatory Commission
245 Peachtree Center Avenue, NE Suite 1200
Atlanta, Georgia 30303-1257

Attention: Mr. XXXXX XXXXXXX: (insert the name of Chief, Operations Branch)

Subject: License Examination Request and Transmittal of Final Applications

Dear Mr. XXXXXX: (insert the name of Regional Administrator, Region II)

In accordance with NUREG 1021, please find attached the Final License Applications, NRC Form 398s-Personal Qualification Statement and the Certification of Medical Examination by Facility Licensee-NRC Form 396, for the following personnel, submitting request for License Examination on mmmm ddth, yyyy at pppp:

NAME	License Applying For	Docket#
Name:	RO	55-XXXXX
Name:	SRO-U	55-XXXXX
Name:	SRO-I	55-XXXXX

If you have any questions about these documents, please contact Training Supervision at XXX-XXX-XXXX.

Sincerely,

X.X. XXXXX
Site Vice President

Enclosures

cc: Site Training Manager (without attachments)
File

Attached documents contain Personally Identifiable Information (PII)

Attachment 8

Medical Conditions Routing Checklist

Plant: ☐ Farley ☐ Hatch ☐ Vogtle 1-2 ☐ Vogtle 3-4

Medical Services shall initiate this form upon notification by a Licensed Operator of a change in medical status.

____

Licensed Operator Name/LMS ID Contact Number Date

> NOTE
>
> Nuclear Regulatory Commission (NRC) Form 396 must be submitted to the NRC within 30 days (i.e., 25 Administrative day limit) of a Medical Services diagnosis of a change in an individual's medical condition.

1. NMP-TR-406-F04 has been initiated.

____

Medical Services Signature Date

2. Log Entry in the Occupational Health Management (OHM) Program.

____

Medical Services Signature Date

3. Requires information from a personal doctor? ☐ Yes ☐ No

If Yes, the information has been received.

____

Medical Services Signature Date

4. The Medical Services Evaluation has been performed.

____

Medical Services Signature Date

Attachment 8

Medical Conditions Routing Checklist

5. New restriction or a change to an existing restriction? ☐ Yes ☐ No

If Yes, continue use of this form.
If No, discontinue using this Checklist and maintain in medical record.

6. Generate a Condition Report (CR) with the following comments in the remarks field "Event Code 21C, Generate two (2) Technical Evaluations (TEs) – One with a 15 day due date and one with a 25 day due date".

CR#: ______________________________

____
Medical Services Signature Date

7. A Medical Director review and concurrence has been received.

____
Medical Services Signature Date

8. The NRC Form 396 and a Medical Review Officer (MRO) letter have been routed to Operations Training.

____
Medical Services Signature Date

9. Operations Training has received the NRC Form 396.

____
Operations Training Staff Signature Date

10. The Cover Letter has been prepared. ____
Operations Training Staff Date

Attachment 8

Medical Conditions Routing Checklist

11. Senior Management Representative Onsite has obtained a signature.

Operations Training Staff Signature ______ Date

12. A copy of the MRO letter and the NRC Form 396 has been sent by certified mail to the NRC.

Operations Training Staff Signature ______ Date

Certified Mail Tracking Number: ____________________

NOTE
Retain this Routing Form with the individual's Training Record until getting receipt of the NRC letter.

13. Operations Training shall provide a copy of NRC Form 396 and a cover letter to the Medical Services Group for inclusion in the medical file.

Operations Training Staff Signature ______ Date

14. A Letter has been received from the NRC with amended license restrictions, if applicable.

Update the LMS, if applicable: ____________________
Operations Training Staff ______ Date

A copy has been retained in Operations Training file: ____________________
Operations Training Staff Signature ______ Date

A copy has been provided to the Licensed Operator: ____________________
Operations Training Staff Signature ______ Date

The original has been provided to Medical Services for the individual's medical file.

Operations Training Staff Signature ______ Date

A.2 RO

TITLE: MOD - Perform A Quadrant Power Tilt Ratio Calculation
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 20 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☐ TIME CRITICAL ☐ PRA

JPM DIRECTIONS:
1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Do not allow the students to use the Core Physics curves on the classroom computer, use the Core Physics curves provided in the HANDOUT.

TASK STANDARD: Upon successful completion of this JPM, the examinee will:
1. Correctly determine the QPTR.
2. Correctly determine whether or not the QPTR meets acceptance criteria

Examinee:				
Overall JPM Performance:	**Satisfactory**	☐	**Unsatisfactory**	☐
Evaluator Comments **(attach additional sheets if necessary)**				

EXAMINER: ____________________________

Developer	S. Jackson	Date: 4/3/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to PERFORM A QUADRANT POWER TILT RATIO CALCULATION. The conditions under which this task is to be performed are:

a. N-41, N-42, & N-43 PR NI detectors are OPERABLE.
b. N-44 PR NI detector is INOPERABLE.
c. You are directed by Shift Supervisor to perform STP-7.0, using curves 71A-D and pictures provided, and determine if the acceptance criteria is met.
d. The IPC and QPTR computer spreadsheet are not available.
e. A DVM will NOT be used to collect data.
f. A pre-job brief is not required.

EVALUATION CHECKLIST

	ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)

_____ START TIME

NOTE: Critical to use the correct 0% AFD values from curves.

*1.	Obtain normalized currents from curves 71A, 71B, & 71C.	Obtains normalized current values (Curve 71A-C) and records them on Attachment 1 of STP-7.0.	S / U
*2.	Record data for power range detector A and detector B from Data sheet 2.	Values from PRNI pictures for detector A and detector B of NI-41, 42, & 43 displays recorded on Attachment 1 of STP-7.0.	S / U
*3.	Calculate upper and lower quadrant power tilt ratios.	Upper ratio calculated at 1.01 to 1.014 Lower ratio calculated at 1.01 to 1.02	S / U

NOTE: Depending on how rounding is performed in the calculation, both upper and lower ratios may be equal.

*4.	Enter the greater of the upper or lower quadrant power tilt ratio.	Greater of the above two values Lower: 1.01 to 1.02 entered.	S / U
5.	Records power level.	Current avg power level recorded.	S / U
*6.	Determines acceptance criteria **MET**.	Determination made that acceptance criteria is **MET**.	S / U

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
7. Reports to Shift Supervisor that acceptance criteria is met.	Reports to Shift Supervisor that acceptance criteria is **MET.** (CUE: Shift Supervisor acknowledges).	S / U
8. Fills out Surveillance Test Review sheet per attached key.	Fills out Surveillance Test Review sheet per attached key.	S / U

_____ **STOP TIME**

Terminate when assessment of acceptance criteria is performed.

CRITICAL ELEMENTS: Critical Elements are denoted with an asterisk (∗) preceding the element number.

GENERAL REFERENCES

1. FNP-1-STP-7.0, Version 23.0
2. Core Physics curves 71A-D Rev. 36.0
3. K/A: G2.1.12 – 3.7 / 4.1

GENERAL TOOLS AND EQUIPMENT

1. Calculator
2. STP-7.0
3. Core Physics curves 71A-D
4. Pictures of PRNI's.

Critical ELEMENT justification:

STEP	Evaluation
1-4	**Critical:** Task completion: required to properly determine QTPR.
5	**Not Critical:** Does not determine the calculation nor the acceptance criteria.
6	**Critical:** Task completion: Must decide whether or not acceptance criteria is met.
7-8	**Not Critical:** Does not determine the calculation nor the acceptance criteria.

COMMENTS:

CONDITIONS

When I tell you to begin, you are to PERFORM A QUADRANT POWER TILT RATIO CALCULATION. The conditions under which this task is to be performed are:

a. N-41, N-42, & N-43 PR NI detectors are OPERABLE.
b. N-44 PR NI detector is INOPERABLE.
c. You are directed by Shift Supervisor to perform STP-7.0, using curves 71A-D and pictures provided, and determine if the acceptance criteria is met.
d. The IPC and QPTR computer spreadsheet are not available.
e. A DVM will NOT be used to collect data.
f. A pre-job brief is not required.

KEY

FARLEY
Unit 1

SAFETY RELATED

FNP-1-STP-7.0

Quadrant Power Tilt Ratio Calculation

VERSION 23.0

Special Considerations:

This is an upgraded procedure. Exercise increased awareness during initial use due to potential technical and/or sequential changes. After initial use, provide comments to the procedure upgrade team.

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous	**ALL**
Reference	**NONE**
Information	**NONE**

Approval: David L Reed (Approved By) 10/11/13 (Date)

Effective Date:

OPERATIONS
Responsible Department

KEY

VERSION SUMMARY
PVR 23.0 DESCRIPTION
Updated to fleet template and writer's guide

KEY

TABLE OF CONTENTS

1.0 PURPOSE

- To determine the quadrant power tilt ratio using power range nuclear instrumentation.
- Acceptance Criteria for this test is the quadrant power tilt ratio shall be ≤ 1.020.

2.0 PRECAUTIONS AND LIMITATIONS

1. Reactor power, rod position and reactor coolant temperature should be constant while taking data. ☐
2. A QPTR calculation should be done prior to rescaling of Power Range Nuclear Instruments, and after completing the rescaling of ALL Power Ranges Nuclear Instruments. A QPTR calculation performed between individual Power Range rescaling may provide erroneous results. ☐
3. IF one Power Range NI is inoperable AND thermal power is ≤ 75% RTP, the remaining power range channels may be used for calculating QPTR. **(SR 3.2.4.1)** ☐
4. Above 75% RTP, with one Power Range NI inoperable, QPTR must be determined by SR 3.2.4.2. ☐
5. The SM/SS shall be notified if any acceptance criteria are NOT satisfied. ☐

3.0 INITIAL CONDITIONS

1. The version of this procedure has been verified to be the current version. **(OR 1-98-498)** SJJ
2. This procedure has been verified to be the correct procedure for the task. **(OR 1-98-498)** SJJ
3. This procedure has been verified to be the correct unit for the task. **(OR 1-98-498)** SJJ

NOTE

This STP may be performed at less than 50% power for verification of power range instrument indications. In this case, the STP is NOT for surveillance credit. ☐

4. Unit 1 is above 50% of rated thermal power. SJJ
5. IF DVM is used to collect data, I&C has obtained a Fluke 45 or equivalent with shielded test leads with NO exposed metal connectors. N/A

DVM Serial number ______________ Cal. due date ____________

KEY

3.0 INITIAL CONDITIONS (continued)

6. This procedure may contain previously evaluated Critical Steps that may not be applicable in certain plant conditions. The evaluation of this procedure for Critical Steps is performed during the Pre-Job briefing. The decision concerning how to address error precursors for critical steps should be governed by NMP-GM-005-GL03, Human Performance Tools. SJJ ______

NOTE
Asterisked (*) steps are those associated with Acceptance Criteria. ☐

4.0 INSTRUCTIONS

N/A

4.1 QPTR Determination Using The IPC.

NOTES
Section 4.2, QPTR Determination Using Manual Calculation: should be used to calculate QPTR when the IPC QPTR application is unavailable. ☐

1. **Open** the QPTR AND TILT FACTORS application on the IPC Applications Menu. ______

2. **Check** the following:

   - UPPER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

   - LOWER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

SJJ

3. IF QPTR data is NOT GOOD quality, **go to** Section 4.2, QPTR Determination Using Manual Calculation: ______

4. IF QPTR data is GOOD quality, perform the following:

   a. **Click** PRINT EXCORE REPORT button. ______

   b. **Include** printed Excore Report with this procedure. ______

   c. **Go to** Section 4.3. ______

N/A

KEY

KEY

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE and THERMAL POWER ≤ 75% RTP, the remaining three power range channels may be used for calculating QPTR. ☐

4.2 QPTR Determination Using Manual Calculation:

1. **Calculate** QPTR using Attachment 1, Quadrant Power Tilt Ratio Calculation without Plant Computer SJJ

2. **Go to** Section 4.3. SJJ

4.3 Determination Of QPTR Acceptance Criteria:

NOTE

QPTR value displayed by the IPC utilizes 3 decimal places (to the thousandths place). If the QPTR value displayed is, for example 1.021, this would exceed the limit of 1.02 and require performance of the LCO 3.2.4 Condition A Required Actions.
(NL-10-0406, dated 2/26/2010) ☐

1. ***Check** Excore Maximum Quadrant Power Tilt Ratio ≤ 1.020 on either the EXCORE REPORT OR Attachment 1. SJJ

ACCEPTANCE CRITERIA

Maximum value of UPPER or LOWER Quadrant Power Tilt Ratio shall be ≤ 1.020.

KEY

KEY

NOTE
Asterisked (*) steps are those associated with Acceptance Criteria.

5.0 ACCEPTANCE CRITERIA

The quadrant power tilt ratio shall be ≤ 1.020.

6.0 RECORDS

Documents created using this procedure will become QA Records when completed unless otherwise stated. The procedures and documents are considered complete when issued in DMS.

QA Record (X)	Non-QA Record (X)	Record Generated	Retention Time	R-Type
X		FNP-1-STP-7.0	LP	H06.045

7.0 REFERENCES

- FSAR - Chapter 4.4.2.4
- Technical Specification 3.2.4

KEY

KEY

ATTACHMENT 1
Page 1 of 5

Quadrant Power Tilt Ratio Calculation without Plant Computer

NOTE
QPTR may be determined using normalized currents from Curves 71A, 71B, 71C, 71D AND either of the following:
- Indicated detector current meter data. ☐
- Detector currents read by DVM using Attachment 2. ☐

1. **Obtain** normalized currents from Curve 71(A, B, C, D). SJJ

2. **Enter** normalized currents from Curve 71 on the Calculation Sheet. SJJ

NOTE
With input from one Power Range Neutron Flux channel INOPERABLE AND THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. ☐

3. **Perform** the following:

   a. IF available, **enter** detector currents indicated on POWER RANGE B drawer meters on the Calculation Sheet for each of the following:
      - N1C55NI0041, N41B DETECTOR A, (Upper) ☐
      - N1C55NI0041, N41B DETECTOR B, (Lower) ☐
      - N1C55NI0042, N42B DETECTOR A, (Upper) ☐
      - N1C55NI0042, N42B DETECTOR B, (Lower) ☐
      - N1C55NI0043, N43B DETECTOR A, (Upper) ☐
      - N1C55NI0043, N43B DETECTOR B, (Lower) ☐
      - N1C55NI0044, N44B DETECTOR A, (Upper) ☐
      - N1C55NI0044, N44B DETECTOR B, (Lower) ☐

CAUTION
DVM readings may be taken in only one drawer at a time. ☐

   b. IF any NI current reading not available on the POWER RANGE B drawer, **enter** detector currents obtained by I&C using Attachment 2 for the affected detectors. N/A

4. **Enter** total number of operable detectors in space provided on the Calculation Sheet. SJJ

KEY

KEY

ATTACHMENT 1
Page 2 of 5

Quadrant Power Tilt Ratio Calculation without Plant Computer

5. **Calculate** the following:
   - Upper Quadrant Power Tilt Ratio. SJJ
   - Lower Quadrant Power Tilt Ratio. SJJ

6. ***Record** the greater of the upper or lower Quadrant Power Tilt Ratio value in the space provided on the Calculation Sheet. SJJ

ACCEPTANCE CRITERIA
Maximum value of upper or lower Quadrant Power Tilt Ratio shall be ≤ 1.020.

7. **Record** the Power Level (Avg) in the space provided. SJJ

KEY

KEY

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

UPPER QUADRANT POWER TILT

POWER RANGE B Drawer	UPPER DET Indicated Current	÷	*UPPER DET 100% Current	=	UPPER DET Calibrated Output
N41	Detector A 124.3	÷	N41T 187.44	=	0.663
N42	Detector A 128.5	÷	N42T 191.11	=	0.672
N43	Detector A 126.0	÷	N43T 185.03	=	0.681
N44	Detector A N/A	÷	N44T N/A	=	N/A

				Total Number Operable Upper Detectors		1 / Average Upper Detector Calibrated Output	X	Maximum Upper Detector Calibrated Output	=	Upper Quadrant Power Tilt Ratio
Total Upper Detector Calibrated Output	=	2.016	÷	3	=	1 / 0.672	X	0.681		1.01 to 1.014

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

KEY

KEY

ATTACHMENT 1
Page 4 of 5

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

LOWER QUADRANT POWER TILT

POWER RANGE B Drawer	LOWER DET Indicated Current	÷	*LOWER DET 100% Current	=	LOWER DET Calibrated Output
N41	Detector B 128.1	÷	N41B 185.63	=	0.690
N42	Detector B 129.6	÷	N42B 186.84	=	0.694
N43	Detector B 126.7	÷	N43B 191.51	=	0.662
N44	Detector B N/A	÷	N44B N/A	=	N/A

	=		÷	Total Number Operable Lower Detectors	=	1 / Average Lower Detector Calibrated Output	X	Maximum Lower Detector Calibrated Output	=	Lower Quadrant Power Tilt Ratio
Total Lower Detector Calibrated Output	=	2.046	÷	3	=	1 / 0.682	X	0.694	=	1.01 to 1.02

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

KEY

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Upper QPTR	Lower QPTR
1.01 to 1.014	1.01 to 1.02

Maximum of Upper or Lower QPTR

1.01 to 1.02

ACCEPTANCE CRITERIA
Maximum of Upper or Lower Quadrant Power Tilt Ratio does not exceed 1.020.

% Reactor Power 72 - 73%

Both may be equal depending on how rounding is done.

Using A DVM To Obtain Detector Current Values

NOTE
Detector current values may be obtained for as many drawers as required. Unused spaces in the Table should be marked N/A. ☐

CAUTIONS
- DVM readings may be taken in only one drawer at a time. ☐
- A Fluke 8600 shall NOT be used to obtain currents ☐

1. Using a Fluke 45 or equivalent AND shielded test leads **connect** to obtain detector voltage readings as follows:

NOTE
Voltage values should be in the 2 to 3 volt range. ☐

a. For Upper Detector **connect** to TP301 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

b. For Lower Detector **connect** to TP302 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

KEY

ATTACHMENT 2
Page 2 of 2

Using A DVM To Obtain Detector Current Values

NOTE

The following formula is used to calculate detector currents:

$$\frac{\text{Measured Detector Voltage}}{2.083} \times \text{Curve 71 "0\% AFD, 100\% Current" Value} = \text{Calculated Detector Current}$$ ☐

2. Using the 0% AFD, 100% current value from Curve 71, **perform** the following:

   a. **Calculate** the detector current value. _______

   b. **Record** in appropriate space of table below. _______

	N41		N42		N43		N44	
	Upper Detector A N41T	Lower Detector B N41B	Upper Detector A N42T	Lower Detector B N42B	Upper Detector A N43T	Lower Detector B N43B	Upper Detector A N44T	Lower Detector B N44B
	DVM Voltage		DVM Voltage		DVM Voltage		DVM Voltage	
Step 1								
	Calculated Current		Calculated Current		Calculated Current		Calculated Current	
Step 2								

KEY

KEY

ATTACHMENT 3
Page 1 of 1

Surveillance Test Review Sheet

TECHNICAL SPECIFICATION REFERENCE	MODE(S) REQUIRING TEST:
SR 3.2.4.1	1 (>50% Rated Thermal Power)

TEST RESULTS (TO BE COMPLETED BY TEST PERFORMER)

PERFORMED BY: Stanley Jackson / [signature] DATE/TIME: TODAY / NOW
(Print) (Signature)

COMPONENT OR TRAIN TESTED (if applicable) N/A

- [x] ENTIRE STP PERFORMED
- [ ] PARTIAL STP PERFORMED
- [x] FOR SURVEILLANCE CREDIT
- [ ] NOT FOR SURVEILLANCE CREDIT

REASON FOR PARTIAL ____________________

TEST COMPLETED [x] Satisfactory [ ] Unsatisfactory

- [ ] The following deficiencies occurred

- [ ] Corrective action taken or initiated

SHIFT SUPERVISOR/ SHIFT SUPPORT SUPERVISOR REVIEW

- [ ] Procedure properly completed and satisfactory per step 9.1 of FNP-0-AP-5
- [ ] Comments ____________________

REVIEWED BY: ____________ / ____________ DATE: ____________
(Print) (Signature)

*Reviewer must be AP-31 Level II certified & cannot be the Performing Individual

ENGINEERING SUPPORT

GROUP SCREENING: SCREENED BY ____________ DATE ____________

(IF APPLICABLE)

- [ ] Comments ____________________

KEY

FARLEY
Unit 1

SAFETY RELATED

FNP-1-STP-7.0

Quadrant Power Tilt Ratio Calculation

VERSION 23.0

Special Considerations:

This is an upgraded procedure. Exercise increased awareness during initial use due to potential technical and/or sequential changes. After initial use, provide comments to the procedure upgrade team.

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous	**ALL**
Reference	**NONE**
Information	**NONE**

Approval: David L Reed (Approved By) 10/11/13 (Date)

Effective Date:

OPERATIONS
Responsible Department

VERSION SUMMARY
PVR 23.0 DESCRIPTION
Updated to fleet template and writer's guide

TABLE OF CONTENTS

1.0 PURPOSE

- To determine the quadrant power tilt ratio using power range nuclear instrumentation.
- Acceptance Criteria for this test is the quadrant power tilt ratio shall be ≤ 1.020.

2.0 PRECAUTIONS AND LIMITATIONS

1. Reactor power, rod position and reactor coolant temperature should be constant while taking data. ☐
2. A QPTR calculation should be done prior to rescaling of Power Range Nuclear Instruments, and after completing the rescaling of ALL Power Ranges Nuclear Instruments. A QPTR calculation performed between individual Power Range rescaling may provide erroneous results. ☐
3. IF one Power Range NI is inoperable AND thermal power is ≤ 75% RTP, the remaining power range channels may be used for calculating QPTR. **(SR 3.2.4.1)** ☐
4. Above 75% RTP, with one Power Range NI inoperable, QPTR must be determined by SR 3.2.4.2. ☐
5. The SM/SS shall be notified if any acceptance criteria are NOT satisfied. ☐

3.0 INITIAL CONDITIONS

1. The version of this procedure has been verified to be the current version. **(OR 1-98-498)** ______
2. This procedure has been verified to be the correct procedure for the task. **(OR 1-98-498)** ______
3. This procedure has been verified to be the correct unit for the task. **(OR 1-98-498)** ______

NOTE

This STP may be performed at less than 50% power for verification of power range instrument indications. In this case, the STP is NOT for surveillance credit. ☐

4. Unit 1 is above 50% of rated thermal power. ______
5. IF DVM is used to collect data, I&C has obtained a Fluke 45 or equivalent with shielded test leads with NO exposed metal connectors. ______

DVM Serial number ______________ Cal. due date ____________

3.0 INITIAL CONDITIONS (continued)

6. This procedure may contain previously evaluated Critical Steps that may not be applicable in certain plant conditions. The evaluation of this procedure for Critical Steps is performed during the Pre-Job briefing. The decision concerning how to address error precursors for critical steps should be governed by NMP-GM-005-GL03, Human Performance Tools. _______

NOTE

Asterisked (*) steps are those associated with Acceptance Criteria. ☐

4.0 <u>INSTRUCTIONS</u>

4.1 QPTR Determination Using The IPC.

NOTES

Section 4.2, QPTR Determination Using Manual Calculation: should be used to calculate QPTR when the IPC QPTR application is unavailable. ☐

1. **Open** the QPTR AND TILT FACTORS application on the IPC Applications Menu. _______
2. **Check** the following:
   - UPPER QPTR data indicates GOOD quality as indicated by affected points displayed in green. _______
   - LOWER QPTR data indicates GOOD quality as indicated by affected points displayed in green. _______
3. <u>IF</u> QPTR data is <u>NOT</u> GOOD quality, **go to** Section 4.2, QPTR Determination Using Manual Calculation: _______
4. <u>IF</u> QPTR data is GOOD quality, perform the following:
   a. **Click** PRINT EXCORE REPORT button. _______
   b. **Include** printed Excore Report with this procedure. _______
   c. **Go to** Section 4.3. _______

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE and THERMAL POWER ≤ 75% RTP, the remaining three power range channels may be used for calculating QPTR. ☐

4.2 QPTR Determination Using Manual Calculation:

1. **Calculate** QPTR using Attachment 1, Quadrant Power Tilt Ratio Calculation without Plant Computer _______

2. **Go to** Section 4.3. _______

4.3 Determination Of QPTR Acceptance Criteria:

NOTE

QPTR value displayed by the IPC utilizes 3 decimal places (to the thousandths place). If the QPTR value displayed is, for example 1.021, this would exceed the limit of 1.02 and require performance of the LCO 3.2.4 Condition A Required Actions.
(NL-10-0406, dated 2/26/2010) ☐

1. ***Check** Excore Maximum Quadrant Power Tilt Ratio ≤ 1.020 on either the EXCORE REPORT OR Attachment 1. _______

ACCEPTANCE CRITERIA

Maximum value of UPPER or LOWER Quadrant Power Tilt Ratio shall be ≤ 1.020.

NOTE
Asterisked (*) steps are those associated with Acceptance Criteria. ☐

5.0 ACCEPTANCE CRITERIA

The quadrant power tilt ratio shall be ≤ 1.020.

6.0 RECORDS

Documents created using this procedure will become QA Records when completed unless otherwise stated. The procedures and documents are considered complete when issued in DMS.

QA Record (X)	Non-QA Record (X)	Record Generated	Retention Time	R-Type
X		FNP-1-STP-7.0	LP	H06.045

7.0 REFERENCES

- FSAR - Chapter 4.4.2.4
- Technical Specification 3.2.4

Quadrant Power Tilt Ratio Calculation without Plant Computer

NOTE

QPTR may be determined using normalized currents from Curves 71A, 71B, 71C, 71D AND either of the following:

- Indicated detector current meter data. ☐
- Detector currents read by DVM using Attachment 2. ☐

1. **Obtain** normalized currents from Curve 71(A, B, C, D). ______

2. **Enter** normalized currents from Curve 71 on the Calculation Sheet. ______

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE AND THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. ☐

3. **Perform** the following:

   a. IF available, **enter** detector currents indicated on POWER RANGE B drawer meters on the Calculation Sheet for each of the following:

      - N1C55NI0041, N41B DETECTOR A, (Upper) ☐
      - N1C55NI0041, N41B DETECTOR B, (Lower) ☐
      - N1C55NI0042, N42B DETECTOR A, (Upper) ☐
      - N1C55NI0042, N42B DETECTOR B, (Lower) ☐
      - N1C55NI0043, N43B DETECTOR A, (Upper) ☐
      - N1C55NI0043, N43B DETECTOR B, (Lower) ☐
      - N1C55NI0044, N44B DETECTOR A, (Upper) ☐
      - N1C55NI0044, N44B DETECTOR B, (Lower) ☐

CAUTION

DVM readings may be taken in only one drawer at a time. ☐

   b. IF any NI current reading not available on the POWER RANGE B drawer, **enter** detector currents obtained by I&C using Attachment 2 for the affected detectors. ______

4. **Enter** total number of operable detectors in space provided on the Calculation Sheet. ______

Quadrant Power Tilt Ratio Calculation without Plant Computer

5. **Calculate** the following:

   - Upper Quadrant Power Tilt Ratio. ______
   - Lower Quadrant Power Tilt Ratio. ______

6. ***Record** the greater of the upper or lower Quadrant Power Tilt Ratio value in the space provided on the Calculation Sheet. ______

ACCEPTANCE CRITERIA
Maximum value of upper or lower Quadrant Power Tilt Ratio shall be ≤ 1.020.

7. **Record** the Power Level (Avg) in the space provided. ______

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

UPPER QUADRANT POWER TILT

POWER RANGE B Drawer	UPPER DET Indicated Current	÷	*UPPER DET 100% Current	=	UPPER DET Calibrated Output						
N41	Detector A	÷	N41T	=							
N42	Detector A	÷	N42T	=							
N43	Detector A	÷	N43T	=							
N44	Detector A	÷	N44T	=		Total Number Operable Upper Detectors	1 / Average Upper Detector Calibrated Output	X	Maximum Upper Detector Calibrated Output	=	Upper Quadrant Power Tilt Ratio
Total Upper Detector Calibrated Output				=		÷	= 1 /	X		=	

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

LOWER QUADRANT POWER TILT

POWER RANGE B Drawer	LOWER DET Indicated Current	÷	*LOWER DET 100% Current	=	LOWER DET Calibrated Output
N41	Detector B	÷	N41B	=	
N42	Detector B	÷	N42B	=	
N43	Detector B	÷	N43B	=	
N44	Detector B	÷	N44B	=	

			Total Number Operable Lower Detectors		$\frac{1}{\text{Average Lower Detector Calibrated Output}}$	X	Maximum Lower Detector Calibrated Output	=	Lower Quadrant Power Tilt Ratio
Total Lower Detector Calibrated Output	=		÷	=	$\frac{1}{}$	X		=	

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Upper QPTR

Lower QPTR

Maximum of Upper or Lower QPTR

*

ACCEPTANCE CRITERIA
Maximum of Upper or Lower Quadrant Power Tilt Ratio does not exceed 1.020.

% Reactor Power ____________

ATTACHMENT 2
Page 1 of 2

Using A DVM To Obtain Detector Current Values

NOTE
Detector current values may be obtained for as many drawers as required. Unused spaces in the Table should be marked N/A. ☐

CAUTIONS
- DVM readings may be taken in only one drawer at a time. ☐
- A Fluke 8600 shall NOT be used to obtain currents ☐

1. Using a Fluke 45 or equivalent AND shielded test leads **connect** to obtain detector voltage readings as follows:

NOTE
Voltage values should be in the 2 to 3 volt range. ☐

a. For Upper Detector **connect** to TP301 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

b. For Lower Detector **connect** to TP302 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

Using A DVM To Obtain Detector Current Values

NOTE

The following formula is used to calculate detector currents:

$$\frac{\text{Measured Detector Voltage}}{2.083} \times \text{Curve 71 "0\% AFD, 100\% Current" Value} = \text{Calculated Detector Current}$$ ☐

2. Using the 0% AFD, 100% current value from Curve 71, **perform** the following:

   a. **Calculate** the detector current value. ______

   b. **Record** in appropriate space of table below. ______

	N41		N42		N43		N44	
	Upper Detector A N41T	Lower Detector B N41B	Upper Detector A N42T	Lower Detector B N42B	Upper Detector A N43T	Lower Detector B N43B	Upper Detector A N44T	Lower Detector B N44B
	DVM Voltage		DVM Voltage		DVM Voltage		DVM Voltage	
Step 1								
	Calculated Current		Calculated Current		Calculated Current		Calculated Current	
Step 2								

ATTACHMENT 3
Page 1 of 1

Surveillance Test Review Sheet

TECHNICAL SPECIFICATION REFERENCE	MODE(S) REQUIRING TEST:
SR 3.2.4.1	1 (>50% Rated Thermal Power)

TEST RESULTS (TO BE COMPLETED BY TEST PERFORMER)

PERFORMED BY: ______________ / ______________ DATE/TIME: ______ / ______
(Print) (Signature)

COMPONENT OR TRAIN TESTED (if applicable) ______________

☐ ENTIRE STP PERFORMED ☐ FOR SURVEILLANCE CREDIT

☐ PARTIAL STP PERFORMED ☐ NOT FOR SURVEILLANCE CREDIT

REASON FOR PARTIAL ______________

TEST COMPLETED ☐ Satisfactory ☐ Unsatisfactory

☐ The following deficiencies occurred

☐ Corrective action taken or initiated

SHIFT SUPERVISOR/ SHIFT SUPPORT SUPERVISOR REVIEW

☐ Procedure properly completed and satisfactory per step 9.1 of FNP-0-AP-5

☐ Comments ______________

REVIEWED BY: ______________ / ______________ DATE: ______________
(Print) (Signature)

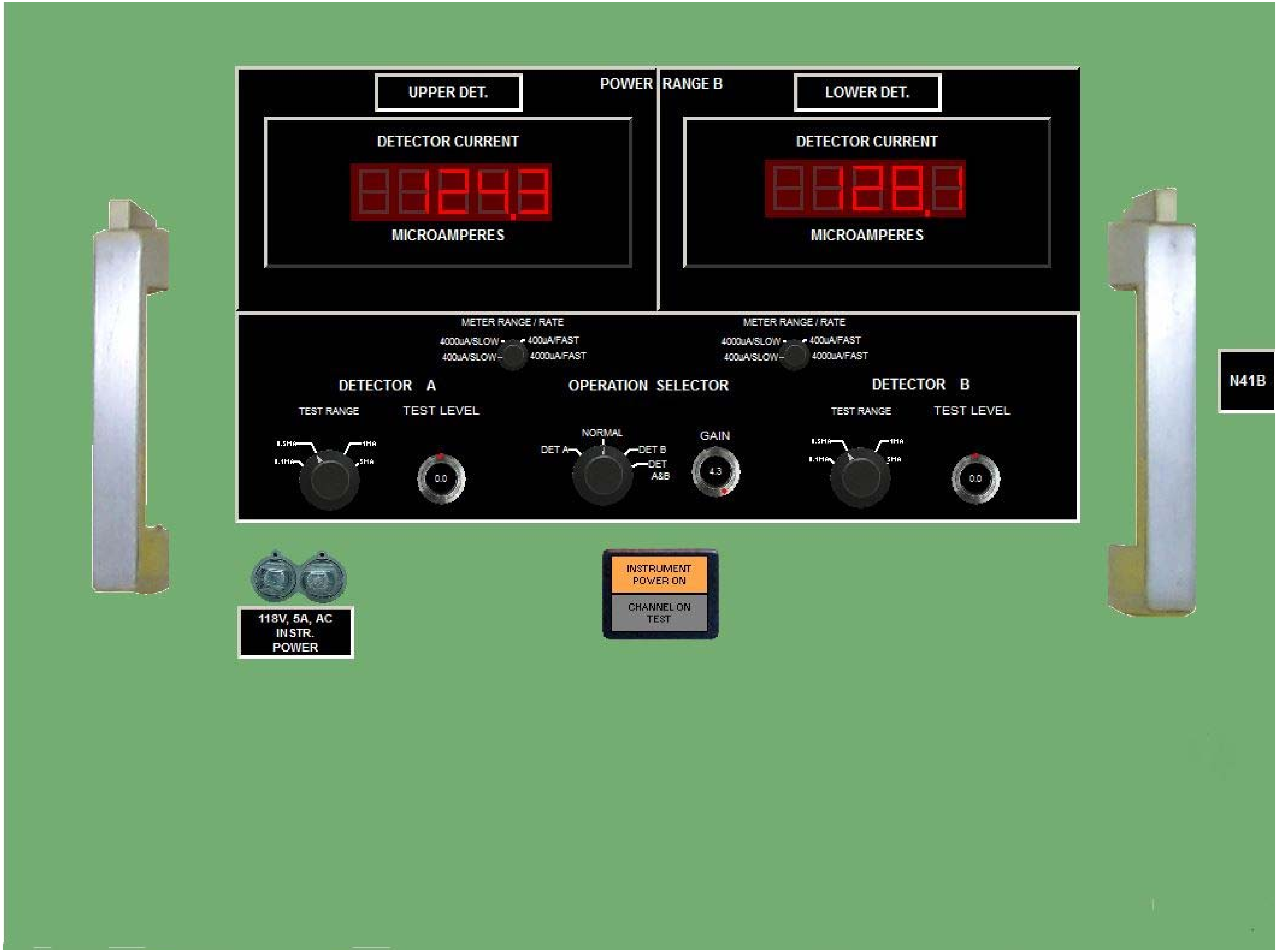
*Reviewer must be AP-31 Level II certified & cannot be the Performing Individual

ENGINEERING SUPPORT

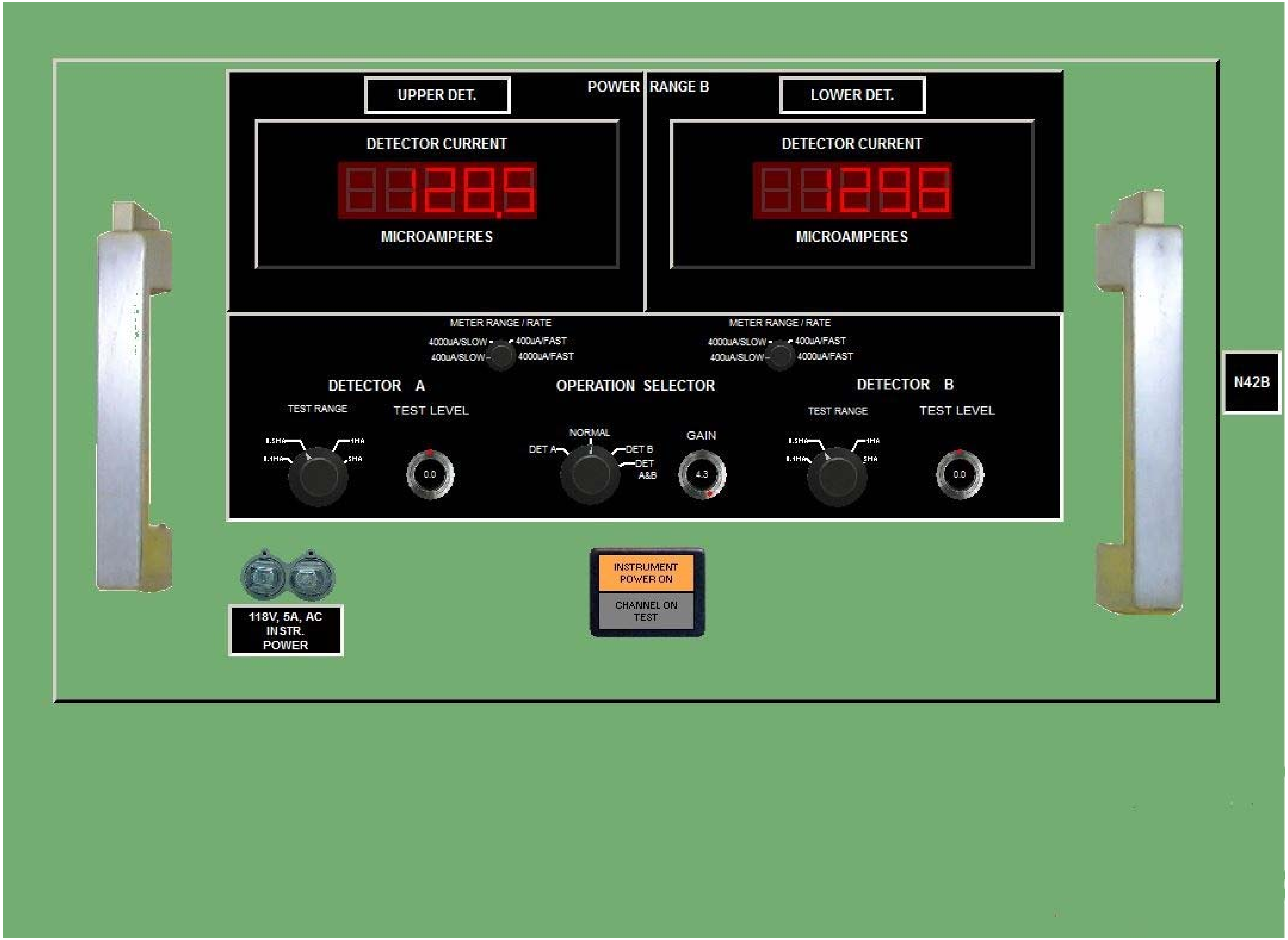
GROUP SCREENING: SCREENED BY ______________ DATE ______

(IF APPLICABLE)

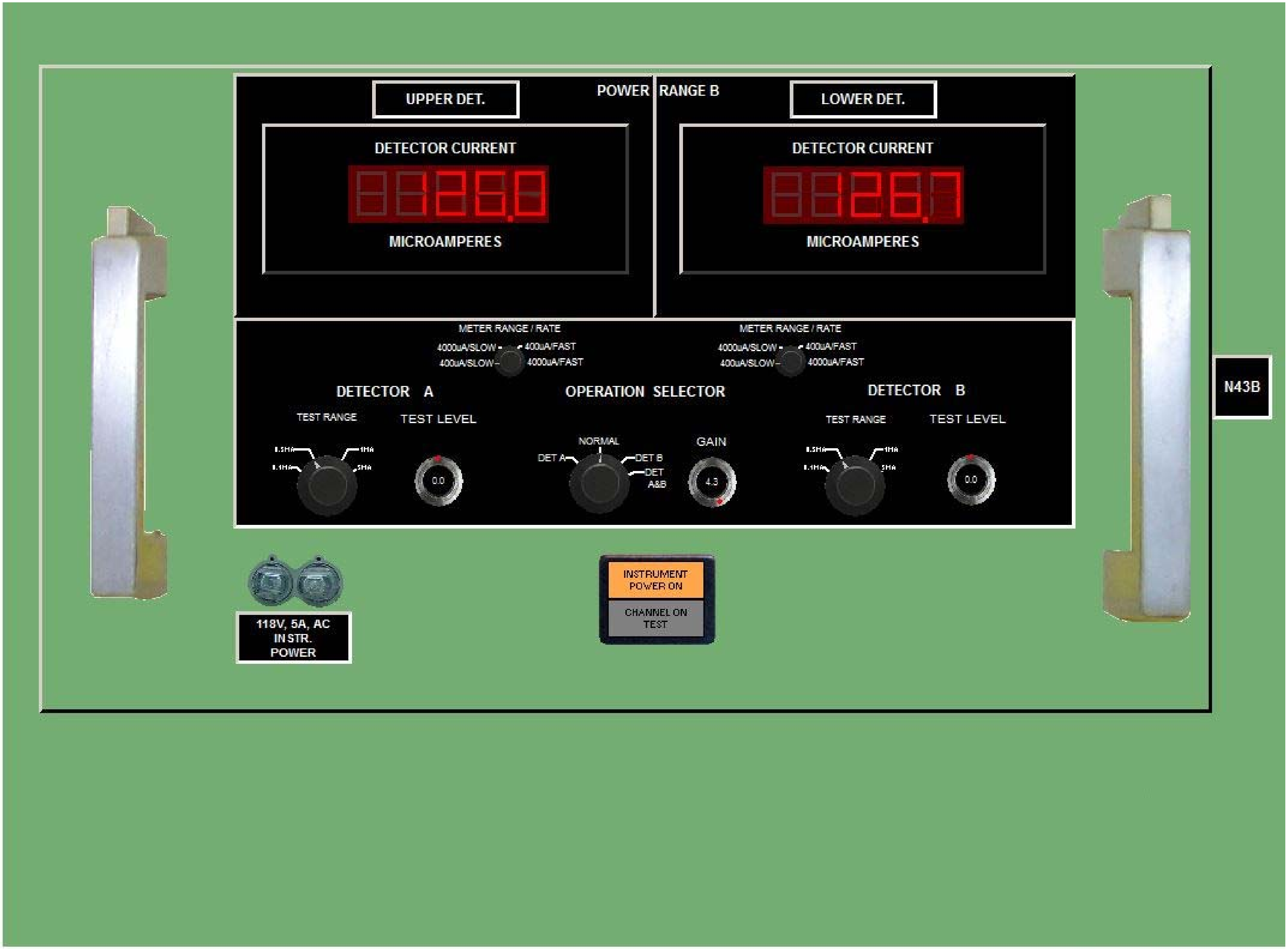
☐ Comments ______________

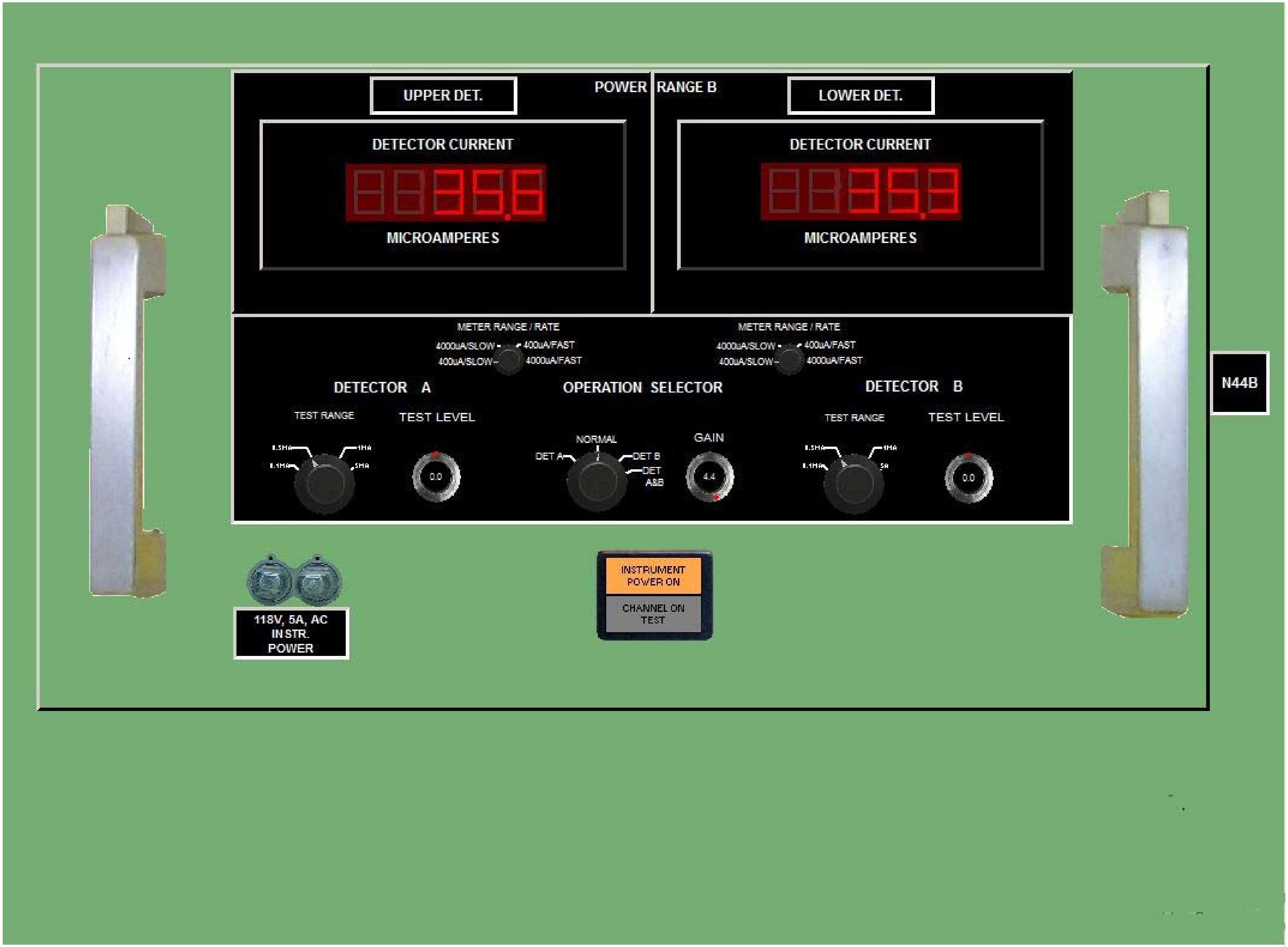


UPPER DET.
POWER RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
OPERATION SELECTOR
NORMAL
DET A
DET B
DET A&B
GAIN
DETECTOR B
TEST RANGE
TEST LEVEL
N41B
118V, 5A, AC INSTR. POWER
INSTRUMENT POWER ON
CHANNEL ON TEST

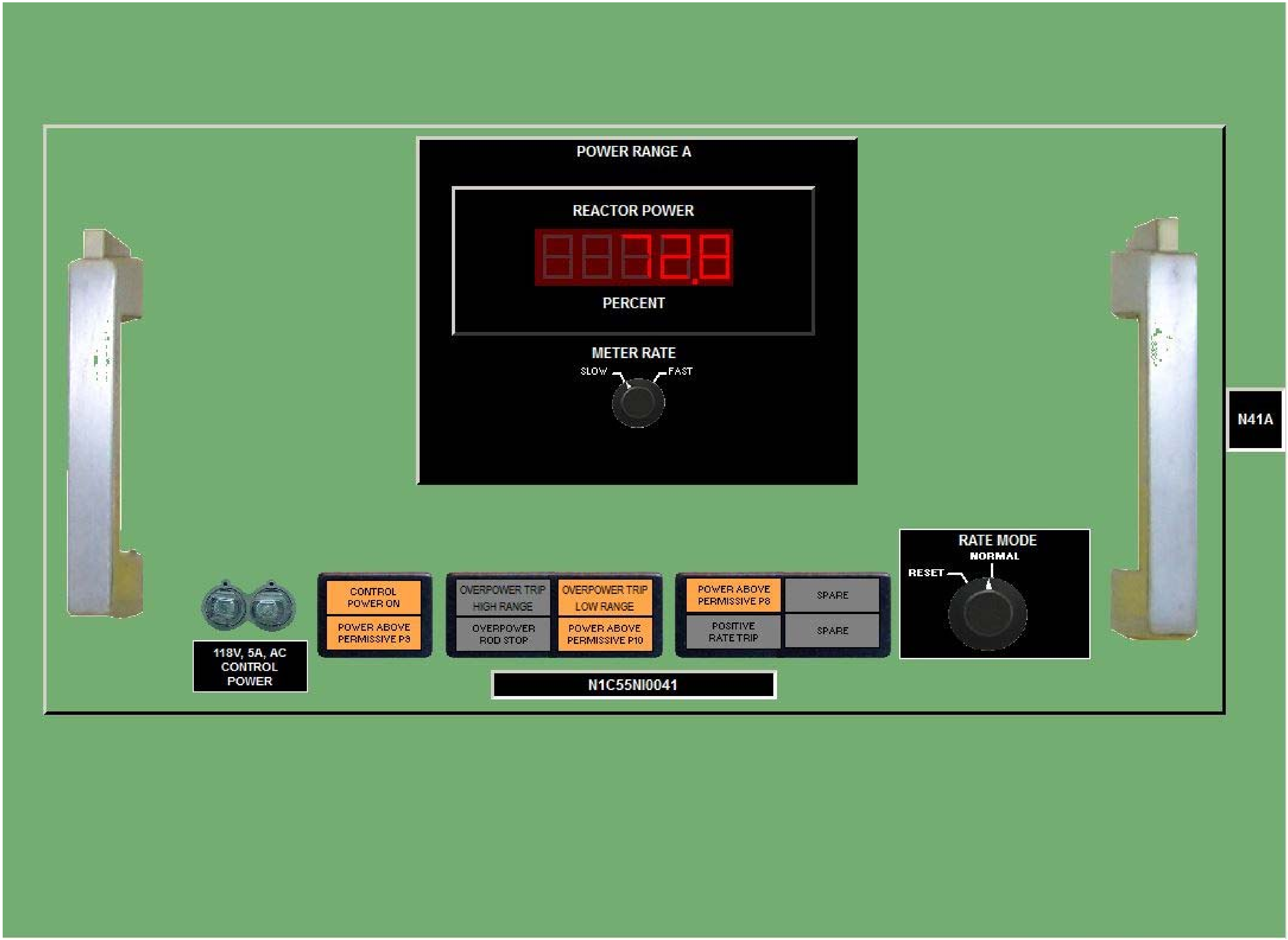


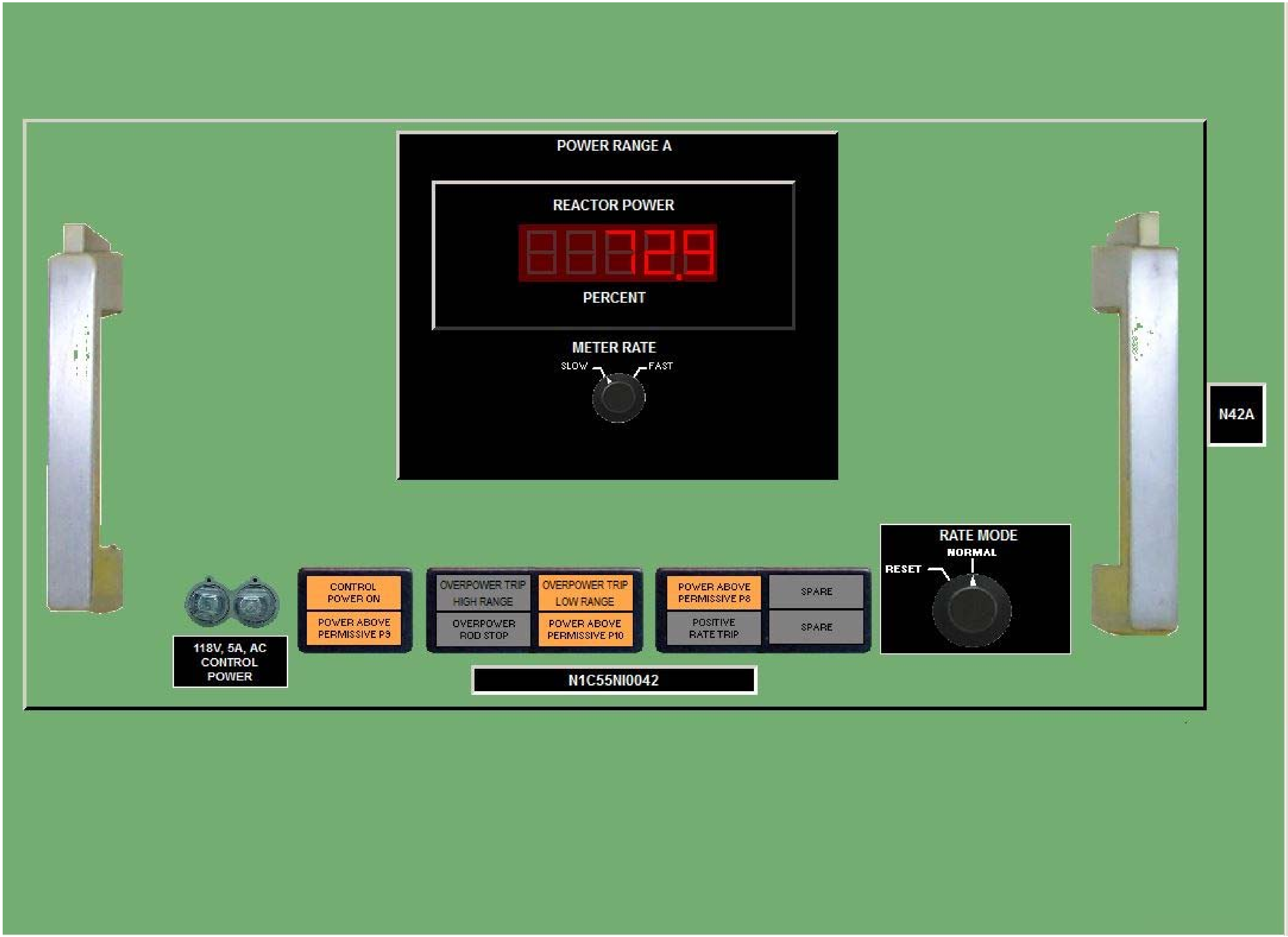
UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
OPERATION SELECTOR
NORMAL
DET A
DET B
DET A&B
GAIN
4.3
DETECTOR B
TEST RANGE
TEST LEVEL
0.0
0.0
N42B
118V, 5A, AC INSTR. POWER
INSTRUMENT POWER ON
CHANNEL ON TEST

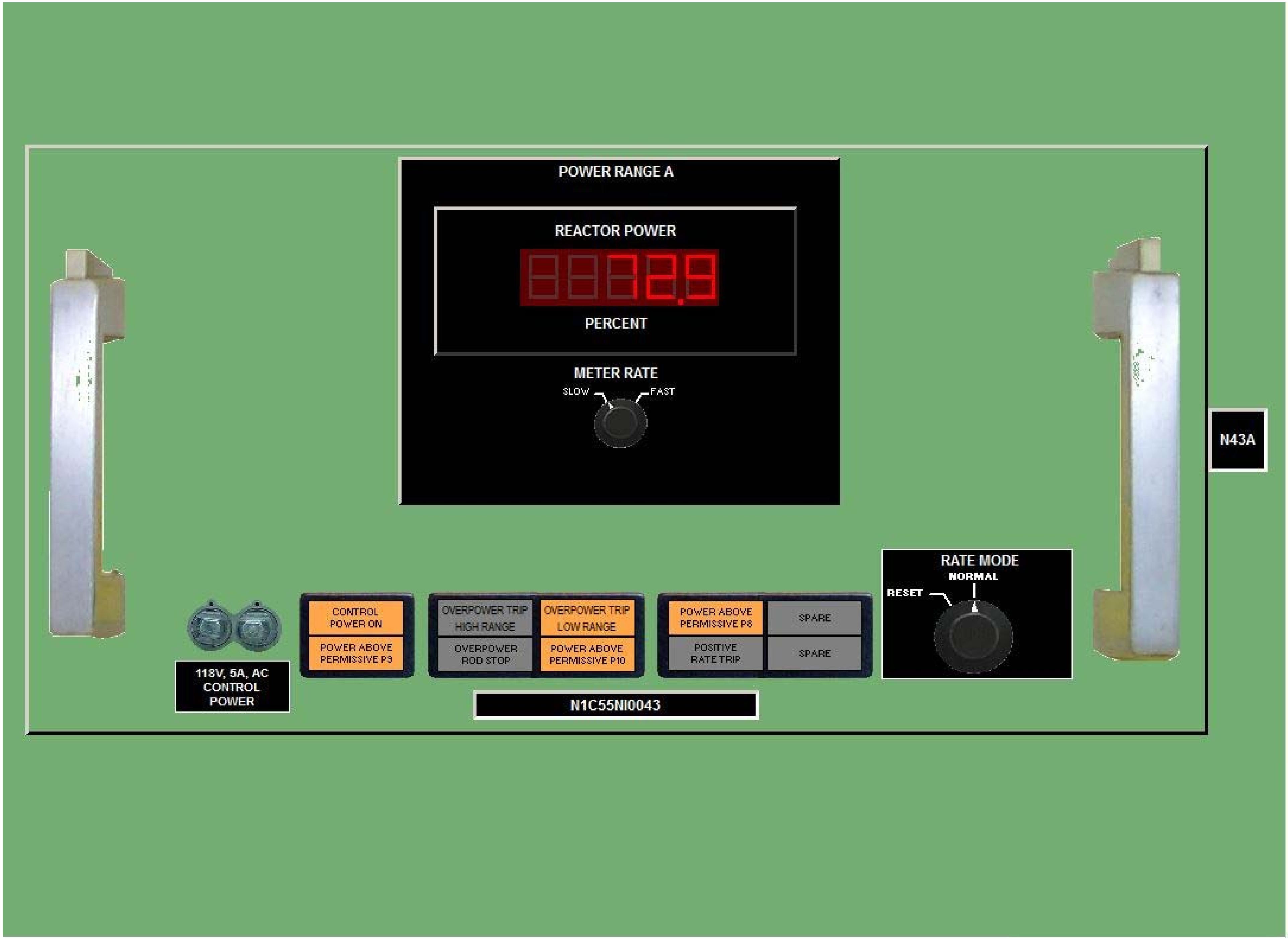

UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
OPERATION SELECTOR
DETECTOR B
TEST RANGE
TEST LEVEL
NORMAL
DET A
DET B
DET A&B
GAIN
TEST RANGE
TEST LEVEL
N43B
118V, 5A, AC
IN STR.
POWER
INSTRUMENT
POWER ON
CHANNEL ON
TEST

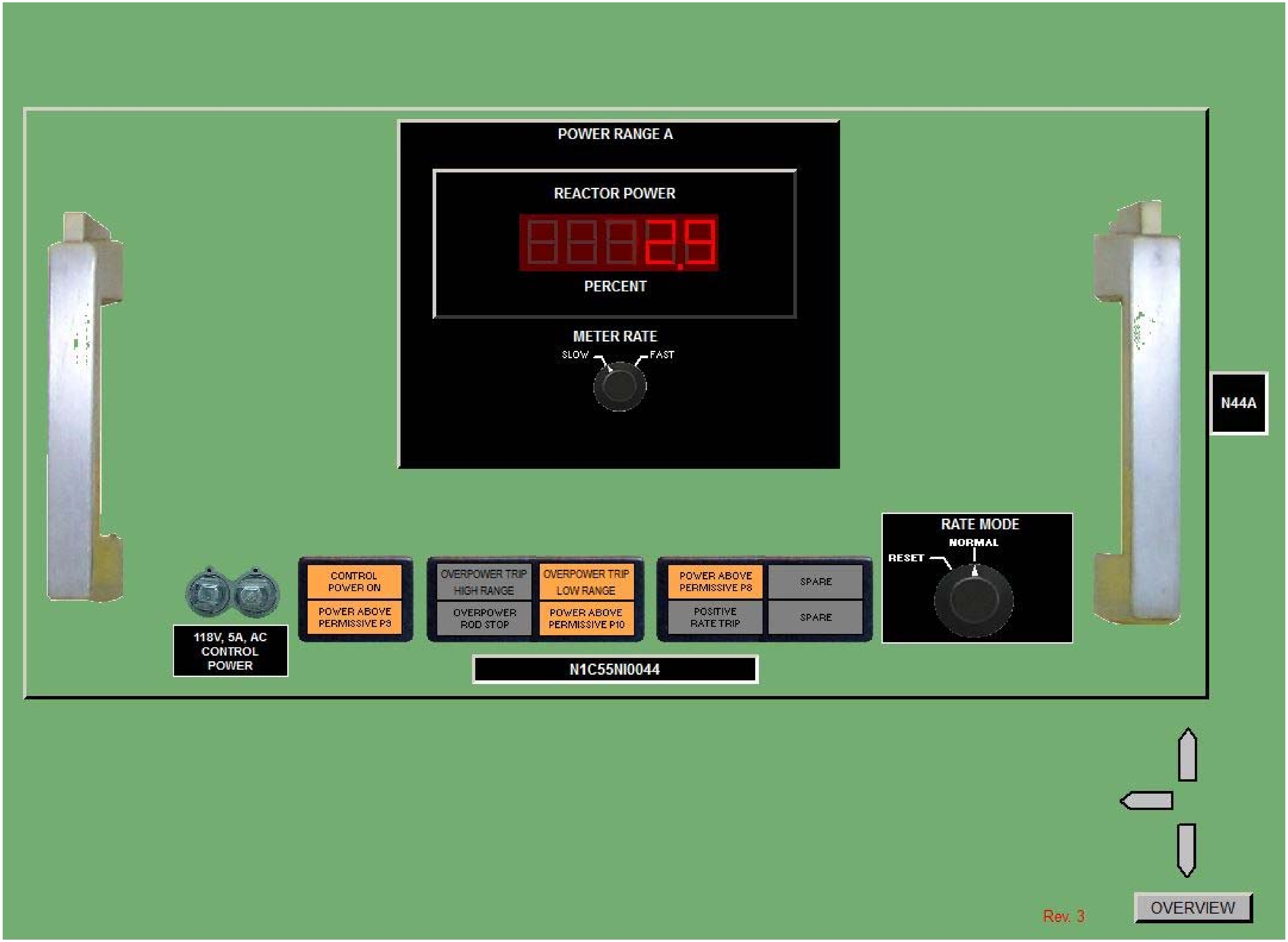


UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
35.6
MICROAMPERES
DETECTOR CURRENT
35.3
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
OPERATION SELECTOR
DETECTOR B
TEST RANGE
TEST LEVEL
NORMAL
DET A
DET B
DET A&B
GAIN
TEST RANGE
TEST LEVEL
N44B
118V, 5A, AC INSTR. POWER
INSTRUMENT POWER ON
CHANNEL ON TEST


POWER RANGE A
REACTOR POWER
72.8
PERCENT
METER RATE
SLOW
FAST
N41A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER TRIP LOW RANGE
OVERPOWER ROD STOP
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
SPARE
POSITIVE RATE TRIP
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0041


POWER RANGE A
REACTOR POWER
72.9
PERCENT
METER RATE
SLOW
FAST
N42A
RATE MODE
NORMAL
RESET
118V, 5A, AC CONTROL POWER
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
N1C55NI0042


POWER RANGE A
REACTOR POWER
PERCENT
METER RATE
SLOW
FAST
N43A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0043



POWER RANGE A
REACTOR POWER
2.9
PERCENT
METER RATE
SLOW
FAST
N44A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0044
Rev. 3
OVERVIEW

UNIT 1 VOLUME 1 CURVE 71A

PRESENT NIS CHANNEL N41 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14

LAW 8-22-14 Reactor Engineering Supervisor Date

PI Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N41T	215.71	187.44	159.18	86.0324	20.6460	0.6194
N41B	148.89	185.63	222.36			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N41T	I Det =	0.9423	* AO +	187.4449
N41B	I Det =	-1.2245	* AO +	185.6270

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-41 Calibration are FNP-1-IMP-228.8 & FNP-1-STP-228.5

Curve Placed in Effect:

_______________ / _______________

Shift Supervisor Date / Time

(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71B

PRESENT NIS CHANNEL N42 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14
Reactor Engineering Supervisor Date
Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N42T	220.92	191.11	161.30	82.8203	19.8780	0.5963
N42B	148.30	186.84	225.38			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N42T	I Det =	0.9937	* AO +	191.1122
N42B	I Det =	-1.2846	* AO +	186.8399

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-42 calibration are FNP-1-IMP-228.9 & FNP-1-STP-228.6

Curve Placed in Effect:

________________ / ____________
Shift Supervisor Date / Time
(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71C

PRESENT NIS CHANNEL N43 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14
Reactor Engineering Supervisor Date
Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N43T	213.18	185.03	156.87	85.6130	20.5530	0.6164
N43B	153.54	191.51	229.48			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N43T	I Det =	0.9385	* AO +	185.0294
N43B	I Det =	-1.2657	* AO +	191.5063

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-43 calibration are FNP-1-IMP-228.10 and FNP-1-STP-228.7

Curve Placed in Effect:

______________________ / ______________________
Shift Supervisor Date / Time
(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71D

PRESENT NIS CHANNEL N44 CURRENT SETTINGS

Rev. 38 08/22/2014 SRM

Approved: [signature] 8-22-14

Reactor Engineering Supervisor Date

Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N44T	233.39	201.99	170.60	82.0409	19.6930	
N44B	154.85	196.07	237.29			N/A

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N44T	I Det =	**1.0466**	* AO +	**201.9949**
N44B	I Det =	**-1.3740**	* AO +	**196.0726**

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-44 Calibration are FNP-1-IMP-228.11 & FNP-1-STP-228.8

Curve Placed in Effect:

______________________ / ______________________

Shift Supervisor Date / Time

(To be completed following scaling in rack)

A.2 SRO

TITLE: Perform A Quadrant Power Tilt Ratio Calculation		
EVALUATION LOCATION: ☐ SIMULATOR	☐CONTROL ROOM	☒CLASSROOM
PROJECTED TIME: 20 MIN	SIMULATOR IC NUMBER: N/A	
☐ ALTERNATE PATH	☐ TIME CRITICAL	☐ PRA

JPM DIRECTIONS:

1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Do not allow the students to use the Core Physics curves on the classroom computer, use the Core Physics curves provided in the HANDOUT.
3. Provide the first Handout initially for the applicant's performance of STP-7.0.
4. Provide Handout 2 only if the applicant determines that the STP is UNSAT and Tech Spec evaluation is required.

TASK STANDARD: Upon successful completion of this JPM, the examinee will:

1. Correctly determine the QPTR.
2. Correctly determine whether or not the QPTR meets acceptance criteria.
3. Correctly determine any actions required based on results of the calculations.

Examinee:			
Overall JPM Performance:	**Satisfactory** ☐		**Unsatisfactory** ☐
Evaluator Comments **(attach additional sheets if necessary)**			

EXAMINER: ____________________________

Developer	S. Jackson	Date: 4/3/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to PERFORM A QUADRANT POWER TILT RATIO CALCULATION. The conditions under which this task is to be performed are:

a. N-41, N-42, & N-43 PR NI detectors are OPERABLE.
b. N-44 PR NI detector is INOPERABLE.
c. You are directed by Shift Supervisor to perform STP-7.0, using curves 71A-D, the pictures provided, and determine if the acceptance criteria is met.
d. The IPC and QPTR computer spreadsheet are not available.
e. A DVM will NOT be used to collect data.
f. A pre-job brief is not required.

EVALUATION CHECKLIST

ELEMENTS:		STANDARDS:	RESULTS: (CIRCLE)
_____ **START TIME**			
NOTE: Critical to use the correct 0% AFD values from curves.			
*1.	Obtain normalized currents from curves 71A, 71B, & 71C.	Obtains normalized current values (Curve 71A-C) and records them on Attachment 1 of STP-7.0.	S / U
*2.	Record data for power range detector A and detector B from Data sheet 2.	Values from PRNI pictures for detector A and detector B of NI-41, 42, & 43 displays recorded on Attachment 1 of STP-7.0.	S / U
*3.	Calculate upper and lower quadrant power tilt ratios.	Upper ratio calculated at 1.03 to 1.04 Lower ratio calculated at 1.01 to 1.02	S / U
*4.	Enter the greater of the upper or lower quadrant power tilt ratio.	Greater of the above two values Lower: 1.03 to 1.04 entered.	S / U
5.	Records power level.	Current avg power level recorded: 72-73%.	S / U
*6.	Determines acceptance criteria **NOT MET**.	Determination made that acceptance criteria is **NOT MET**.	S / U
7.	Reports to Shift Supervisor that acceptance criteria is NOT met.	Reports to Shift Supervisor that acceptance criteria is **NOT MET.** (**CUE: Shift Supervisor acknowledges**).	S / U

EVALUATION CHECKLIST

ELEMENTS:		STANDARDS:	RESULTS: (CIRCLE)
8.	Fills out Surveillance Test Review sheet per attached key.	Fills out Surveillance Test Review sheet per attached key. (If applicant states they would write a CR then **CUE: CR#123456 has been written**)	S / U

If the applicant correctly states that acceptance criteria is <u>NOT MET</u>, then provide the second HANDOUT.
TECH SPEC EVALUATION: (The Tech Spec will be in the examiner's key package)

*9.	Evaluates Tech Spec 3.2.4 – Quadrant Power Tilt Ratio (QPTR). The QTPR shall be $\leq$ 1.02.	Determines LCO 3.2.4 Condition A applies but no power reduction is required.	S / U

_____ **STOP TIME**

Terminate when assessment of acceptance criteria is performed and the Tech Spec evaluation is completed.

<u>CRITICAL ELEMENTS</u>: Critical Elements are denoted with an asterisk (*) preceding the element number.

<u>GENERAL REFERENCES</u>

1. FNP-1-STP-7.0, Version 23.0
2. Core Physics curves 71A-D Rev. 36.0
3. Tech Specs, Version 198
4. K/A: G2.1.12 – 3.7 / 4.1

<u>GENERAL TOOLS AND EQUIPMENT</u>

1. Calculator
2. STP-7.0
3. Core Physics curves 71A-D
4. Pictures of PRNI's.
5. Tech Specs

Critical ELEMENT justification:

STEP	Evaluation
1-4	**Critical:** Task completion: required to properly determine QTPR.
5	**Not Critical:** Does not determine the calculation nor the acceptance criteria.
6	**Critical:** Task completion: Must decide whether or not acceptance criteria is met.
7-8	**Not Critical:** Does not determine the calculation nor the acceptance criteria.
9	**Critical:** Task completion: required to comply with Tech Specs and operate within the facility's license.

COMMENTS:

HANDOUT

CONDITIONS

When I tell you to begin, you are to PERFORM A QUADRANT POWER TILT RATIO CALCULATION. The conditions under which this task is to be performed are:

a. N-41, N-42, & N-43 PR NI detectors are OPERABLE.
b. N-44 PR NI detector is INOPERABLE.
c. You are directed by Shift Supervisor to perform STP-7.0, using curves 71A-D, the pictures provided, and determine if the acceptance criteria is met.
d. The IPC and QPTR computer spreadsheet are not available.
e. A DVM will NOT be used to collect data.
f. A pre-job brief is not required.

A.2 SRO

HANDOUT 2

PROVIDE TO THE APPLICANT AFTER THEY COMPLETE THE CALCULATIONS

1. Determine what action(s) are to be taken, if any, based on the results you have determined in STP-7.0.

FARLEY
Unit 1

SAFETY RELATED

FNP-1-STP-7.0

Quadrant Power Tilt Ratio Calculation

VERSION 23.0

Special Considerations:

This is an upgraded procedure. Exercise increased awareness during initial use due to potential technical and/or sequential changes. After initial use, provide comments to the procedure upgrade team.

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous	**ALL**
Reference	**NONE**
Information	**NONE**

Approval: David L Reed — Approved By; 10/11/13 — Date

Effective Date:

OPERATIONS
Responsible Department

VERSION SUMMARY
PVR 23.0 DESCRIPTION
Updated to fleet template and writer's guide

TABLE OF CONTENTS

1.0 PURPOSE

- To determine the quadrant power tilt ratio using power range nuclear instrumentation.
- Acceptance Criteria for this test is the quadrant power tilt ratio shall be ≤ 1.020.

2.0 PRECAUTIONS AND LIMITATIONS

1. Reactor power, rod position and reactor coolant temperature should be constant while taking data. ☒
2. A QPTR calculation should be done prior to rescaling of Power Range Nuclear Instruments, and after completing the rescaling of ALL Power Ranges Nuclear Instruments. A QPTR calculation performed between individual Power Range rescaling may provide erroneous results. ☒
3. IF one Power Range NI is inoperable AND thermal power is ≤ 75% RTP, the remaining power range channels may be used for calculating QPTR. **(SR 3.2.4.1)** ☒
4. Above 75% RTP, with one Power Range NI inoperable, QPTR must be determined by SR 3.2.4.2. ☒
5. The SM/SS shall be notified if any acceptance criteria are NOT satisfied. ☒

3.0 INITIAL CONDITIONS

1. The version of this procedure has been verified to be the current version. **(OR 1-98-498)** SJJ
2. This procedure has been verified to be the correct procedure for the task. **(OR 1-98-498)** SJJ
3. This procedure has been verified to be the correct unit for the task. **(OR 1-98-498)** SJJ

NOTE
This STP may be performed at less than 50% power for verification of power range instrument indications. In this case, the STP is NOT for surveillance credit. ☒

4. Unit 1 is above 50% of rated thermal power. SJJ
5. IF DVM is used to collect data, I&C has obtained a Fluke 45 or equivalent with shielded test leads with NO exposed metal connectors. N/A

DVM Serial number ______________ Cal. due date _____________

KEY

3.0 INITIAL CONDITIONS (continued)

6. This procedure may contain previously evaluated Critical Steps that may not be applicable in certain plant conditions. The evaluation of this procedure for Critical Steps is performed during the Pre-Job briefing. The decision concerning how to address error precursors for critical steps should be governed by NMP-GM-005-GL03, Human Performance Tools. SJJ ______

NOTE

Asterisked (*) steps are those associated with Acceptance Criteria. ☐

4.0 INSTRUCTIONS

N/A

4.1 QPTR Determination Using The IPC.

NOTES

Section 4.2, QPTR Determination Using Manual Calculation: should be used to calculate QPTR when the IPC QPTR application is unavailable. ☐

1. **Open** the QPTR AND TILT FACTORS application on the IPC Applications Menu. ______

2. **Check** the following:

   - UPPER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

   - LOWER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

SJJ

3. IF QPTR data is NOT GOOD quality, **go to** Section 4.2, QPTR Determination Using Manual Calculation: ______

4. IF QPTR data is GOOD quality, perform the following:

   a. **Click** PRINT EXCORE REPORT button. ______

   b. **Include** printed Excore Report with this procedure. ______

   c. **Go to** Section 4.3. ______

N/A

KEY

KEY

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE and THERMAL POWER ≤ 75% RTP, the remaining three power range channels may be used for calculating QPTR. ☑

4.2 QPTR Determination Using Manual Calculation:

1. **Calculate** QPTR using Attachment 1, Quadrant Power Tilt Ratio Calculation without Plant Computer SJJ

2. **Go to** Section 4.3. SJJ

4.3 Determination Of QPTR Acceptance Criteria:

NOTE

QPTR value displayed by the IPC utilizes 3 decimal places (to the thousandths place). If the QPTR value displayed is, for example 1.021, this would exceed the limit of 1.02 and require performance of the LCO 3.2.4 Condition A Required Actions.
(NL-10-0406, dated 2/26/2010) ☑

1. ***Check** Excore Maximum Quadrant Power Tilt Ratio ≤ 1.020 on either the EXCORE REPORT <u>OR</u> Attachment 1. SJJ

ACCEPTANCE CRITERIA

Maximum value of UPPER <u>or</u> LOWER Quadrant Power Tilt Ratio shall be ≤ 1.020.

KEY

NOTE
Asterisked (*) steps are those associated with Acceptance Criteria.

5.0 ACCEPTANCE CRITERIA

The quadrant power tilt ratio shall be ≤ 1.020.

6.0 RECORDS

Documents created using this procedure will become QA Records when completed unless otherwise stated. The procedures and documents are considered complete when issued in DMS.

QA Record (X)	Non-QA Record (X)	Record Generated	Retention Time	R-Type
X		FNP-1-STP-7.0	LP	H06.045

7.0 REFERENCES

- FSAR - Chapter 4.4.2.4
- Technical Specification 3.2.4

Quadrant Power Tilt Ratio Calculation without Plant Computer

NOTE

QPTR may be determined using normalized currents from Curves 71A, 71B, 71C, 71D AND either of the following:

- Indicated detector current meter data. ☑
- Detector currents read by DVM using Attachment 2. ☑

1. **Obtain** normalized currents from Curve 71(A, B, C, D). SJJ
2. **Enter** normalized currents from Curve 71 on the Calculation Sheet. SJJ

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE AND THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. ☑

3. **Perform** the following:

   a. IF available, **enter** detector currents indicated on POWER RANGE B drawer meters on the Calculation Sheet for each of the following:

      - N1C55NI0041, N41B DETECTOR A, (Upper) ☑
      - N1C55NI0041, N41B DETECTOR B, (Lower) ☑
      - N1C55NI0042, N42B DETECTOR A, (Upper) ☑
      - N1C55NI0042, N42B DETECTOR B, (Lower) ☑
      - N1C55NI0043, N43B DETECTOR A, (Upper) ☑
      - N1C55NI0043, N43B DETECTOR B, (Lower) ☑
      - N1C55NI0044, N44B DETECTOR A, (Upper) ☑
      - N1C55NI0044, N44B DETECTOR B, (Lower) ☑

CAUTION

DVM readings may be taken in only one drawer at a time. ☑

   b. IF any NI current reading not available on the POWER RANGE B drawer, **enter** detector currents obtained by I&C using Attachment 2 for the affected detectors. N/A

4. **Enter** total number of operable detectors in space provided on the Calculation Sheet. SJJ

Quadrant Power Tilt Ratio Calculation without Plant Computer

5. **Calculate** the following:
   - Upper Quadrant Power Tilt Ratio. SJJ
   - Lower Quadrant Power Tilt Ratio. SJJ

6. ***Record** the greater of the upper or lower Quadrant Power Tilt Ratio value in the space provided on the Calculation Sheet. SJJ

ACCEPTANCE CRITERIA

Maximum value of upper or lower Quadrant Power Tilt Ratio shall be ≤ 1.020.

7. **Record** the Power Level (Avg) in the space provided. SJJ

KEY

ATTACHMENT 1
Page 3 of 5

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

UPPER QUADRANT POWER TILT

POWER RANGE B Drawer	UPPER DET Indicated Current	÷	*UPPER DET 100% Current	=	UPPER DET Calibrated Output
N41	Detector A 124.3	÷	N41T 187.44	=	0.663
N42	Detector A 128.5	÷	N42T 191.11	=	0.672
N43	Detector A 130.6	÷	N43T 185.03	=	0.706
N44	Detector A N/A	÷	N44T N/A	=	N/A

	=		÷	Total Number Operable Upper Detectors	=	1 / Average Upper Detector Calibrated Output	X	Maximum Upper Detector Calibrated Output	=	Upper Quadrant Power Tilt Ratio
Total Upper Detector Calibrated Output	=	2.041	÷	3	=	1 / 0.680	X	0.706		1.03 to 1.04

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

KEY

KEY

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

LOWER QUADRANT POWER TILT

POWER RANGE B Drawer	LOWER DET Indicated Current	÷	*LOWER DET 100% Current	=	LOWER DET Calibrated Output
N41	Detector B: 128.1	÷	N41B: 185.63	=	0.690
N42	Detector B: 129.6	÷	N42B: 186.84	=	0.694
N43	Detector B: 135.3	÷	N43B: 191.51	=	0.706
N44	Detector B: N/A	÷	N44B: N/A	=	N/A

				Total Number Operable Lower Detectors		1 / Average Lower Detector Calibrated Output	X	Maximum Lower Detector Calibrated Output	=	Lower Quadrant Power Tilt Ratio
Total Lower Detector Calibrated Output	=	2.09	÷	3	=	1 / 0.697	X	0.706	=	1.01 to 1.02

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

KEY

KEY

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Upper QPTR	Lower QPTR
1.03 to 1.04	1.01 to 1.02

Maximum of Upper or Lower QPTR

1.03 to 1.04

ACCEPTANCE CRITERIA
Maximum of Upper or Lower Quadrant Power Tilt Ratio does not exceed 1.020.

% Reactor Power 72 - 73%

KEY

ATTACHMENT 2
Page 1 of 2

Using A DVM To Obtain Detector Current Values

NOTE

Detector current values may be obtained for as many drawers as required. Unused spaces in the Table should be marked N/A. ☐

CAUTIONS

- DVM readings may be taken in only one drawer at a time. ☐
- A Fluke 8600 shall NOT be used to obtain currents ☐

1. Using a Fluke 45 or equivalent AND shielded test leads **connect** to obtain detector voltage readings as follows:

NOTE

Voltage values should be in the 2 to 3 volt range. ☐

a. For Upper Detector **connect** to TP301 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

b. For Lower Detector **connect** to TP302 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

KEY

ATTACHMENT 2
Page 2 of 2

Using A DVM To Obtain Detector Current Values

NOTE

The following formula is used to calculate detector currents:

$$\frac{\text{Measured Detector Voltage}}{2.083} \times \text{Curve 71 "0\% AFD, 100\% Current" Value} = \text{Calculated Detector Current}$$ ☐

2. Using the 0% AFD, 100% current value from Curve 71, **perform** the following:

   a. **Calculate** the detector current value. _______

   b. **Record** in appropriate space of table below. _______

	N41		N42		N43		N44	
	Upper Detector A N41T	Lower Detector B N41B	Upper Detector A N42T	Lower Detector B N42B	Upper Detector A N43T	Lower Detector B N43B	Upper Detector A N44T	Lower Detector B N44B
	DVM Voltage		DVM Voltage		DVM Voltage		DVM Voltage	
Step 1								
	Calculated Current		Calculated Current		Calculated Current		Calculated Current	
Step 2								

KEY

KEY

ATTACHMENT 3
Page 1 of 1

Surveillance Test Review Sheet

TECHNICAL SPECIFICATION REFERENCE	MODE(S) REQUIRING TEST:
SR 3.2.4.1	1 (>50% Rated Thermal Power)

TEST RESULTS (TO BE COMPLETED BY TEST PERFORMER)

PERFORMED BY: Stanley Jackson / [signature] DATE/TIME: TODAY / NOW
(Print) (Signature)

COMPONENT OR TRAIN TESTED (if applicable) N/A

☒ ENTIRE STP PERFORMED ☒ FOR SURVEILLANCE CREDIT

☐ PARTIAL STP PERFORMED ☐ NOT FOR SURVEILLANCE CREDIT

REASON FOR PARTIAL ______

TEST COMPLETED ☐ Satisfactory ☒ Unsatisfactory

☒ The following deficiencies occurred

Upper QPTR does NOT meet acceptance criteria.

☐ Corrective action taken or initiated

CR# 123456 written

SHIFT SUPERVISOR/ SHIFT SUPPORT SUPERVISOR REVIEW

☐ Procedure properly completed and satisfactory per step 9.1 of FNP-0-AP-5

☐ Comments ______

REVIEWED BY: ______ / ______ DATE: ______
(Print) (Signature)

*Reviewer must be AP-31 Level II certified & cannot be the Performing Individual

ENGINEERING SUPPORT

GROUP SCREENING: SCREENED BY ______ DATE ______

(IF APPLICABLE)

☐ Comments ______

KEY

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be $\leq$ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPTR not within limit.	A.1 Limit THERMAL POWER to $\geq$ 3% below RTP for each 1% of QPTR > 1.00.	2 hours after each QPTR determination
	<u>AND</u>	
	A.2 Determine QPTR.	Once per 12 hours
	<u>AND</u>	
	A.3 Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Action A.1 <u>AND</u> Once per 7 days thereafter
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		
	A.5	----------NOTES----------- 1. Perform Required Action A.5 only after Required Action A.4 is completed. 2. Required Action A.6 shall be completed if Required Action A.5 is performed. -------------------------------- Normalize excore detectors to restore QPTR to within limits.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	<u>AND</u>		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 -----------NOTE------------ Perform Required Action A.6 only after Required Action A.5 is completed. -------------------------------- Perform SR 3.2.1.1 and SR 3.2.2.1.	 24 hours after achieving equilibrium conditions at RTP <u>OR</u> Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to $<$ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	------------------------------NOTES------------------------------ 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program
SR 3.2.4.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. --- Confirm that the normalized symmetric power distribution is consistent with QPTR.	In accordance with the Surveillance Frequency Control Program

FARLEY
Unit 1

SAFETY RELATED

FNP-1-STP-7.0

Quadrant Power Tilt Ratio Calculation

VERSION 23.0

Special Considerations:

This is an upgraded procedure. Exercise increased awareness during initial use due to potential technical and/or sequential changes. After initial use, provide comments to the procedure upgrade team.

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous	**ALL**
Reference	**NONE**
Information	**NONE**

Approval: David L Reed (Approved By) 10/11/13 (Date)

Effective Date:

OPERATIONS
Responsible Department

VERSION SUMMARY
PVR 23.0 DESCRIPTION
Updated to fleet template and writer's guide

TABLE OF CONTENTS

1.0 PURPOSE

- To determine the quadrant power tilt ratio using power range nuclear instrumentation.
- Acceptance Criteria for this test is the quadrant power tilt ratio shall be ≤ 1.020.

2.0 PRECAUTIONS AND LIMITATIONS

1. Reactor power, rod position and reactor coolant temperature should be constant while taking data. ☐
2. A QPTR calculation should be done prior to rescaling of Power Range Nuclear Instruments, and after completing the rescaling of ALL Power Ranges Nuclear Instruments. A QPTR calculation performed between individual Power Range rescaling may provide erroneous results. ☐
3. IF one Power Range NI is inoperable AND thermal power is ≤ 75% RTP, the remaining power range channels may be used for calculating QPTR. **(SR 3.2.4.1)** ☐
4. Above 75% RTP, with one Power Range NI inoperable, QPTR must be determined by SR 3.2.4.2. ☐
5. The SM/SS shall be notified if any acceptance criteria are NOT satisfied. ☐

3.0 INITIAL CONDITIONS

1. The version of this procedure has been verified to be the current version. **(OR 1-98-498)** ______
2. This procedure has been verified to be the correct procedure for the task. **(OR 1-98-498)** ______
3. This procedure has been verified to be the correct unit for the task. **(OR 1-98-498)** ______

NOTE

This STP may be performed at less than 50% power for verification of power range instrument indications. In this case, the STP is NOT for surveillance credit. ☐

4. Unit 1 is above 50% of rated thermal power. ______
5. IF DVM is used to collect data, I&C has obtained a Fluke 45 or equivalent with shielded test leads with NO exposed metal connectors. ______

DVM Serial number ______________ Cal. due date ____________

3.0 INITIAL CONDITIONS (continued)

6. This procedure may contain previously evaluated Critical Steps that may not be applicable in certain plant conditions. The evaluation of this procedure for Critical Steps is performed during the Pre-Job briefing. The decision concerning how to address error precursors for critical steps should be governed by NMP-GM-005-GL03, Human Performance Tools. ______

NOTE

Asterisked (*) steps are those associated with Acceptance Criteria. ☐

4.0 <u>INSTRUCTIONS</u>

4.1 QPTR Determination Using The IPC.

NOTES

Section 4.2, QPTR Determination Using Manual Calculation: should be used to calculate QPTR when the IPC QPTR application is unavailable. ☐

1. **Open** the QPTR AND TILT FACTORS application on the IPC Applications Menu. ______

2. **Check** the following:

   - UPPER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

   - LOWER QPTR data indicates GOOD quality as indicated by affected points displayed in green. ______

3. <u>IF</u> QPTR data is <u>NOT</u> GOOD quality, **go to** Section 4.2, QPTR Determination Using Manual Calculation: ______

4. <u>IF</u> QPTR data is GOOD quality, perform the following:

   a. **Click** PRINT EXCORE REPORT button. ______

   b. **Include** printed Excore Report with this procedure. ______

   c. **Go to** Section 4.3. ______

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE and THERMAL POWER ≤ 75% RTP, the remaining three power range channels may be used for calculating QPTR. ☐

4.2 QPTR Determination Using Manual Calculation:

1. **Calculate** QPTR using Attachment 1, Quadrant Power Tilt Ratio Calculation without Plant Computer _______

2. **Go to** Section 4.3. _______

4.3 Determination Of QPTR Acceptance Criteria:

NOTE

QPTR value displayed by the IPC utilizes 3 decimal places (to the thousandths place). If the QPTR value displayed is, for example 1.021, this would exceed the limit of 1.02 and require performance of the LCO 3.2.4 Condition A Required Actions.
(NL-10-0406, dated 2/26/2010) ☐

1. ***Check** Excore Maximum Quadrant Power Tilt Ratio ≤ 1.020 on either the EXCORE REPORT OR Attachment 1. _______

ACCEPTANCE CRITERIA

Maximum value of UPPER or LOWER Quadrant Power Tilt Ratio shall be ≤ 1.020.

NOTE

Asterisked (*) steps are those associated with Acceptance Criteria. ☐

5.0 ACCEPTANCE CRITERIA

The quadrant power tilt ratio shall be ≤ 1.020.

6.0 RECORDS

Documents created using this procedure will become QA Records when completed unless otherwise stated. The procedures and documents are considered complete when issued in DMS.

QA Record (X)	Non-QA Record (X)	Record Generated	Retention Time	R-Type
X		FNP-1-STP-7.0	LP	H06.045

7.0 REFERENCES

- FSAR - Chapter 4.4.2.4
- Technical Specification 3.2.4

Quadrant Power Tilt Ratio Calculation without Plant Computer

NOTE

QPTR may be determined using normalized currents from Curves 71A, 71B, 71C, 71D AND either of the following:

- Indicated detector current meter data. ☐
- Detector currents read by DVM using Attachment 2. ☐

1. **Obtain** normalized currents from Curve 71(A, B, C, D). _______

2. **Enter** normalized currents from Curve 71 on the Calculation Sheet. _______

NOTE

With input from one Power Range Neutron Flux channel INOPERABLE AND THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. ☐

3. **Perform** the following:

   a. IF available, **enter** detector currents indicated on POWER RANGE B drawer meters on the Calculation Sheet for each of the following:

      - N1C55NI0041, N41B DETECTOR A, (Upper) ☐
      - N1C55NI0041, N41B DETECTOR B, (Lower) ☐
      - N1C55NI0042, N42B DETECTOR A, (Upper) ☐
      - N1C55NI0042, N42B DETECTOR B, (Lower) ☐
      - N1C55NI0043, N43B DETECTOR A, (Upper) ☐
      - N1C55NI0043, N43B DETECTOR B, (Lower) ☐
      - N1C55NI0044, N44B DETECTOR A, (Upper) ☐
      - N1C55NI0044, N44B DETECTOR B, (Lower) ☐

CAUTION

DVM readings may be taken in only one drawer at a time. ☐

   b. IF any NI current reading not available on the POWER RANGE B drawer, **enter** detector currents obtained by I&C using Attachment 2 for the affected detectors. _______

4. **Enter** total number of operable detectors in space provided on the Calculation Sheet. _______

Quadrant Power Tilt Ratio Calculation without Plant Computer

5. **Calculate** the following:

   - Upper Quadrant Power Tilt Ratio. ______

   - Lower Quadrant Power Tilt Ratio. ______

6. ***Record** the greater of the upper or lower Quadrant Power Tilt Ratio value in the space provided on the Calculation Sheet. ______

ACCEPTANCE CRITERIA
Maximum value of upper or lower Quadrant Power Tilt Ratio shall be ≤ 1.020.

7. **Record** the Power Level (Avg) in the space provided. ______

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

UPPER QUADRANT POWER TILT

POWER RANGE B Drawer	UPPER DET Indicated Current	÷	*UPPER DET 100% Current	=	UPPER DET Calibrated Output
N41	Detector A	÷	N41T	=	
N42	Detector A	÷	N42T	=	
N43	Detector A	÷	N43T	=	
N44	Detector A	÷	N44T	=	

		Total Number Operable Upper Detectors		$\frac{1}{\text{Average Upper Detector Calibrated Output}}$	X	Maximum Upper Detector Calibrated Output	=	Upper Quadrant Power Tilt Ratio
Total Upper Detector Calibrated Output	=	÷	=	$\frac{1}{\quad}$	X		=	

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Calculation Performed Using: Meter Data/DVM Data (Circle One)

LOWER QUADRANT POWER TILT

POWER RANGE B Drawer	LOWER DET Indicated Current	÷	*LOWER DET 100% Current	=	LOWER DET Calibrated Output
N41	Detector B	÷	N41B	=	
N42	Detector B	÷	N42B	=	
N43	Detector B	÷	N43B	=	
N44	Detector B	÷	N44B	=	

			Total Number Operable Lower Detectors		1 / Average Lower Detector Calibrated Output	X	Maximum Lower Detector Calibrated Output	=	Lower Quadrant Power Tilt Ratio
Total Lower Detector Calibrated Output	=		÷	=	1 /	X		=	

*Obtained from Curve 71(A, B, C, D), 0% AFD Current

Quadrant Power Tilt Ratio Calculation without Plant Computer

Calculation Sheet

Upper QPTR

Lower QPTR

Maximum of Upper or Lower QPTR

*

ACCEPTANCE CRITERIA
Maximum of Upper or Lower Quadrant Power Tilt Ratio does not exceed 1.020.

% Reactor Power ____________

ATTACHMENT 2
Page 1 of 2

Using A DVM To Obtain Detector Current Values

NOTE
Detector current values may be obtained for as many drawers as required. Unused spaces in the Table should be marked N/A. ☐

CAUTIONS
- DVM readings may be taken in only one drawer at a time. ☐
- A Fluke 8600 shall NOT be used to obtain currents ☐

1. Using a Fluke 45 or equivalent AND shielded test leads **connect** to obtain detector voltage readings as follows:

NOTE
Voltage values should be in the 2 to 3 volt range. ☐

a. For Upper Detector **connect** to TP301 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

b. For Lower Detector **connect** to TP302 (+) and TP305 (-). ______ I&C

(1) **Record** indicated voltage in appropriate space of table on page 2 of 2. ______ I&C

ATTACHMENT 2
Page 2 of 2

Using A DVM To Obtain Detector Current Values

NOTE

The following formula is used to calculate detector currents:

$$\frac{\text{Measured Detector Voltage}}{2.083} \times \text{Curve 71 "0\% AFD, 100\% Current" Value} = \text{Calculated Detector Current}$$ ☐

2. Using the 0% AFD, 100% current value from Curve 71, **perform** the following:

   a. **Calculate** the detector current value. ______

   b. **Record** in appropriate space of table below. ______

	N41		N42		N43		N44	
	Upper Detector A N41T	Lower Detector B N41B	Upper Detector A N42T	Lower Detector B N42B	Upper Detector A N43T	Lower Detector B N43B	Upper Detector A N44T	Lower Detector B N44B
	DVM Voltage		DVM Voltage		DVM Voltage		DVM Voltage	
Step 1								
	Calculated Current		Calculated Current		Calculated Current		Calculated Current	
Step 2								

ATTACHMENT 3
Page 1 of 1

Surveillance Test Review Sheet

TECHNICAL SPECIFICATION REFERENCE	MODE(S) REQUIRING TEST:
SR 3.2.4.1	1 (>50% Rated Thermal Power)

TEST RESULTS (TO BE COMPLETED BY TEST PERFORMER)

PERFORMED BY: ______________ / ______________ DATE/TIME: ______ / ______
(Print) (Signature)

COMPONENT OR TRAIN TESTED (if applicable) ______________

☐ ENTIRE STP PERFORMED ☐ FOR SURVEILLANCE CREDIT
☐ PARTIAL STP PERFORMED ☐ NOT FOR SURVEILLANCE CREDIT

REASON FOR PARTIAL ______________

TEST COMPLETED ☐ Satisfactory ☐ Unsatisfactory

☐ The following deficiencies occurred

____

____

☐ Corrective action taken or initiated

____

____

SHIFT SUPERVISOR/ SHIFT SUPPORT SUPERVISOR REVIEW

☐ Procedure properly completed and satisfactory per step 9.1 of FNP-0-AP-5

☐ Comments ______________

REVIEWED BY: ______________ / ______________ DATE: ______________
(Print) (Signature)

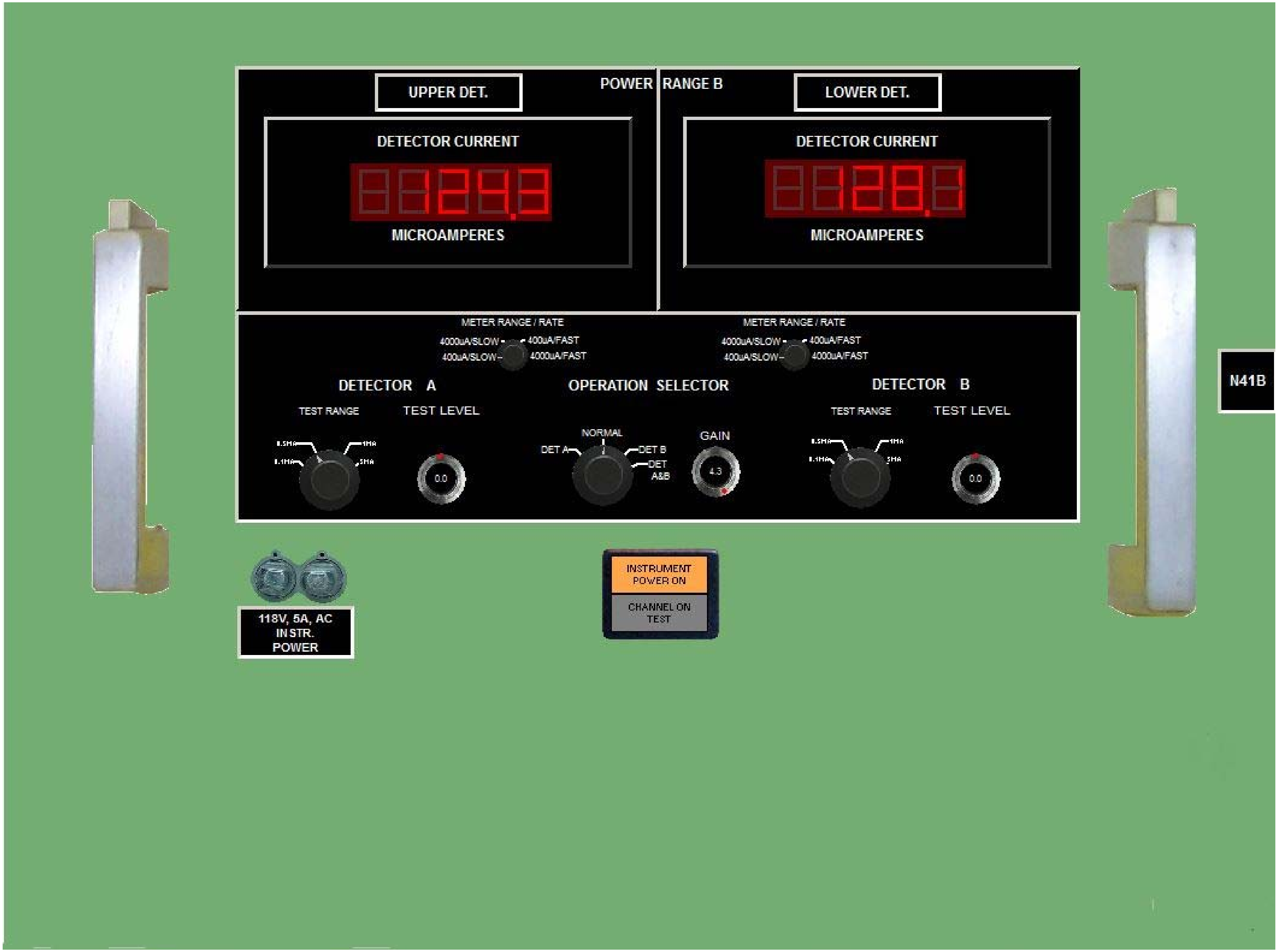
*Reviewer must be AP-31 Level II certified & cannot be the Performing Individual

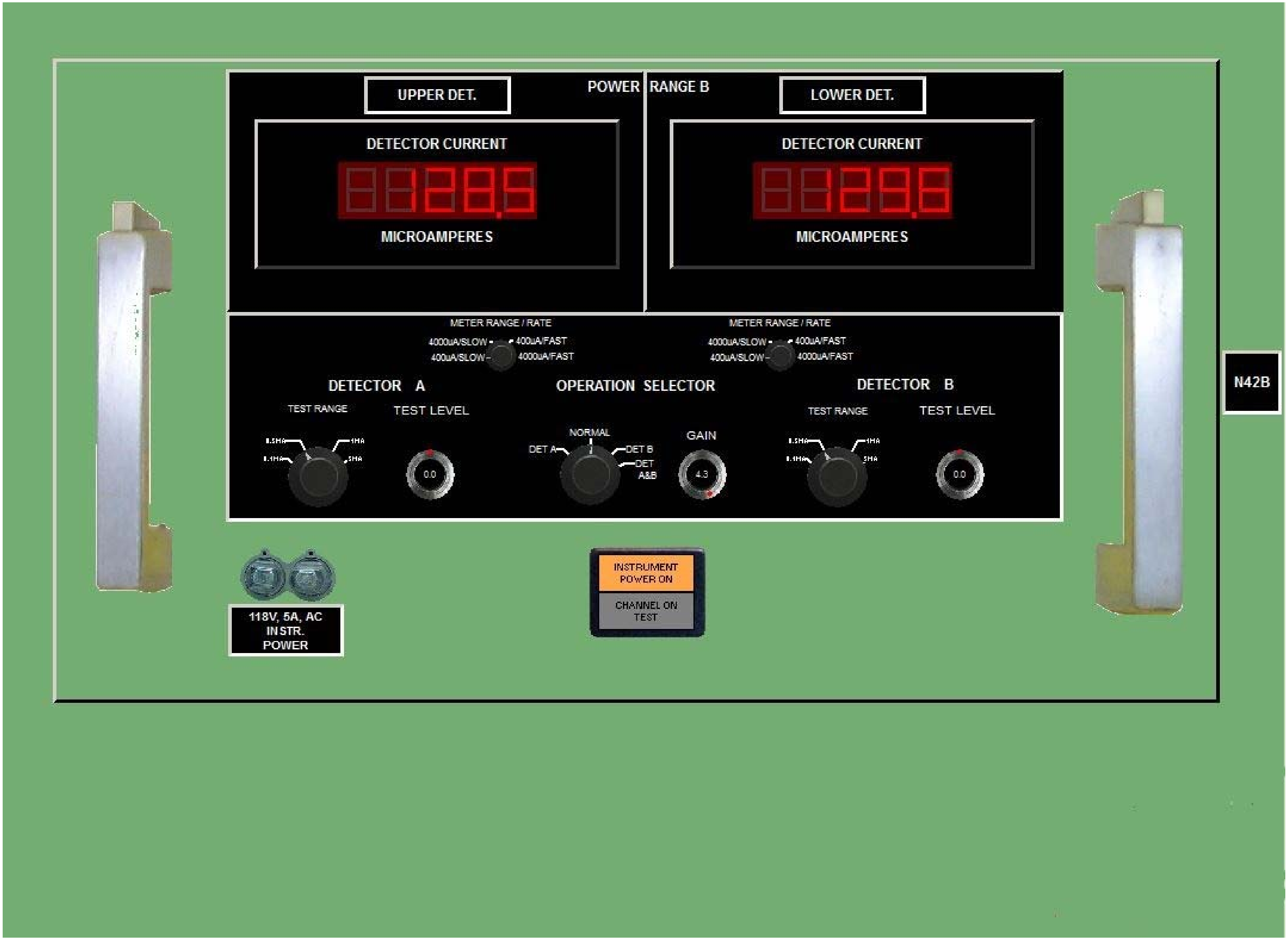
ENGINEERING SUPPORT

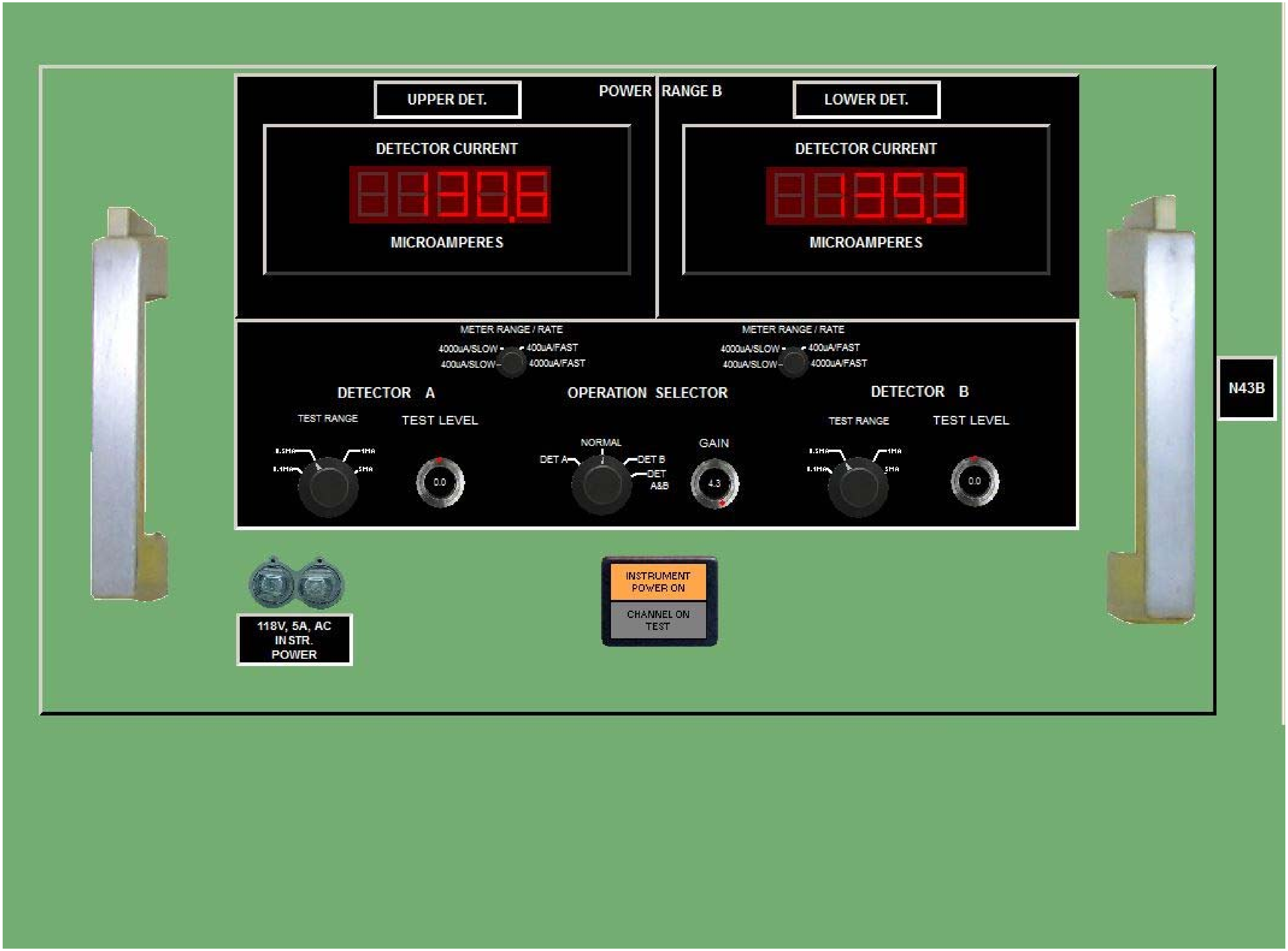
GROUP SCREENING: SCREENED BY ______________ DATE ______

(IF APPLICABLE)

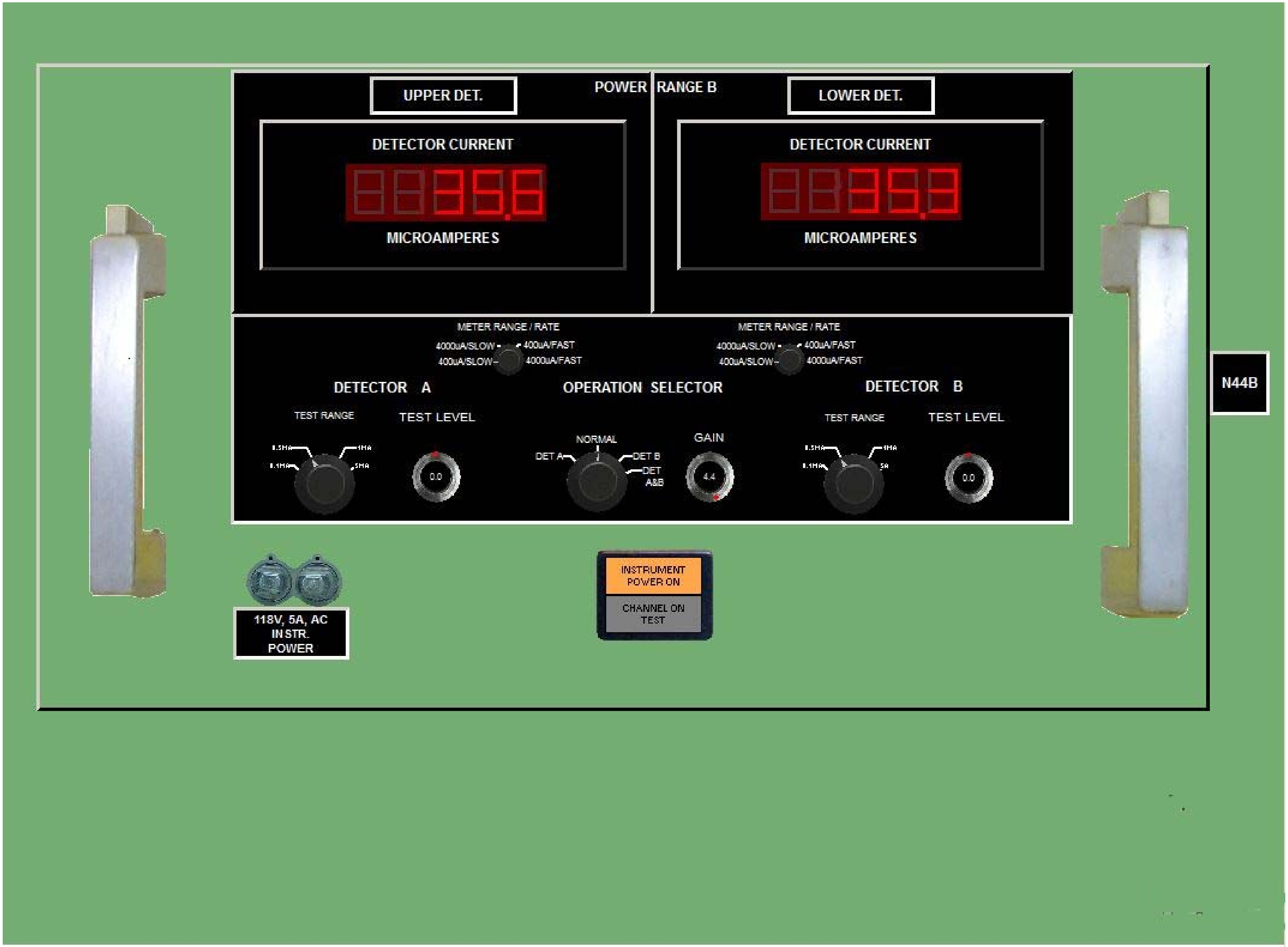
☐ Comments ______________

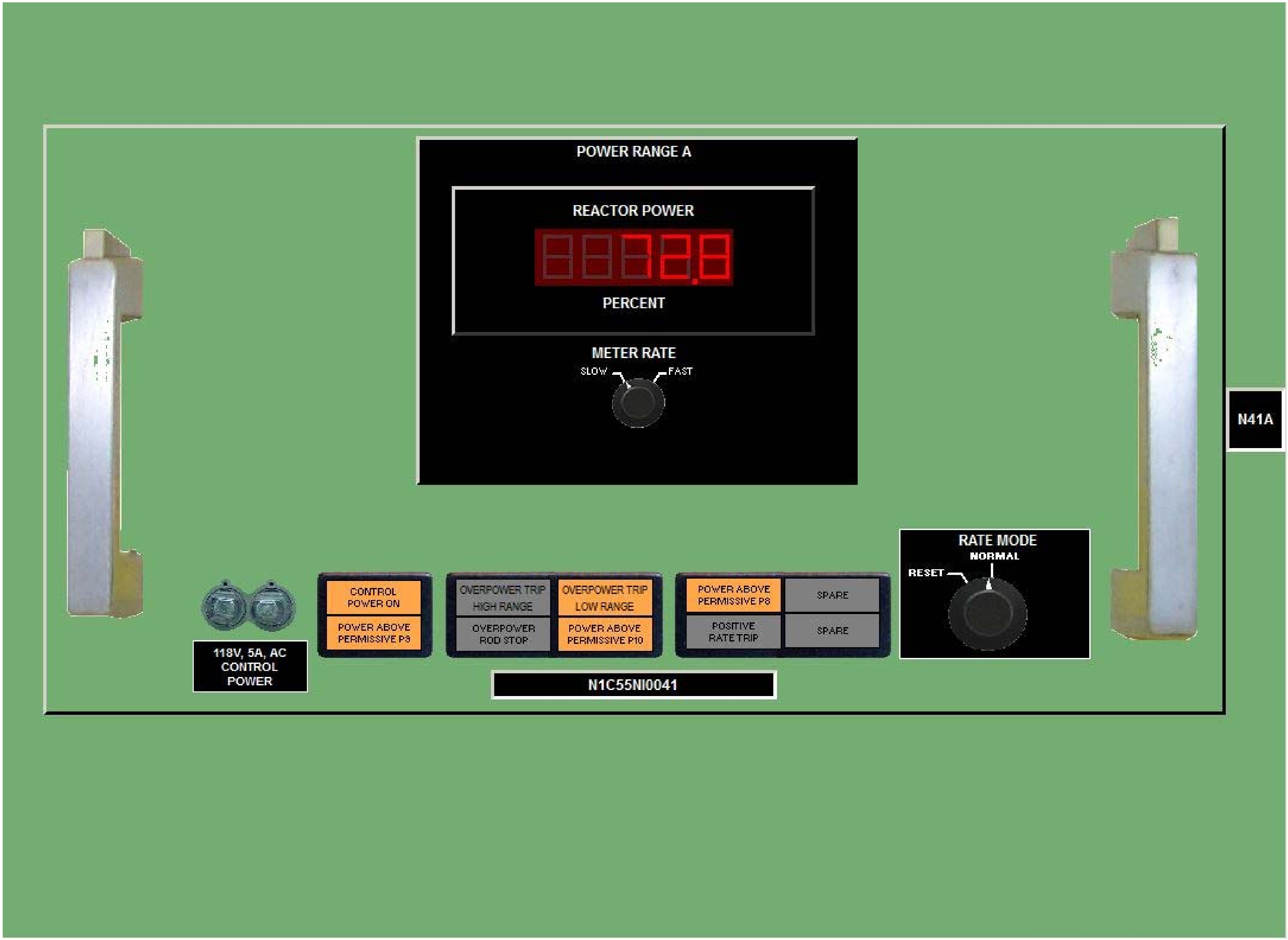

UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
OPERATION SELECTOR
NORMAL
DET A
DET B
DET A&B
GAIN
DETECTOR B
TEST RANGE
TEST LEVEL
N41B
118V, 5A, AC
INSTR.
POWER
INSTRUMENT POWER ON
CHANNEL ON TEST


UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
OPERATION SELECTOR
NORMAL
DET A
DET B
DET A&B
GAIN
DETECTOR B
TEST RANGE
TEST LEVEL
N42B
118V, 5A, AC INSTR. POWER
INSTRUMENT POWER ON
CHANNEL ON TEST

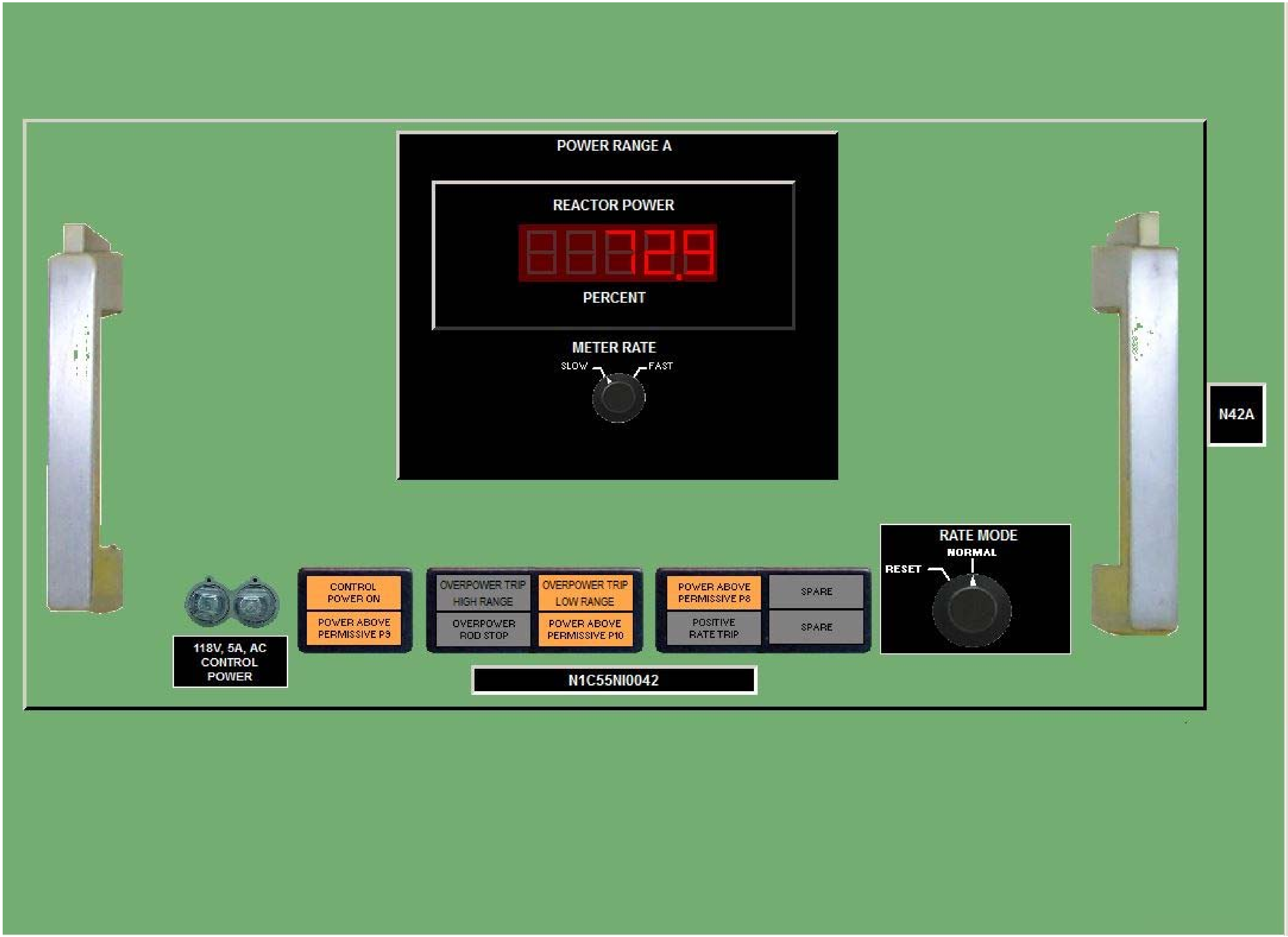


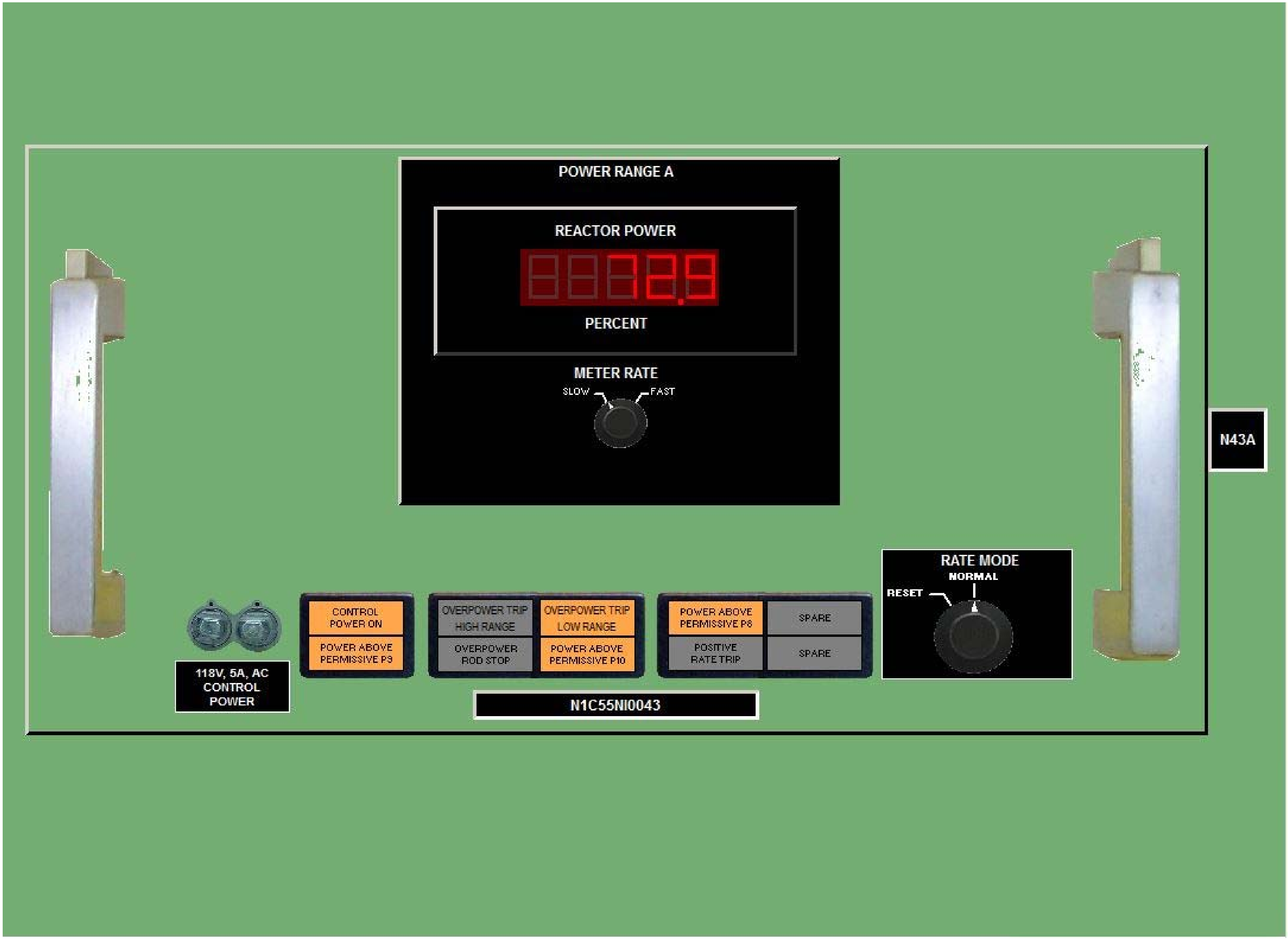
UPPER DET.
POWER RANGE B
LOWER DET.
DETECTOR CURRENT
MICROAMPERES
DETECTOR CURRENT
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
N43B
DETECTOR A
OPERATION SELECTOR
DETECTOR B
TEST RANGE
TEST LEVEL
NORMAL
DET A
DET B
DET A&B
GAIN
TEST RANGE
TEST LEVEL
118V, 5A, AC
IN STR.
POWER
INSTRUMENT POWER ON
CHANNEL ON TEST

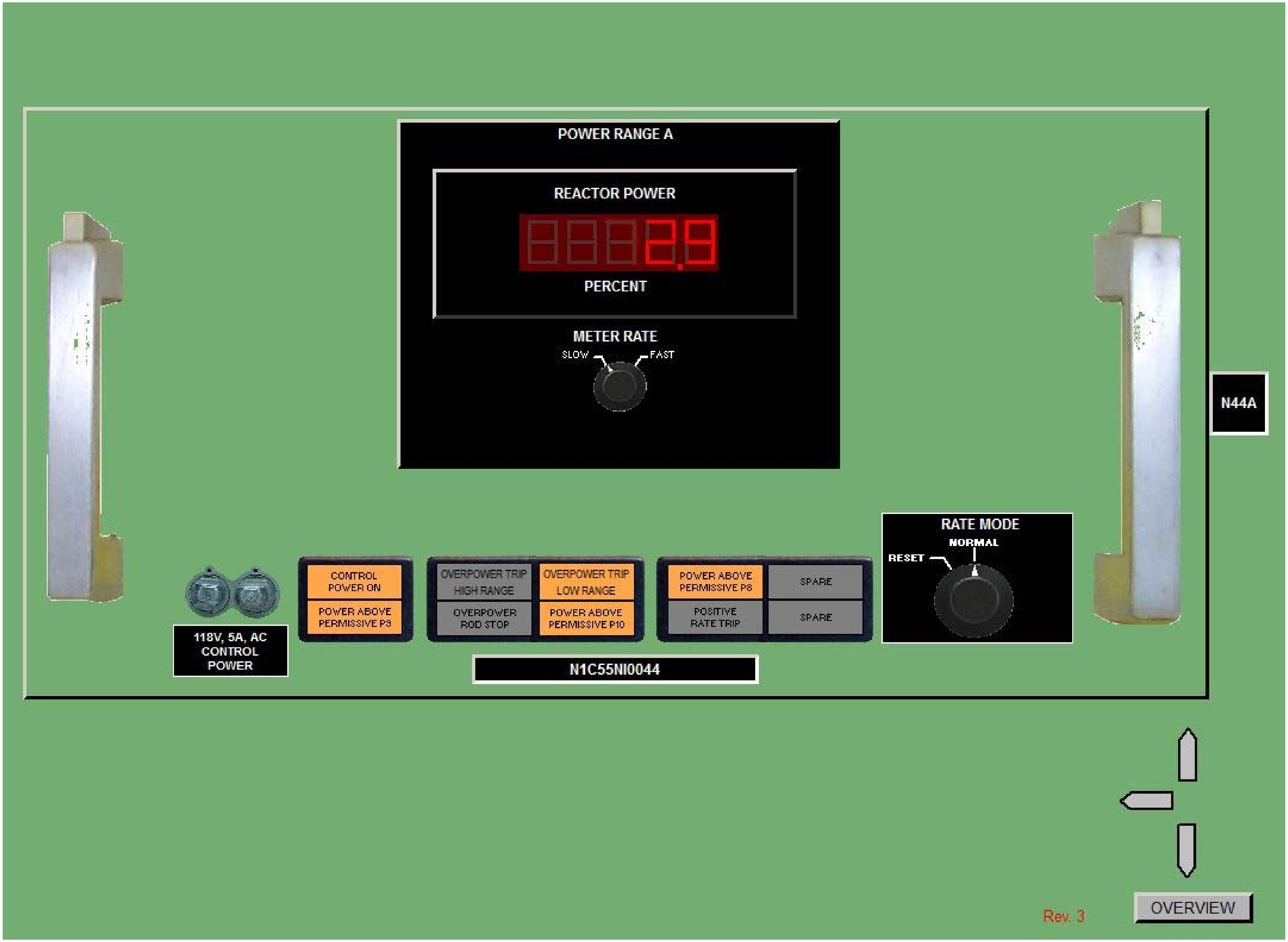

UPPER DET.
POWER
RANGE B
LOWER DET.
DETECTOR CURRENT
35.6
MICROAMPERES
DETECTOR CURRENT
35.3
MICROAMPERES
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
METER RANGE / RATE
4000uA/SLOW
400uA/FAST
400uA/SLOW
4000uA/FAST
DETECTOR A
TEST RANGE
TEST LEVEL
OPERATION SELECTOR
NORMAL
DET A
DET B
DET A&B
GAIN
DETECTOR B
TEST RANGE
TEST LEVEL
N44B
118V, 5A, AC
INSTR.
POWER
INSTRUMENT
POWER ON
CHANNEL ON
TEST



POWER RANGE A
REACTOR POWER
72.8
PERCENT
METER RATE
SLOW
FAST
N41A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0041


POWER RANGE A
REACTOR POWER
72.9
PERCENT
METER RATE
SLOW
FAST
N42A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0042


POWER RANGE A
REACTOR POWER
12.9
PERCENT
METER RATE
SLOW
FAST
N43A
RATE MODE
NORMAL
RESET
CONTROL POWER ON
POWER ABOVE PERMISSIVE P9
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
118V, 5A, AC CONTROL POWER
N1C55NI0043



POWER RANGE A
REACTOR POWER
PERCENT
METER RATE
SLOW
FAST
N44A
CONTROL POWER ON
POWER ABOVE PERMISSIVE P3
OVERPOWER TRIP HIGH RANGE
OVERPOWER ROD STOP
OVERPOWER TRIP LOW RANGE
POWER ABOVE PERMISSIVE P10
POWER ABOVE PERMISSIVE P8
POSITIVE RATE TRIP
SPARE
SPARE
RATE MODE
NORMAL
RESET
118V, 5A, AC CONTROL POWER
N1C55NI0044
Rev. 3
OVERVIEW

UNIT 1 VOLUME 1 CURVE 71A

PRESENT NIS CHANNEL N41 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14
LAW 8-22-14 PT Reactor Engineering Supervisor Date
Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N41T	215.71	187.44	159.18	86.0324	20.6460	0.6194
N41B	148.89	185.63	222.36			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N41T	I Det =	0.9423	* AO +	187.4449
N41B	I Det =	-1.2245	* AO +	185.6270

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-41 Calibration are FNP-1-IMP-228.8 & FNP-1-STP-228.5

Curve Placed in Effect:

_______________ / _______________
Shift Supervisor Date / Time
(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71B

PRESENT NIS CHANNEL N42 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14
Reactor Engineering Supervisor Date
Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N42T	220.92	191.11	161.30	82.8203	19.8780	0.5963
N42B	148.30	186.84	225.38			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N42T	I Det =	0.9937	* AO +	191.1122
N42B	I Det =	-1.2846	* AO +	186.8399

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-42 calibration are FNP-1-IMP-228.9 & FNP-1-STP-228.6

Curve Placed in Effect:

_______________ / _______________
Shift Supervisor Date / Time
(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71C

PRESENT NIS CHANNEL N43 CURRENT SETTINGS

Rev. 36 08/22/2014 SRM

Approved: [signature] 8-22-14

Reactor Engineering Supervisor Date

Signed for Brian Kern

Channel	AFD % (Values at 100% Power) 30	0	-30	K	Computer Constant	Gpot
	Detector Current					
N43T	213.18	185.03	156.87	85.6130	20.5530	0.6164
N43B	153.54	191.51	229.48			

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N43T	I Det =	**0.9385**	* AO +	**185.0294**
N43B	I Det =	**-1.2657**	* AO +	**191.5063**

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:

1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-43 calibration are FNP-1-IMP-228.10 and FNP-1-STP-228.7

Curve Placed in Effect:

______________________ / ______________________

Shift Supervisor Date / Time

(To be completed following scaling in rack)

UNIT 1 VOLUME 1 CURVE 71D

PRESENT NIS CHANNEL N44 CURRENT SETTINGS

Rev. 38 08/22/2014 SRM

Approved: [signature] 8-22-14

Reactor Engineering Supervisor Date

Signed for Brian Kern

Channel	AFD % (Values at 100% Power) Detector Current 30	0	-30	K	Computer Constant	Gpot
N44T	233.39	201.99	170.60	82.0409	19.6930	
N44B	154.85	196.07	237.29			N/A

Revised Detector Equations

Channel	I Det =	(M)	* AO +	IO
N44T	I Det =	**1.0466**	* AO +	**201.9949**
N44B	I Det =	**-1.3740**	* AO +	**196.0726**

THIS CURVE IS FOR CYCLE 26 EXCORE CHANNEL

RENORMALIZATION CALCULATED PER FNP-1-STP-121

NOTES:
1) At 100% Power AFD% = AO%
2) T refers to the Top or Upper Detector, and B refers to the Bottom or Lower Detector
3) I&C Procedures for N-44 Calibration are FNP-1-IMP-228.11 & FNP-1-STP-228.8

Curve Placed in Effect:

______________________ / ______________________

Shift Supervisor Date / Time

(To be completed following scaling in rack)

A.3 RO - SRO

TITLE: Determine the correct RWP, total projected dose And determine if an oil addition and venting can be performed to the 2A RHR pump without exceeding limits defined.
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 20 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☐ TIME CRITICAL ☐ PRA

JPM DIRECTIONS:

1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Requiring the examinee to acquire the required materials may or may not be included as part of the JPM.

TASK STANDARD: Upon successful completion of this JPM, the examinee will perform the following for the task of adding oil to the 2A RHR pump and venting the suction:

- Identify the location of Q2E11V100A
- Identify the correct RWP to perform the task.
- Calculate the total projected dose for the job.
- Determine if the task can or cannot be performed without exceeding Administrative Limits or RWP limits on a single entry, and if NOT then state the reason.

Examinee:		
Overall JPM Performance:	**Satisfactory** ☐	**Unsatisfactory** ☐
Evaluator Comments (attach additional sheets if necessary)		

EXAMINER: ________________________

Developer	S Jackson	Date: 4/9/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to **Determine the correct RWP, total projected dose And determine if an oil addition and venting can be performed to the 2A RHR pump without exceeding limits defined.**

The conditions under which this task is to be performed are:

1. You are a trainee on shift and will be accomplishing the following task under instruction.
2. You are qualified as a Fully Documented Radiation Worker.
3. You will be draining and adding oil to the 2A RHR Pump Motor upper and lower reservoirs and venting the suction of the 2A RHR pump.
4. All needed tools, oil, and equipment have been staged.
5. All necessary briefings to perform the task have been completed.
6. Your accumulated dose for this year to date is 1260 mRem.
7. Contamination levels: All areas are less than ALPHA 3 levels and < 200 dpm/100 cm^2.
8. The following tasks are required to be performed:

#	TASK	TIME REQUIRED	DOSE RATE
1	Drain and fill the RHR pump motor (upper reservoir)	5 min	25 mR/hr
2	Drain and fill the RHR pump motor (lower reservoir)	15 min	60 mR/hr
3	Remove pipe cap, attach hose to Q2E11V100A, and open the vent valves, Q2E11V100A and Q2E11V100B until air free water issues from the vent.	25 min	120 mR/hr

Note: Assume no additional dose received while traveling between tasks.

Your task is to perform all of the following and DOCUMENT your conclusions on the table provided:

a. Identify the location (room) of Q2E11V100A, CTMT SUMP TO 2A RHR PUMP HDR VENT ISO.
b. Select the correct RWP to use for this task.
c. For yourself ONLY, calculate the Total projected dose to perform this task.
d. Determine whether the task can or cannot be performed without exceeding the Farley Administrative Dose Limit or RWP limits. If the task cannot be performed, then state the reason.

INITIATING CUE: **"IF you have no questions, you may begin."**

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	(CIRCLE)
_____ **START TIME**		
* 1. Identifies the location Q2E11V100A	Using MAXIMO, or FNP-2-SOP-7.0A or other methods, identifies the location of Q2E11V100A. e.g.: • 83' Foot elevation in the 2A RHR pump room OR • Room 2131	S / U
* 2. Determines RWP to use.	Reviews the dose rates and identifies that the highest General Area dose rate for the jobs to be performed is 120 mR/hr. Determines that the task will require a High Radiation Area entry. References the RWPs and determines that RWP 15-0101 is a Training RWP, but it cannot be used for a High Radiation Area entry. Determines that RWP 15-0503 has allowance for OPS Activities in High Radiation Areas, and is the correct RWP to use.	S / U

Total dose from task calculation:

Dose-upper oil addition + Dose-lower oil addition + Dose-venting = Total dose for the task

1. 5 minutes * 25 mRem/ hr * 1 hr/60 minutes = **2.08 mRem (dose at jobsite) {2 – 2.1}**
2. 15 minutes * 60 mRem/ hr * 1 hr/60 minutes = **15 mRem (dose at jobsite) { no range }**
3. 25 minutes * 120 mRem/ hr * 1 hr/60 minutes = **50 mRem (dose at jobsite) { no range }**

2.08 + 15 + 50 = Total Dose = **67 to 67.1 mRem total**

EVALUATION CHECKLIST

ELEMENTS:	STANDARDS:	(CIRCLE)
* 3. Calculates total projected dose.	Calculates dose received while performing the job. **Documents the total of 67 mRem** {RANGE 67 – 67.1 mRem}	S / U
* 4. Determine if any dose limits will be exceeded by performing the task.	Determines if allowable dose limits will be exceed: — **Admin dose limit** Total dose = 1260 + 67.1 = 1327.1 mR 1327.1 mR < Admin dose limit of 2000 mR. — **RWP Task dose limit** 67.1 mR < RWP 15-0503 Task dose limit of 90 mR — **RWP Task dose rate limit** 120 mR/hr < RWP 15-0503 Task dose rate limit of 140 mR/hr. **Determines** that dose limits of the RWP will not be exceeded. *IDENTIFIES that the task actions can be completed as assigned by circling YES.	S / U

Total ANNUAL dose:

(1260 accumulated) + 67.1 = 1327.1 mR {1327 – 1327.1}

FNP Administrative Annual Dose limit from FNP-0-M-001, Southern Nuclear Company Joseph M. Farley Nuclear Plant Health Physics Manual, is 2000 mR for a Fully Documented Radiation worker.

_____ STOP TIME

Terminate when all elements of the task have been completed.

CRITICAL ELEMENTS: Critical Elements are denoted with an asterisk (∗) before the element number.

GENERAL REFERENCES:

1. FNP-0-M-001, v19.0
2. KA: G2.3.4 – 3.2 / 3.7
   G2.3.7 – 3.5 / 3.6

GENERAL TOOLS AND EQUIPMENT:

1. Calculator
2. RWP 15-0503 and 15-0101 (For Training USE ONLY)
3. Health Physics Manual, FNP-0-M-001, v19.0.

Critical ELEMENT justification:

STEP	**Evaluation**
1.	**Critical:** Task completion: required to determine proper location for the task given
2.	**Critical:** Task completion: required to determine proper Radiation Work Permit for the task given.
3.	**Critical:** Task completion: required to determine the total projected dose.
4.	**Critical:** Task completion: required to identify that the task can be done within limits permitting task completion.

KEY

Determination of Task Performance	
Q2E11V100A, CTMT SUMP TO 2A RHR PUMP HDR VENT **is** located:	**2A RHR Pump RM** **{Also acceptable: RM 2131}** **(Room)**
CORRECT RWP to use **(CIRCLE the correct RWP)**	**15-0101** **15-0503**
Projected dose for this task	***67 to 67.1 mRem** **{range of 67-67.1 mRem}**
Can you complete this task without exceeding limits?	**(CIRCLE ONE)** YES* NO
REASON, if applicable:	N/A

A.3

CONDITIONS

When I tell you to begin, you are to **Determine the correct RWP, total projected dose And determine if an oil addition and venting can be performed to the 2A RHR pump without exceeding limits defined.**

The conditions under which this task is to be performed are:

1. You are a trainee on shift and will be accomplishing the following task under instruction.
2. You are qualified as a Fully Documented Radiation Worker.
3. You will be draining and adding oil to the 2A RHR Pump Motor upper and lower reservoirs and venting the suction of the 2A RHR pump.
4. All needed tools, oil, and equipment have been staged.
5. All necessary briefings to perform the task have been completed.
6. Your accumulated dose for this year to date is 1260 mRem.
7. Contamination levels: All areas are less than ALPHA 3 levels and < 200 dpm/100 cm^2.
8. The following tasks are required to be performed:

#	TASK	TIME REQUIRED	DOSE RATE
1	Drain and fill the RHR pump motor (upper reservoir)	5 min	25 mR/hr
2	Drain and fill the RHR pump motor (lower reservoir)	15 min	60 mR/hr
3	Remove pipe cap, attach hose to Q2E11V100A, and open the vent valves, Q2E11V100A and Q2E11V100B until air free water issues from the vent.	25 min	120 mR/hr

Note: Assume no additional dose received while traveling between tasks.

Your task is to perform all of the following and DOCUMENT your conclusions on the table provided:

a. Identify the location (room) of Q2E11V100A, CTMT SUMP TO 2A RHR PUMP HDR VENT ISO.
b. Select the correct RWP to use for this task.
c. For yourself ONLY, calculate the Total projected dose to perform this task.
d. Determine whether the task can or cannot be performed without exceeding the Farley Administrative Dose Limit or RWP limits. If the task cannot be performed, then state the reason.

Determination of Task Performance		
Q2E11V100A, CTMT SUMP TO 2A RHR PUMP HDR VENT ISO, is located:	(Room)	
CORRECT RWP to use (CIRCLE the correct RWP)	15-0101	15-0503
Projected dose for this task		
Can you complete this task without exceeding limits?	(CIRCLE ONE) YES	NO
REASON, if applicable:		

Radiation Work Permit	Plant Farley **15-0101** **FOR TRAINING USE ONLY**	REV **99**	UNIT **0**

Job Description	Administration;This RWP is designated for administrative departments to include Executive & Senior Management, Security, Information Technology, Fleet Oversight, Safety and Health, Training, Performance Improvement, and Work Control that involve radiological work categorized as 'Low Risk' AND not specifically addressed by other RWPs. Caution: This RWP cannot be used for entries into Containment, work in High Radiation Areas, Airborne Areas, or posted Alpha Level 2 or Level 3 Contaminated Areas.
Location	RADIATION CONTROL AREAS

HP Coverage	Authorization	Briefing
INTERMITTENT	ALL	ALL

Start Date	1/1/2015 12:00 AM	End Date	12/31/2015 11:59 PM
Job Supv.	B. Thornton	EXT	6143

Radiological Conditions

AIRBORNE LEVELS: < 0.3 DAC PART AND IODINE, <1.0 DAC NOBLE GAS/TRITIUM

CONTAMINATION: < 200,000 DPM/100CM2 BETA GAMMA, LESS THAN ALPHA LEVEL 1

RAD LEVELS: LESS < 100 MREM/HR

Dosimetry

TLD and DAD

Protective Clothing Requirements

DRESS REQUIRMENTS AS HP DIRECTS

Respirators

Usage is Prohibited

TASKS

Description	DAD Alarms Dose (mr)	DAD Alarms Rate(mr/h)
NRC, INPO, ETC. ACTIVITIES	10	50
FLEET OVERSIGHT ACTIVITIES	10	50
TRAINING AND EP ACTIVITIES	25	50
WORK CONTROL ACTIVITIES	25	50
SECURITY ACTIVITIES	10	50

INSTRUCTIONS

There are certain worker instructions that are applicable to ALL RWPs. These instructions are provided at the Main RCA entrance.

Worker MUST ensure they understand and comply with these instructions at all times while working in RCAs

ENTRY INTO AREAS WITH KNOWN RADIOLOGICAL CONDITIONS GREATER THAN THOSE SPECIFIED ABOVE IS PROHIBITED AND REQUIRES AUTHORIZATION AND BRIEFING ON THE APPROPRIATE RWP.

Contact Health Physics Prior to entering any Radiation Area posted as Neutron Monitoring Required.

Exposure limits for Neutron is as follows: 10 mrem per entry and 50 mrem/hour.

The following special instructions may be deviated from with SNC HP ANSI 3.1 or higher qualified individual's permission.

___ Dressout requirements for areas >50,000 dpm/100cm2 beta/gamma: Double coveralls, or 1 set of Ultra Orex or equivalent, two sets of booties, two sets rubber shoecovers, and two sets gloves.

FOR TRAINING USE ONLY

Prepared	NRC EXAM TEAM	APPROVED	4/18/2015 12:00:00 AM **by NRC EXAM WRITER**

Radiation Work Permit	Plant Farley **15-0503** **FOR TRAINING USE ONLY**	Rev **99**	UNIT **0**

Job Description	Operations: This RWP is designated for Operations Activities involving work in High Radiation Areas and other work classified as "Medium Radiological Risk" not specifically addressed by other RWPs. CAUTION: This RWP cannot be used for entries into Containment, Locked High Radiation Areas, OR work in Alpha Level 3 Areas.
Location	RADIATION CONTROL AREAS

HP Coverage	Authorization	Briefing
INTERMITTENT	INDIVIDUAL	INDIVIDUAL

Start Date	1/1/2015 12:00 AM	End Date	12/31/201□ 11:59 PM
Job Supv.	B. Thornton	EXT	6143

Radiological Conditions

AIRBORNE LEVELS: LESS THAN 4 DAC HOURS PER ENTRY

CONTAMINATION: < 500,000 DPM/100CM2 BETA GAMMA, LESS THAN ALPHA LEVEL 3

RAD LEVELS: LESS < 1000 MREM/HR

Dosimetry

TLD and DAD

Protective Clothing Requirements

DRESS REQUIRMENTS AS HP DIRECTS

Respirators

Usage is Conditional per HP

TASKS

Description	DAD Alarms Dose (mr)	DAD Alarms Rate(mr/h)
OPS ACTIVITIES (NON HIGH RAD AREAS)	20	70
OPS TRAINING & JPM ACTIVITIES (NON HIGH RAD AREAS)	20	70
OPS ACTIVITIES (HIGH RAD AREAS)	90	140
OPS TRAINING & JPM ACTIVITIES (HIGH RAD AREAS)	90	140

INSTRUCTIONS

There are certain worker instructions that are applicable to ALL RWPs. These instructions are provided at the Main RCA entrance.

Worker MUST ensure they understand and comply with these instructions at all times while working in RCAs

All workers must receive an Initial RWP Briefing prior to using this RWP for the FIRST time.

Prior to commencing work, individuals will receive a High Rad briefing for every posted High Radiation Area to be entered.

Prior to commencing any work in Neutron Radiation Areas, individuals will contact Health Physics to ensure appropriate Neutron monitoring is performed.

Exposure limits for Neutron is as follows: 10 mrem per entry and 50 mrem/hour.

Health Physics Technician must give approval before any tool or piece of equipment is raised such that the item comes out of the water.

Health Physics Technician must give approval before any activated or high dose object is brought near the surface of the water.

When there is extended work evolutions in the SFP Room, the Operations Refueling Supervisor should consider placing the SFP on recirc to the demin.

Notify HP prior to moving any fuel or irradiated components so that HP can ensure the movement will not cause any possible streaming from the light poles in the SFP Transfer canal.

Identified Irradiated FME must have a retrieval plan approved by the HP Supervisor and can not be vacuumed to a filter without approval of HP Radwaste

DAD alarm values may be adjusted by a SNC HP ANSI 3.1 or higher qualified individual based on expected conditions but can not exceed the limits listed for this RWP.

The following special instructions may be deviated from with a SNC HP ANSI 3.1 or higher qualified individual's permission.

___ Dressout requirements for areas >50,000 dpm/100cm2 beta/gamma: Double coveralls, or 1 set of Ultra Orex or equivalent, two sets of booties, two sets rubber shoecovers, and two sets gloves.

FOR TRAINING USE ONLY

Prepared	NRC EXAM TEAM	APPROVED	4/□□/201□ 12:00:00 AM **by NRC EXAM WRITER**

A.4 SRO

TITLE: CLASSIFY AN EMERGENCY EVENT PER NMP-EP-110, EMERGENCY CLASSIFICATION
EVALUATION LOCATION: ☐ SIMULATOR ☐CONTROL ROOM ☒CLASSROOM
PROJECTED TIME: 20 MIN SIMULATOR IC NUMBER: N/A
☐ ALTERNATE PATH ☒ TIME CRITICAL ☐ PRA
∗THIS JPM IS TIME CRITICAL∗

JPM DIRECTIONS:

1. Initiation of task may be in group setting, evaluation performed individually upon completion.
2. Provide a copy of the HANDOUT and Checklist 1 - Classification Determination to allow the applicant to review the task.
3. When the applicant is ready begin, provide NMP-EP-110, Emergency Classification Determination, and NMP-EP-110-GL01, FNP EALs – ICs, Threshold Values and Basis, and allow the student to begin the task.

TASK STANDARD: Upon successful completion of this JPM, the examinee will be able to:

1. Classify an Emergency Event per NMP-EP-110, Emergency Classification Determination and Initial Action, and complete Checklist 1, Classification Determination.

Examinee:		
Overall JPM Performance:	**Satisfactory** ☐	**Unsatisfactory** ☐
Evaluator Comments (attach additional sheets if necessary)		

EXAMINER: ______________________

Developer	S Jackson	Date: 4/10/15
NRC Approval	SEE NUREG 1021 FORM ES-301-3	

CONDITIONS

When I tell you to begin, you are to CLASSIFY AN EMERGENCY EVENT PER NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION.

This task is to be performed based on the following information:

A rampdown was initiated on Unit 2 due to high RCS activity.
Current conditions are as follows:

a. Chemistry reports that RCS gross activity is 105/Ē µCi/gm.
b. R-4 has risen from 2 mr/hr to 200 mr/hr
c. R-2 is 900 mr/hr
d. R-7 is 450 mr/hr
e. The plant initiated a manual Safety Injection based on excessive RCS leakage.
f. Pressurizer pressure is stable at 1900 psig and Pressurizer level is stable with 200 gpm HHSI flow.
g. RCS Tavg is 539ºF & decreasing slowly.
h. This JPM contain Time Critical Elements.

NOTE: The classification should NOT be based on ED discretion.

Your task is to classify the event and fill out NMP-EP-110, Checklist 1, Classification Determination Form, through step 5.

EVALUATION CHECKLIST

	ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)

_______ CRITICAL START TIME

NOTE: THE TIME IT TAKES TO CLASSIFY THE EVENT IS TIME CRITICAL AND MUST BE COMPLETED IN 15 MINUTES.

	ELEMENTS	STANDARDS	RESULTS
*1.	1 Determine the appropriate Initiating Condition Matrix for classification of the event based on the current operating mode: □ HOT IC/EAL Matrix Evaluation Chart (Go To Step 2) to evaluate the Barriers) □ COLD IC/EAL Matrix Evaluation Chart (Go To Step 3) □ Both HOT & COLD IC/EAL Matrix Evaluation Chart apply (Go To Step 2)	1) Checks - □ HOT IC/EAL Matrix Evaluation Chart (Go To Step 2) to evaluate the Barriers) and proceeds to **Step 2**	S / U
2.	2 Evaluate the status of the fission product barrier using Figure 1, Fission Product Barrier Evaluation.	2) Evaluates Fission Product Barrier Evaluation.	S / U
	a Select the condition of each fission product barrier: Loss, Potential Loss or Intact on: Fuel Cladding, RCS and Containment Integrity.	a) Selects: Fuel Cladding – INTACT RCS – POTENTIAL LOSS Containment Integrity - INTACT	
*3.	2b Determine the highest applicable fission product barrier Initiating Condition (IC):	2b) Selects FA1	S / U
4.	3 Evaluate AND determine the highest applicable IC/EAL using the Matrix Evaluation Chart(s) identified in step 1 THEN Go To step 4. Hot IC#________ Unit___ and/or Cold IC# ___________ Unit___ or □ None	3) Selects: Hot IC# SU4 Unit 2	S / U

EVALUATION CHECKLIST

	ELEMENTS:	STANDARDS:	RESULTS: (CIRCLE)
Note: The Remarks section of Step 4 is NOT critical.			
*5.	4 Check the highest emergency classification level identified from either step 2b or 3: Check the highest emergency classification level identified from either step 2b or 3: Classification Based on IC# Classification Based on IC# □ General □ Alert □ Site-Area □ NOUE □ None	4) *Selects ALERT *Based on IC# FA1	S / U
	Remarks (Identify the specific EAL, as needed):	Loss or potential loss of either fuel clad or RCS	S / U
6.	5 Declare the event by approving the Emergency Classification.	5) Signs and enters today's date and time the classification is completed	S / U

_______ CRITICAL STOP TIME

Terminate JPM when Classification is completed

CRITICAL ELEMENTS: Critical Elements are denoted with an Asterisk (*) before the element number.

GENERAL REFERENCES:

1. NMP-EP-110, ver 8
2. NMP-EP-110-GL01, ver 8
3. KA: G2.4.41 RO-2.3 SRO-4.1

GENERAL TOOLS AND EQUIPMENT:

1. NMP-EP-110, ver 8
2. NMP-EP-110-GL01, ver 8 (EAL BOARD)

Critical ELEMENT justification:

1. Critical – Task Completion. Selection of proper EAL board is critical to completing proper classification.
2. Not Critical – These steps are not required to be documented to get the correct classification.
3. Critical – Task Completion. This is the proper classification.
4. Not Critical – This classification is LOWER than FA1. FA1 is the proper classification.

5a. Critical – Task completion. information provided is essential for correct Emergency Notification of state and local authorities.

5b. Not Critical – This is merely a description of the classification

6 Not Critical – Administrative action

COMMENTS:

A.4 SRO

CONDITIONS

When I tell you to begin, you are to CLASSIFY AN EMERGENCY EVENT PER NMP-EP-110, EMERGENCY CLASSIFICATION DETERMINATION.

This task is to be performed based on the following information:

A rampdown was initiated on Unit 2 due to high RCS activity.
Current conditions are as follows:

a. Chemistry reports that RCS gross activity is 105/Ē µCi/gm.
b. R-4 has risen from 2 mr/hr to 200 mr/hr
c. R-2 is 900 mr/hr
d. R-7 is 450 mr/hr
e. The plant initiated a manual Safety Injection based on excessive RCS leakage.
f. Pressurizer pressure is stable at 1900 psig and Pressurizer level is stable with 200 gpm HHSI flow.
g. RCS Tavg is 539ºF & decreasing slowly.
h. This JPM contain Time Critical Elements.

NOTE: The classification should NOT be based on ED discretion.

Your task is to classify the event and fill out NMP-EP-110, Checklist 1, Classification Determination Form, through step 5.

Emergency Classification Determination and Initial Action		NMP-EP-110
	SNC Unit S	Version 8.0
		Page 12 of 23

ATTACHMENT 1
Page 1 of 1

Checklist 1 - Classification Determination

NOTE
Key Parameters should be allowed to stabilize to accurately represent plant conditions prior to classifying an event

Initial Actions — Completed by

1. **Determine** the appropriate Initiating Condition Matrix for classification of the event based on the current operating mode: SJJ
   - ☑ HOT IC/EAL Matrix Evaluation Chart (**Go To** Step **2)** to evaluate the Barriers)
   - ☐ COLD IC/EAL Matrix Evaluation Chart (**Go To** Step **3**)
   - ☐ Both HOT & COLD IC/EAL Matrix Evaluation Chart apply (**Go To** Step **2)**
2. **Evaluate** the status of the fission product barrier using Figure 1, Fission Product Barrier Evaluation.

   a. **Select** the condition of each fission product barrier: SJJ

	LOSS	POTENTIAL LOSS	INTACT
Fuel Cladding Integrity	☐	☐	☑
Reactor Coolant System	☐	☑	☐
Containment Integrity	☐	☐	☑

   b. **Determine** the highest applicable fission product barrier Initiating Condition (IC): SJJ

   (select one) ☐ FG1 ☐ FS1 ☑ FA1 ☐ FU1 ☐ None

3. **Evaluate** AND **determine** the highest applicable IC/EAL using the Matrix Evaluation Chart(s) identified in step 1 THEN **Go To** step **4**.

   Hot IC# SU4 Unit 2 and/or Cold IC# ________ Unit___ or ☐ None — SJJ

4. **Check** the highest emergency classification level identified from either step 2b or 3: SJJ

Classification	Based on IC#	Classification	Based on IC#
☐ General		☑ Alert	FA1
☐ Site-Area		☐ NOUE	
		☐ None	N/A

Remarks (**Identify** the specific EAL, as needed): Loss or potential loss of either fuel clad or RCS

5. **Declare** the event by approving the Emergency Classification.

   APPLICANT SIGNATURE Date: ____ / TODAY / ____ Time: NOW — SJJ
   Emergency Director

6. **Obtain** Meteorological Data (not required prior to event declaration):

Wind Direction (from)_____ Wind Speed_____ Stability Class_____ Precipitation______ ____________

7. **Initiate Attachment 2, Checklist 2 - Emergency Plan Initiation**. ____________

ATTACHMENT 1
Page 1 of 1

Checklist 1 - Classification Determination

NOTE
Key Parameters should be allowed to stabilize to accurately represent plant conditions prior to classifying an event

Initial Actions Completed by

1. **Determine** the appropriate Initiating Condition Matrix for classification of the event based on the current operating mode: ____________
   - ☐ HOT IC/EAL Matrix Evaluation Chart (**Go To** Step **2)** to evaluate the Barriers)
   - ☐ COLD IC/EAL Matrix Evaluation Chart (**Go To** Step **3**)
   - ☐ Both HOT & COLD IC/EAL Matrix Evaluation Chart apply (**Go To** Step **2)**
2. **Evaluate** the status of the fission product barrier using Figure 1, Fission Product Barrier Evaluation.

   a. **Select** the condition of each fission product barrier: ____________

	LOSS	POTENTIAL LOSS	INTACT
Fuel Cladding Integrity	☐	☐	☐
Reactor Coolant System	☐	☐	☐
Containment Integrity	☐	☐	☐

   b. **Determine** the highest applicable fission product barrier Initiating Condition (IC): ____________

   (select one) ☐ FG1 ☐ FS1 ☐ FA1 ☐ FU1 ☐ None

3. **Evaluate** AND **determine** the highest applicable IC/EAL using the Matrix Evaluation Chart(s) identified in step 1 THEN **Go To** step **4**.

   Hot IC#________ Unit___ and/or Cold IC# __________ Unit___ or ☐ None ____________

4. **Check** the highest emergency classification level identified from either step 2b or 3: ____________

Classification	Based on IC#	Classification	Based on IC#
☐ General	____________	☐ Alert	____________
☐ Site-Area	____________	☐ NOUE	____________
		☐ None	N/A

Remarks (**Identify** the specific EAL, as needed):____________________________

5. **Declare** the event by approving the Emergency Classification.

   ____________________________Date: ________/______/______Time: ________ ____________
   Emergency Director

6. **Obtain** Meteorological Data (not required prior to event declaration):

Wind Direction (from)_____ Wind Speed_____ Stability Class_____ Precipitation______ ____________

7. **Initiate Attachment 2, Checklist 2 - Emergency Plan Initiation**. ____________

SNC
Unit S

NON-SAFETY RELATED

NMP-EP-110

Emergency Classification Determination and Initial Action

VERSION 8.0

Special Considerations:

Applicable to Corporate, FNP, HNP, VEGP 1-2

PROCEDURE LEVEL OF USE CLASSIFICATION PER NMP-AP-003	
CATEGORY	**SECTIONS**
Continuous	Attachment 1
Reference	Att. 2, NMP-EP-110-GL01, GL02, GL03, Att. 3
Information	Remainder of Procedure

Approval: Approved by Scott Odom — 05/28/2015
Approved By — Date

EMERGENCY PLANNING
Responsible Department

VERSION SUMMARY
PVR 6.1 **DESCRIPTION**
Procedure revision to incorporate the following editorial changes (FCR 643144): • Aligned Section 4.4.2 and Section 4.4.3 with Checklist 2. • Corrected procedure numbers for Guidelines in Section 4.2.1 Step 3 • Removed revision from NUREG 1022 (FTE 643364) and deleted duplicate guidance note already contained in Section 4.5 Step 1.
PVR 7.0 **DESCRIPTION**
Revised to incorporate lessons learned from Southern Nuclear Operating Company (SNC) Hostile Action Based (HAB) drills (CTE 791146). These changes include guidance for communication and coordination of the emergency response effort between the Shift Manager, Security and the Incident Commander. Step 1 of Checklist 2 enhanced to add direction to update the crew of potential upgrade criteria and whether or not a release is in progress. In addition to these changes the procedure incorporated various editorial changes to include clarifying document numbers in level of use section, changed “checklist items” to “checklist 2 items” in Attachment 3 (CTE 604910) and reformatted procedure to the SNC fleet standard template.
PVR 7.1 **DESCRIPTION**
Editorial change to correct references errors.
PVR 7.2 **DESCRIPTION**
Editorial change to reflect Dose Assessment references as new NMP-EP-104 and remove site specific FNP-0-EIP-9.3, 73EP-EIP-015, 73EP-EIP-108, and 91304-C. Editorial change to add reference to Hatch commitments.
PVR 8.0 **DESCRIPTION**
Added guidance allowing ED or designee to authorize deviations from normal work processes under specific conditions (IER 13-10, rec 5F), corrected Hatch commitment SNC23649 and added SNC24353

TABLE OF CONTENTS

1.0 PURPOSE/SCOPE/APPLICABILITY

The purpose of this procedure is to provide instructions for the classification of off-normal events into one of four emergency classification levels. This procedure establishes the methodology for emergency classification and delineates the initial actions required by the Emergency Director.

This procedure applies to emergency classification determinations and associated initial responses. This procedure will be utilized for actual emergencies, emergency drills/exercises, or training as required. This procedure is applicable to all SNC sites.

2.0 RESPONSIBILITIES

2.1 EMERGENCY DIRECTOR (ED)

1. The ED has the following non-delegable responsibilities:

   - The decision to declare, escalate, or terminate emergency classifications.
   - The decision to notify offsite emergency response agencies.
   - The decision to recommend protective actions to offsite authorities.
   - The decision to request federal assistance.
   - Authorization for plant personnel to exceed 10CFR20 radiation exposure limits.
   - Authorization for use of potassium iodide (KI) tablets during a declared emergency.
   - The decision to dismiss nonessential personnel from the site at an ALERT or higher emergency classification.

2.1 EMERGENCY DIRECTOR (ED) (continued)

2. The ED has the following delegable responsibilities:

   - **Maintaining** communications with offsite authorities regarding all aspects of emergency response.
   - **Providing** overall direction for management of procurement of site-needed materials, equipment, and supplies, documentation, accountability, and security function.
   - **Directing** the notification AND activation of the emergency organization; including emergency response facility activation.
   - **Coordinating** AND directing emergency operations.
   - IF requested by offsite agencies, the ED SHALL **dispatch** SNC representatives to offsite government centers.
   - **Modifying** Emergency Plan Implementing Procedures, Security Plan, Security Plan Implementing Procedures AND **adjusting** Emergency Response Organization staffing during the emergency situation
   - **Coordinating** NRC activities to reduce the duplication of effort and reduce the impact on the plant staff during the emergency situation.
   - **IF** Severe Accident Management Guidelines (SAMG) are implemented, THEN Directing the assignment of an individual as Decision Maker.
   - **Authorizing** deviations from normal work processes IF a Beyond Design Basis External Even (BDBEE), a Site Area Emergency or General Emergency has been declared and the deviation is necessary in order to take action to restore core cooling, mitigate an offsite release in progress, or maintain SFP cooling may be given. Deviations are documented on Attachment 4 and are only for normal work practices. Deviations from station license including technical specifications continue to need authorization via the 10 CFR 50.54 (x) and (y) process.

3. The Shift Manager (SM) is responsible for initial classification of events. The SM SHALL **assume** the responsibilities of the ED until relieved by another qualified ED.

4. **IF** the SM is unavailable, THEN an ED qualified person WILL **assume** the responsibilities of the ED until relieved.

5. Transfer of ED responsibilities is completed in accordance with Attachment 3, Checklist 3 - Emergency Director Transfer of Responsibilities.

6. After turnover of ED responsibilities, the SM then continues to be responsible for recognizing changes in plant conditions AND advising the ED concerning classification of events.

2.1 EMERGENCY DIRECTOR (ED) (continued)

7. Any qualified ED may assume the position of ED **after** receipt of turnover information from the off going ED.

8. The Technical Support Center (TSC) Manager AND Emergency Operations Facility (EOF) Manager are responsible for providing recommendations on emergency classifications to the ED.

3.0 INSTRUCTIONS

3.1 PRECAUTIONS / LIMITATIONS

1. This procedure establishes minimum requirements for emergency classification. The ED may use judgment as the final criterion for determining the classification of off-normal events that are NOT included in this procedure.

2. Personnel AND plant safety MUST be addressed as the highest priority, if necessary, prior to an emergency classification.

3. Classification should NOT be delayed in anticipation of either events being terminated.

4. The value of any emergency actions, which may require movement of the plant personnel, MUST be judged against the danger to personnel or nuclear safety.

5. The ED SHALL **assess**, **classify**, AND **declare** an emergency condition within 15 minutes **after** the availability of indications to plant operators that an EAL has been exceeded. The 15-minute period encompasses all assessment, classification, and declaration actions associated with making an emergency declaration from the first availability of a plant indication or receipt of a report of an off-normal condition by plant operators up to and including the declaration of the emergency. If classifications and declarations are performed away from the control room, all delays incurred in transferring information from the control room (where the alarms, indications, and reports are first received) to the Emergency Response Facility (at which declarations are made) are included within the 15 minute criterion.

6. Once indication of an abnormal condition is available, classification declaration MUST be made within 15 minutes. This time period is meant to provide sufficient time to accurately assess the emergency conditions and then evaluate the need for an emergency classification based on the assessment performed. It does NOT allow a delay of 15 minutes if the classification is recognized to be necessary.

7. When evaluating EALs that specify a duration of the off normal condition, the Emergency Director should declare the event as soon as the condition has exceeded, or it is determined that it is likely to exceed the applicable duration. The declaration process should run concurrently with the specified threshold in the EAL (i.e., the ED should be preparing paperwork, evaluating emergency classification, etc. during the time of the specified duration).

3.1 PRECAUTIONS / LIMITATIONS (continued)

8. If the ED believes a threshold is exceeded and is classifying an event when new key information becomes available, it may be appropriate to allow time for evaluation of this new information. All pertinent available indications that could change the classification level should be considered. There are no absolutes when making this call but the ED has the latitude to evaluate new information as it becomes available. Once sufficient information is available to verify that the EAL has been exceeded it should be classified promptly.

9. Emergency Plan initiation is accomplished through the use of Attachment 2, Checklist 2 - Emergency Plan Initiation. Checklist items are listed in order of priority. Actions may be taken in parallel if resources are available.

10. Events should be classified based on meeting the Initiating Condition (IC) AND Threshold Value (TV) for an EAL considering each Unit independently. IF both Units are in concurrent events, THEN the highest classification MUST be made and used for the Offsite notifications with the other Unit events noted on the Emergency Notification form.

11. Strategies for coping with extreme or extensive damage to plant components will be implemented utilizing Emergency Management Guidelines (NMP-EP-400 series) AND associated Extensive Damage Mitigation Guideline (EDMG) as appropriate.

12. A security related emergency may delay the ordering of assembly and accountability in order to protect plant personnel from the security threat. The decision to delay the order for assembly and accountability will be made by the Emergency Director. (2002342760)

13. A security incident/emergency may require the ED to modify security procedures and/or emergency plan implementing procedures. See Attachment 2, Checklist 2 - Emergency Plan Initiation, of this procedure for procedure modification instructions.

14. A Beyond Design Basis External Event (BDBEE)a Site Area Emergency or General Emergency may require the ED to authorize deviations from normal work processes in order to restore core cooling, mitigate an off-site release in progress, or maintain SFP cooling. Deviations may also be authorized to protect fission product barriers or the health and safety of the public if deemed necessary. Deviation and their basis are documented using Attachment 4, Deviating From Normal Work Processes.

3.2 CLASSIFICATION AND EMERGENCY PLAN INITIATION

> **NOTE**
> For those events which are corrected, or the threat to the level of safety of the plant has ended prior to completion of classification/notification processes, it is permissible to classify the event and terminate the event with the initial emergency notification message. In this circumstance, termination does NOT require consultation with off-site authorities.

3.2.1 Classify the Event

1. **Assume** the role of Emergency Director.

> **NOTE**
> Whenever possible the Emergency Director should assign other qualified personnel to perform an independent assessment of event conditions and determine the highest emergency classification level to provide a peer check for the Emergency Director.

2. **Obtain** a copy of NMP-EP-110, Attachment 1, Checklist 1 - Classification Determination.

3. **Obtain** copies of the following from the appropriate site specific documents below.

Farley	Hatch	Vogtle
NMP-EP-110-GL01	NMP-EP-110-GL02	NMP-EP-110-GL03

4. Using Attachment 1, Checklist 1 - Classification Determination, **determine** the highest emergency classification level based on events which are in progress, considering past events, AND their impact on the current plant conditions.

5. The time annotated on Attachment 1, Checklist 1 - Classification Determination, line 5 constitutes the official emergency declaration time.

3.2.2 Emergency Plan Initiation

1. **Perform**NMP-EP-110, Attachment 2, Checklist 2 - Emergency Plan Initiation.

2. **Delegate** initial actions as specified in Attachment 2, Checklist 2 - Emergency Plan Initiation, utilizing, to delegate roles and responsibilities.

3. **Maintain** oversight of the event response.

4. The ED may operate from the Control Room or TSC at their discretion.

3.3 PERIODIC REVIEW OF THE CLASSIFICATION LEVEL

1. The ED SHALL periodically **review** current or projected plant conditions to determine if the emergency should be upgraded or terminated.

2. The Main Control Room (MCR) Staff shall periodically review plant conditions to make recommendations for EAL declarations.

3. The TSC Manager WILL **ensure** that the TSC Staff periodically reviews plant conditions, **determines** if the emergency should be upgraded based on current or projected status, AND **makes** recommendations to the ED.

4. The EOF Manager WILL **ensure** that the EOF Staff periodically reviews plant status AND offsite radiological conditions, **determines** if the emergency should be upgraded based on current status, current field surveys or projected releases, AND **makes** recommendations to the ED.

3.4 TERMINATING THE EMERGENCY CLASSIFICATION

NOTE

For those events which are corrected, or the threat to the level of safety of the plant has ended prior to completion of classification/notification processes, it is permissible to classify the event and terminate the event with the initial emergency notification message. In this circumstance, termination does NOT require consultation with off-site authorities.

1. SNC policy is that once an emergency classification is made, it can only be upgraded or terminated. It cannot be downgraded to a lower classification. Termination criteria are contained in Emergency Plan Implementing Procedures related to 'Recovery'. At termination, on an event specific basis, the site can either enter normal operating conditions or enter a recovery condition with a recovery organization established for turnover from the ERO.

2. For a NOUE, the ED may terminate the Emergency when plant conditions have stabilized AND the reason for the NOUE has been corrected. A NOUE can be terminated without coordination with offsite authorities. Following termination a written closeout for NOUE declarations shall be submitted to State and Local Agencies within 24 hours of termination.

3. For an Alert, Site Area Emergency, or General Emergency, the ED may terminate the Emergency **after** discussions with plant management, applicable members of the plant emergency response organization, the NRC, state and local officials as specified in site specific termination and recovery procedures. Following terminations a written closeout for Alert, Site Area and General Emergency declarations shall be submitted to State and Local Agencies within 8 hours of termination.

4. For those events which are corrected, or the threat to the level of safety of the plant has ended prior to completion of classification and notification processes, the condition may be reported using the guidance of NUREG 1022. See Section 3.5 below.

3.5 LATE OR MISSED CLASSIFICATIONS

1. IF an event has occurred that meets a threshold for declaration but no emergency has been declared at the time of discovery AND the basis for the emergency class no longer exists, THEN the condition should be reported using the guidance of NUREG 1022.

2. **Contact** the site Emergency Preparedness Supervisor or designee to contact and inform the State and local emergency response agencies.

3. **Use** applicable site specific NRC reporting procedures for NRC notification.

4.0 RECORDS

1. All data and information generated during the emergency event WILL be maintained by applicable emergency response personnel in each facility. This information WILL be utilized to generate a written close-out report upon termination of the emergency event. The report WILL be prepared as described in accordance with applicable procedures.

2. Records generated during actual emergencies WILL be maintained in accordance with applicable procedures.

QA record (X)	Non-QA record (X)	Record Generated	Retention Time	R-Type
X		FNP - EMERGENCY PREPAREDNESS DRILL OR TRAINING RECORDS	36 LP+99	TR0.001
X		FNP - EMERGENCY PLAN DRILL		K02.041
X		HNP – 70 SERIES DATA PKG TRAINING & EP	24ol +2	G16.072
	X	VEGP - COMPLETED DATA SHEET/CHECKLIST		NONQ43
	X	VEGP - ENN AND ENS NOTIFICATION FORMS		NONQ43
X		EMERGENCY PLANNING – SNC ACTUAL EVENTS (NUE, ALERT, SAE, GE)	60LP	EEP.008

5.0 REFERENCES

- Vogtle Emergency Plan
- HNP Emergency Plan
- FNP Emergency Plan

5.0 REFERENCES (continued)

- NMP-EP-111, "Emergency Notifications"
- NMP-EP-104, “Dose Assessment”
- NMP-EP-112, "Protective Action Recommendations"
- NUREG 1022 "Event Reporting Guidelines 10 CFR 50.72 and 50.73"
- INPO L2IER 1-39, "Lack of Timely Emergency Response Organization and Emergency Response Facility Activation"
- INPO L1IER 13-10, “Nuclear Accident at the Fukushima Daiichi Nuclear Power Station”

6.0 COMMITMENTS

6.1 VOGTLE

1985304559,	1985304561,	1985304579,	1985304580,	1985304581,
1985304588,	1985304589,	1985304591,	1985304593,	1985304607,
1985304608,	1985304614,	1985304615,	1985304616,	1985304620,
1985304624,	1985304625,	1985304664,	1985304671,	1985304693,
1985304761,	1985304810,	1986307894,	1986307896,	1986308706,
1986308709,	2002342760,	2002342880,	2002343236,	1985304590,
1985304560,	1985304555			

6.2 HATCH

SNC623394, SNC1743, SNC19090, SNC21006, SNC23431,

SNC23438, SNC23649, SNC24733, SNC26643, SNC24353

6.3 FARLEY

N/A

ATTACHMENT 1
Page 1 of 1

Checklist 1 - Classification Determination

NOTE
Key Parameters should be allowed to stabilize to accurately represent plant conditions prior to classifying an event

Initial Actions Completed by

1. **Determine** the appropriate Initiating Condition Matrix for classification of the event based on the current operating mode: ____________
   - ☐ HOT IC/EAL Matrix Evaluation Chart (**Go To** Step **2)** to evaluate the Barriers)
   - ☐ COLD IC/EAL Matrix Evaluation Chart (**Go To** Step **3**)
   - ☐ Both HOT & COLD IC/EAL Matrix Evaluation Chart apply (**Go To** Step **2)**
2. **Evaluate** the status of the fission product barrier using Figure 1, Fission Product Barrier Evaluation.

   a. **Select** the condition of each fission product barrier: ____________

	LOSS	POTENTIAL LOSS	INTACT
Fuel Cladding Integrity	☐	☐	☐
Reactor Coolant System	☐	☐	☐
Containment Integrity	☐	☐	☐

   b. **Determine** the highest applicable fission product barrier Initiating Condition (IC): ____________

   (select one) ☐ FG1 ☐ FS1 ☐ FA1 ☐ FU1 ☐ None

3. **Evaluate** <u>AND</u> **determine** the highest applicable IC/EAL using the Matrix Evaluation Chart(s) identified in step 1 <u>THEN</u> **Go To** step **4**.

   Hot IC#________ Unit___ and/or Cold IC# __________ Unit___ or ☐ None ____________

4. **Check** the <u>highest</u> emergency classification level identified from either step 2b or 3: ____________

Classification	**Based on IC#**	**Classification**	**Based on IC#**
☐ General	______________	☐ Alert	______________
☐ Site-Area	______________	☐ NOUE	______________
		☐ None	N/A

Remarks (**Identify** the specific EAL, as needed):____________________________

5. **Declare** the event by approving the Emergency Classification.

   ____________________________Date: ________/______/______Time: ________ ____________
   Emergency Director

6. **Obtain** Meteorological Data (not required prior to event declaration):

Wind Direction (from)_____ Wind Speed_____ Stability Class_____ Precipitation______ ____________

7. **Initiate Attachment 2, Checklist 2 - Emergency Plan Initiation**. ____________

ATTACHMENT 2
Page 1 of 6

Checklist 2 - Emergency Plan Initiation

Entry Assumptions:

- An emergency declaration has been made.
- Notification to state and local authorities will be required.

Initial Actions listed in order of priority. Take actions in parallel if resources are available. Subsequent actions should NOT be delayed pending the completion of prior actions. Utilize Figure 1 (Control Room) or Figure 2 (TSC) to delegate roles and responsibilities.	Completed by
1. **Update** personnel of assumption of the Emergency Director role, the classification of the event, most likely upgrade criteria, whether a release is in progress or not, AND **direct** that a log of the events be maintained.	____________
2. IF NOT already performed, THEN **direct** the merging of the Plant Public Address system to ensure notification of onsite personnel. (Hatch and Farley only)	____________
3. **Direct** the completion of the appropriate page announcement using NMP-EP-111, Emergency Notifications as soon as possible following either: • Changes in Emergency Classification level **OR** • Significant changes in Plant conditions	____________

NOTE

For security based events the ERO recall system should be activated using the security event activation protocol. This will direct off-site ERO personnel to alternative facilities and on-site ERO personnel to seek protective cover. This will minimize delays in overall site response by permitting ERO assembly without exposing responders to the danger of hostile action.

4. IF NOT already performed (i.e., at the ALERT or higher or Emergency Director Discretion), THEN **direct** the activation of the Emergency Response Organization. [1985304614] Initiation of the ERO recall system should be performed using the applicable instructions for the event. The envelope containing the activation instructions is located in the Control Room Emergency Director notebook/packet.	____________
5. **Direct** the communicator to establish communications and perform notifications in accordance with NMP-EP-111 [1985304621]	____________

Checklist 2 - Emergency Plan Initiation

Initial Actions listed in order of priority. Take actions in parallel if resources are available. Subsequent actions should NOT be delayed pending the completion of prior actions. Utilize Figure 1 (Control Room) or Figure 2 (TSC) to delegate roles and responsibilities.	Completed by

6. IF a General Emergency has been declared, THEN **develop** AND **approve** Protective Action Recommendations (PARs) in accordance with NMP-EP-112. PARs MUST be communicated to state and local authorities within 15 minutes of the existence of conditions requiring PARs OR the need to modify previous recommendations due to changing meteorological conditions. ____________

7. **Direct** the completion of the Emergency Notification Form in accordance with NMP-EP-111, Emergency Notifications. ____________

8. **Ensure** transmission of the EN form. Receipt of Emergency Notification must be verified within 15 minutes of emergency declaration, in accordance with NMP-EP-111, Emergency Notifications , Emergency Communicator Electronic Method, Emergency Communicator, Manual method, as appropriate. ____________

9. **Designate** an individual to complete Control Room assembly and accountability in accordance with site procedures. ____________

10. **Direct** an individual to complete NRC notifications immediately after notification of the appropriate state or local agencies but no later than one hour after the time the licensee declares one of the emergency classes IAW NMP-EP-111. Ensure an open line with the NRC is maintained, IF requested. ____________

11. **Direct** appropriate personnel to perform dose assessment per procedure. ____________

Farley	Hatch	Vogtle
FNP-0-EIP-9.1 **FNP-0-EIP-9.5** **NMP-EP-104**	**NMP-EP-104**	**NMP-EP-104**

12. **Direct** ERDS activation to transmit data to the NRC within one hour of the declaration of the emergency (ALERT and higher) using the appropriate site specific procedure below. ____________

Farley	Hatch	Vogtle
NMP-EP-111-001	73EP-EIP-063 (Att 1)	91111-C

Checklist 2 - Emergency Plan Initiation

Additional Actions – performed as conditions warrant. Take actions in parallel if resources are available.

Completed by

1. **Direct** follow-up Emergency Notification form completion as appropriate using NMP-EP-111. Follow-up notifications shall be performed following a significant change in plant conditions or at least every hour. ____________

2. **Direct** additional notifications required for personnel injury or fire as appropriate. ____________

Farley	Hatch	Vogtle
FNP-0-EIP-11.0	34AB-X43-001-1	92005-C
FNP-0-EIP-13.0	34AB-X43-001-2	70302-C
FNP-0-EIP-8.0	73EP-EIP-013-O	91309-C

3. **Direct** notification to the U.S. Army Explosive Ordinance Division (EOD) group as necessary. ____________

4. **Direct** notification to the Savannah River Operations Office, for Radiological Assistance Program (RAP) support as necessary. ____________

5. **Notify** the Duty Manager AND **remind** them to contact the Nuclear Duty Officer. ____________

6. IF there is a security event involved, THEN **ensure** appropriate notifications and actions of site specific AOP are performed. ____________

Farley	Hatch	Vogtle
FNP-0-AOP-49	34AB-Y22-004-0	18037-C
FNP-0-SP 37.0	34AB-Y22-005-0	

7. IF there is a security event involved, THEN **coordinate** all movement of personnel with security and the Incident Commander. ____________

Checklist 2 - Emergency Plan Initiation

Additional Actions – performed as conditions warrant. Take actions in parallel if resources are available. Completed by

8. **IF** there is a security event and site access is restricted by local law enforcement (LLE), THEN **perform** the following:

NOTE
Dedicated OPS Status, Security Status and Rad Status bridge lines are available to provide communication links for coordinating with the ICP and other Facilities.

- **Coordinate** with security and the Incident Commander via Incident Command Post (ICP) Liaisons in conjunction with the Alternative Facility to arrange for dispatch of needed personnel and resources to the site. ______________
- **Coordinate** with the Incident Commander and the Alternative Facility to **determine** on site AND off site staging areas for off-site resources using Emergency Management Guidelines (NMP-EP 400 series). ______________

9. **After** security reports that the immediate security threat has been eliminated, **determine** viability of activation of TSC and OSC AND **activate** as appropriate. Until an “all clear” has been given by security, in agreement with the Incident Commander, all movement of personnel must be coordinated with Security and the Incident Commander. ______________
10. When on-site conditions permit **coordinate** with Security and the Incident Commander to **perform** site accountability. ______________
11. **Determine** what should be done with a unit that is Not affected by the declared emergency. **Consider** the effect on the emergency unit, manpower utilization, plant and grid stability, and other relevant factors. ______________

NOTE
IF an ERO position is NOT filled by a LMS qualified individual, the vacant position may be filled by an individual with equivalent training and experience to perform the duties of the vacant position until an LMS qualified individual arrives at the discretion of the Emergency Director.

12. **Authorize** the filling of vacant ERO positions as needed to support activation of emergency response facilities and the response to the event. ______________

Checklist 2 - Emergency Plan Initiation

Additional Actions – performed as conditions warrant. Take actions in parallel if resources are available. Completed by

13**. Evaluate** Long term concerns.

- Within 8 hours, provide for full TSC and OSC reliefs. ____________
- Within 16 hours, provide for 24 hour TSC and OSC coverage. ____________
- Within 24 hrs, establish a 24 hour coverage schedule for events projected to last longer than 7 days. Include relief periods, for CR, TSC, and OSC personnel in this schedule.

14. IF an LOSP has occurred, THEN **evaluate** the event to ensure that an adequate supply of fuel oil is available for the Diesel Generators for 7 days. ____________

Farley	Hatch	Vogtle
REA 00-2337	TS 3.8.3	19100-C
FNP-0-SOP-42.0 Fig. 1		TS 3.8.3
TS 3.8.3		

15. **Verify** the Public Address (PA) system has been merged. ____________

16. IF the declared emergency involves extreme or extensive damage to the plant and/or plant components, THEN **refer** to the Emergency Management Guidelines (EMGs) (NMP-EP-400 series) and/or SAMGs. ____________

17. **Ensure** Health Physics confirms habitability of the rally points/assembly areas. IF unforeseen security/radiological/ weather conditions preclude dismissal through the normal exit point, **determine** and have announced over the Public Address (PA) system an alternate location for a rally point/assembly area and a site exit route. ____________

18. **Consider** ordering an early dismissal of non-involved personnel from the site IF the potential for degrading plant conditions or a threat to the safety of onsite personnel exist. ____________

19. **Ensure** Security initiates accountability for personnel within the Protected Area AND provides a status of the release of personnel (ALERT and higher emergency classifications), as appropriate. A security related emergency may delay the ordering of assembly and accountability in order to protect plant personnel from the security threat. The decision NOT to order assembly and accountability WILL be made by the Emergency Director. ____________

UATTACHMENT 2
Page 6 of 6

Checklist 2 - Emergency Plan Initiation

Additional Actions – performed as conditions warrant. Take actions in parallel if resources are available. Completed by

20. **Activate** emergency response teams to mitigate event consequences by contacting the TSC. IF the TSC is NOT activated, **contact** the Health Physics office and/or other support departments directly. ____________

- **Send** extra SOs to the OSC as conditions allow. SOs for shutdown activities WILL remain in the Control Room. ____________

21. **Maintain accountability of all teams dispatched.** ____________

22. IF necessary, **authorize** radiation exposures in excess of 10CFR20 limits. ____________

23. **Monitor** Area Radiation Monitor (ARM) readings and plant conditions for the location where teams have been dispatched by the control room. IF conditions change, THEN **withdraw** all personnel dispatched to that location. ____________

24. **Authorize** deviations from the Emergency Plan Implementing Procedures or Security Procedures. IF these deviations result in a departure from a regulatory commitment (Emergency Plan or Security Plan) or a technical specification under the provisions of 10CFR50.54(x), THEN, as a minimum, a licensed SRO MUST **approve** the action in accordance with 10CFR50.54(y) AND the NRC notified in accordance with 10CFR50.72 & 50.73. ____________

25. Diagnose plant conditions AND evaluate if a Severe Accident Management Guidelines entry is required ____________

26. IF terminating the emergency, THEN **ensure** the following actions have been performed: ____________

- **Ensure** verbal closeout to State and Local Authorities using Emergency Notification Network (ENN) or alternate communications. ____________
- **Ensure** verbal closeout to NRC using ENS or alternate communications. ____________
- **Ensure** all facilities AND applicable offsite agencies/ organizations are provided emergency termination information AND the status of recovery activities. ____________
- **Ensure** written closeout for NOUE declarations within 24 hours of termination to State and Local Agencies. ____________
- **Ensure** written closeout for Alert and higher declarations within 8 hours of termination to State and Local Agencies. ____________

Checklist 3 - Emergency Director Transfer of Responsibilities

1. **Review** with the ED:
   a. Logs ______
   b. Status boards (if applicable) ______
   c. Summary of events ______
   d. Plant status ______
   e. Equipment status ______
   f. Emergency classification ______
   g. Status of notifications of offsite authorities ______
   h. Protective and corrective actions ______
   i. Radiological releases ______
   j. Completed Checklist 2 items ______
   k. Status of facilities activation ______
   l. Emergency teams dispatched before activation of the ERF's. ______
   m. Any noted deficiencies ______
   n. Status of assembly and accountability, if initiated ______
   o. Outstanding orders ______
   p. Recovery plan of action, if known ______
2. **Review** facility readiness with facility managers. ______

NOTE

IF an ERO position is NOT filled by a LMS qualified individual, the vacant position may be filled by an individual with equivalent training and experience to perform the duties of the vacant position until an LMS qualified individual arrives at the discretion of the Emergency Director.

3. **Ensure** that log keeper maintains a log of ED actions AND records any transfer of responsibility. ______

Checklist 3 - Emergency Director Transfer of Responsibilities

4. **Formally assume** from the incumbent ED the position of ED using the following message format: ______

AT _____________ ON ________________ I AM ASSUMING THE EMERGENCY
(Time) (Date)

DIRECTOR POSITION AND HEREBY RELIEVE YOU OF ALL EMERGENCY DIRECTOR RESPONSIBILITIES.

Previous ED Signature __________________________________

Relieving ED Signature __________________________________

5. Following relief, **make** an announcement to the facility staff regarding the transfer of Emergency Director responsibility. ______

ATTACHMENT 4
Page 1 of 1

Checklist 4 - Deviating From Normal Work Processes

When a Site Area Emergency or General Emergency is declared deviations from normal processes may be authorized by the Emergency Director, or designee, in order to take action to restore core cooling, mitigate an off-site release in progress, or maintain spent fuel pool cooling. Deviations may also be authorized to protect fission product barriers or the health and safety of the public if deemed necessary. Consideration of deviations from the normal established processes should be considered prior to dispatch of work. The basis for needing to deviate from normal work practices is documented in the combined status log. NOTE: this process is only for deviations from normal work processes. Deviations from license requirement require use of the 10 CFR 50.54 (x) and (y) process.

Task being performed:____

Date/Time: ____

Approval to deviate: ____

✓ Indicates deviations from normal work processes	
Configuration Control	**Work Process**
☐ Clearance/Equipment Tagging ☐ Procedure (used as status control mechanism) ☐ Boundary Control ☐	☐ Work Order Process ☐ Manual Process ☐
Radiation Protection	**Industrial Safety**
☐ HP Job Coverage ☐ Dose Limits ☐ Turn Back Dose/Dose Rate ☐ Protective Clothing/Contamination Control ☐	☐ Working at heights ☐ Working over or near water ☐ Heat Stress ☐ PPE ☐ Electrical Safety ☐
List minimum requirements required for each deviation indicated:	
Post-job debrief (describe the as-found plant conditions, actions taken, as-left plant conditions):	

FIGURE 1
Page 1 of 1

Timeline for Implementation of Emergency Plan from the Control Room

Emergency Plan Implementation Roles, Responsibilities and Timeline

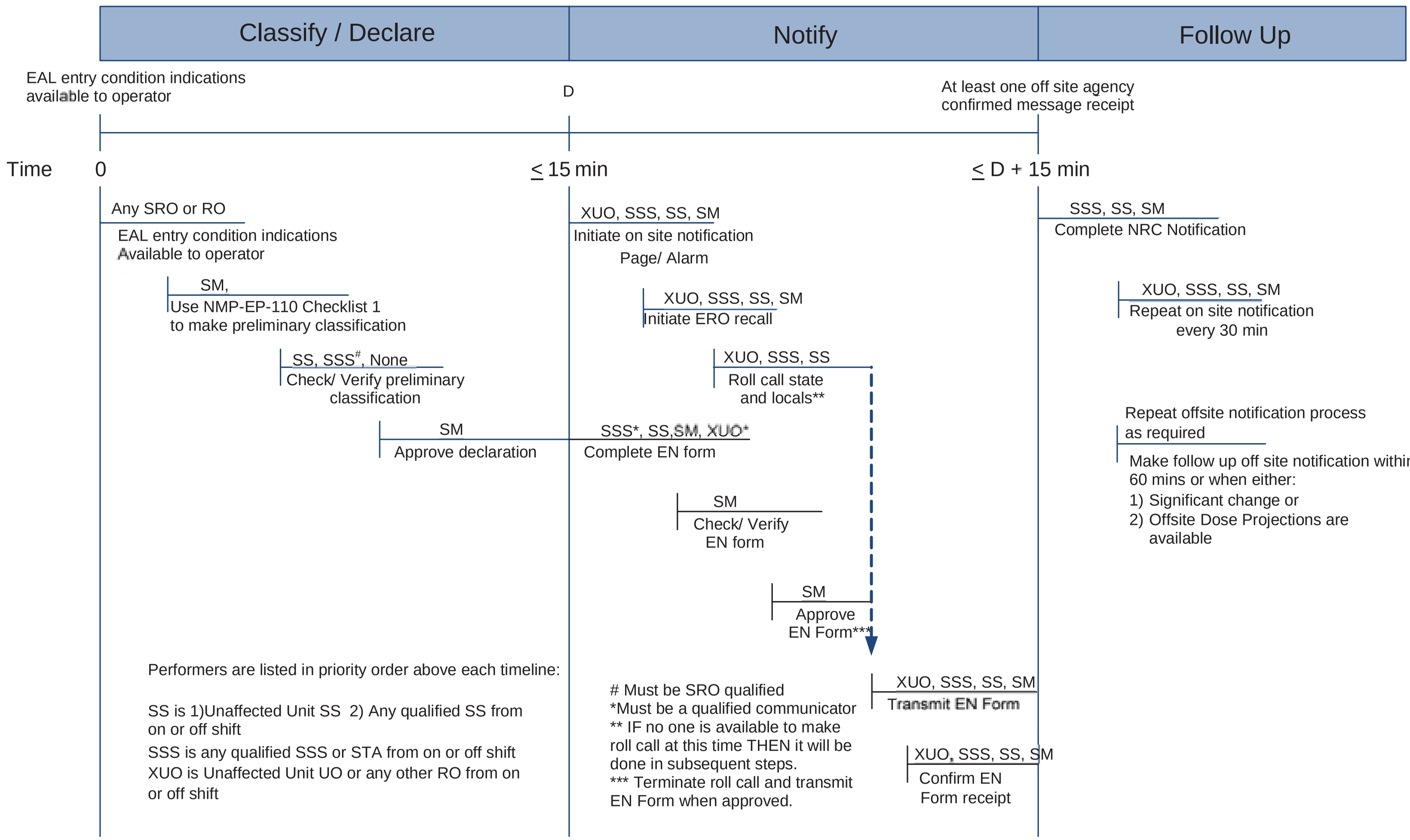
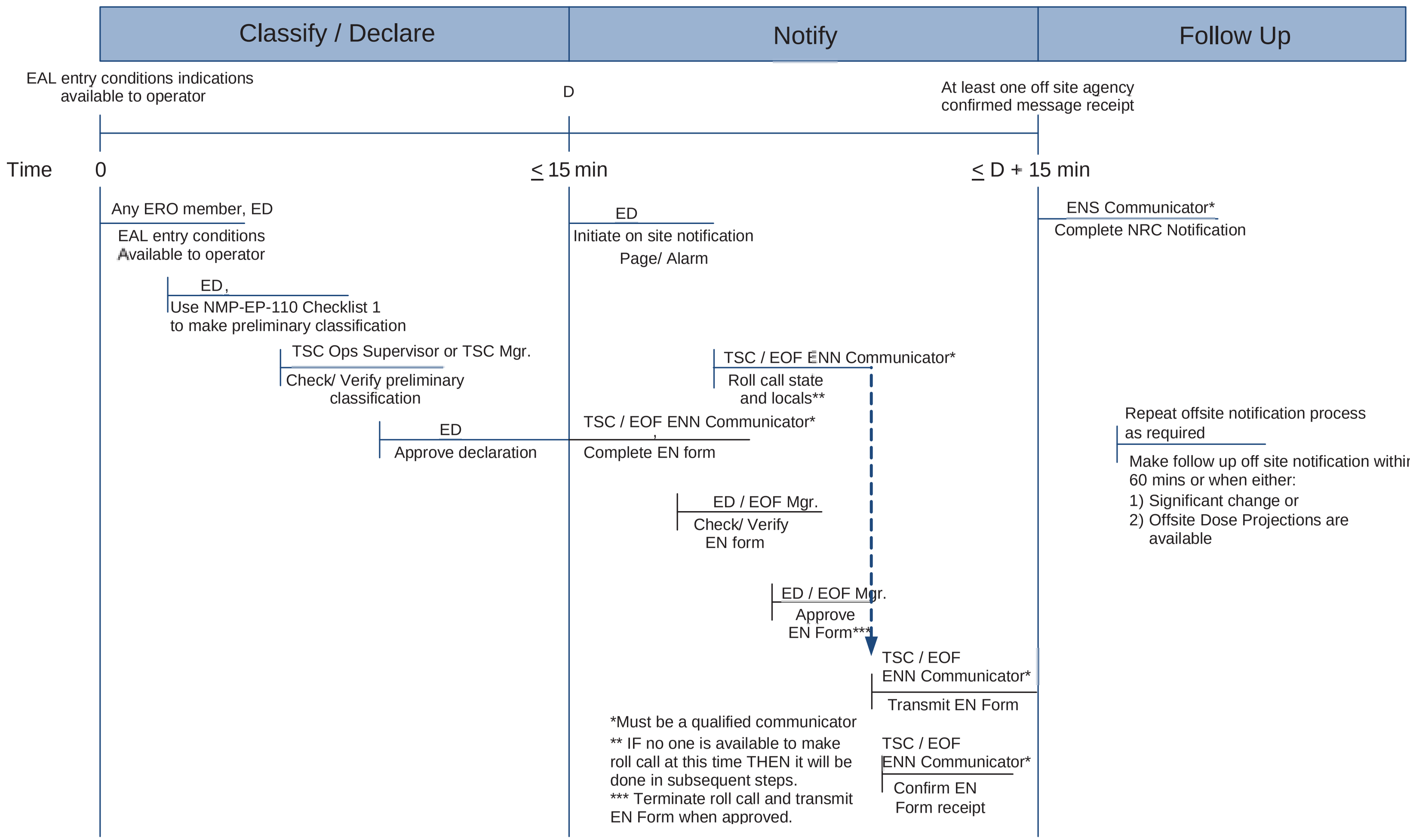


FIGURE 2
Page 1 of 1

Timeline for Implementation of Emergency Plan from the TSC and EOF

Emergency Plan Implementation Roles, Responsibilities, and Timeline



Southern Nuclear Company
Nuclear Management Guideline (NMG)

NMP-EP-110-GL01
FNP EALs - ICs, Threshold Values and Basis

Version 8.0
April 2015

Special Considerations:

Applicable to FNP

Approved by Lee Mansfield
(Guideline Owner/Preparer **Signature)

Approved by Dennis Drawbaugh
(Farley Review - **Signature)

Approved by Lee Mansfield
(Final Approval - **Signature)

** *Electronic Signature is located in DMS "Attributes" then "Acknowledgments".*

Guideline Version Description

Version Number	Version Description
4.0	• Added definitions for Notification Of Unusual Event (NOUE), ALERT, Site Area Emergency (SAE), and General Emergency to Definitions section. • EAL-HU2, Removed 2nd definition of "Contiguous" as it was listed twice. • EAL - HU2, Removed: 1) Note concerning disproving fires. 2) Sentence in Basis "The alarm can be determined to be spurious if it can be reset on the Pyro Panel." TE 574390
5.0	• Corrected Values for RG1 TV1 • Corrected Values for RS1 TV1 TE649999
6.0	• Corrected setpoint values for RG1, RS1, RA1 and RU1 • Corrected setpoint values for CG1 and CS2 • Corrected setpoint values for of Fuel Clad Barrier Loss #5, RCS Barrier Loss #4 and Containment Barrier Potential Loss #6 • Editorial change to basis section of SU2 (changed 1 hour report to four hour report) • Added clarification to bases section of HG1 • Deleted Figure 4 and all references to Figure 4 • Updated Figures 1, 2 and 3 to reflect new EAL values • Editorial changes to EAL Category Section Title Pages
7.0	This is an editorial revision to the EAL matrices pages to maintain consistency with the approved wording of the EAL Initiating Conditions and Threshold Values. These corrections consisted of: • Typographical errors • Punctuation differences • Formatting differences • Page number reference differences • Wording in the ICs and TVs missing or different from the matrices
8.0	• Added clarifying information to EAL Basis for HA3, HU3 • Added clarifying information to EAL Basis for Containment Barrier Loss • Corrected spelling and page number references • Unit conversions for RA1, RA3, RU1 to match instrument units • Added "or Threat" to HU4 to align with emergency plan wording

Table of Contents

INTRODUCTION

FNP must respond to a formal set of threshold conditions that require plant personnel to take specific actions with regard to notifying state and local governments and the public when certain off-normal indicators or events are recognized. Emergency classes are defined in 10 CFR 50. The levels of response and conditions leading to those responses are defined in a joint NRC/FEMA guidelines contained in Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” October 1980. The nuclear industry has developed NEI 99-01, Revision 4, a set of generic EAL guidelines and supporting basis, to use to develop the site specific EALs, their Threshold Values and Basis. This generic guidance is intended to clearly define conditions that represent increasing risk to the public and can give consistent classifications when applied at different sites. It is a NRC endorsed acceptable alternative to the guidance in NUREG-0654.

This information is presented by Recognition Category:

- R - Abnormal Rad Levels/Radiological Effluent
- C - Cold Shutdown./ Refueling System Malfunction
- E - Events Related to Independent Spent Fuel Storage Installations (ISFSI)
- F - Fission Product Barrier Degradation
- H - Hazards and Other Conditions Affecting Plant Safety
- S - System Malfunction

In this guideline each of the EALs in Recognition Categories R, C, E, F, H, and S are structured in the following way:

- Recognition Category - As listed above.
- Emergency Class - NUE, Alert, Site Area Emergency or General Emergency.
- Initiating Condition - Symptom or Event-Based, Generic Identification and Title.
- Operating Mode Applicability - Power Operation, Startup, Hot Shutdown, Cold Shutdown, Refueling, Defueled, or All.
- Threshold Value(s) corresponding to the IC.
- Basis information for plant-specific readings and factors that may relate to changing the generic IC or EAL to a different emergency class, such as for Loss of All AC Power.

For Recognition Category F, the EAL information is presented in a matrix format. The method was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. For category F, the EALs are arranged by fission product barrier. Classifications are based on various combinations of barrier challenges.

Emergency classes were established by the Nuclear Regulatory Commission (NRC), for grouping off-normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time-sensitive onsite and off-site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency.

With the emergency classes defined, the Initiating Conditions and Threshold Values that must be met for each EAL to be placed under the emergency class can be determined. There are two basic approaches to determining these EALs. EALs and emergency class boundaries coincide for those continuously measurable, instrumented ICs, such as radioactivity, core temperature, coolant levels, etc. For these ICs, the EAL will be the threshold reading that most closely corresponds to the emergency class description using the best available information. For discrete (discontinuous) events, the approach will have to be somewhat different. Typically, in this category are internal and external hazards such as fire or earthquake. The purpose for including hazards in EALs is to assure that station personnel and offsite emergency response organizations are prepared to deal with consequential damage these hazards may cause. If, indeed, hazards have caused damage to safety functions or fission product barriers, this should be confirmed by symptoms or by observation of such failures. Of course, security events must reflect potential for increasing security threat levels.

The EALs and ICs can be grouped in one of several schemes. The classification scheme incorporates symptom-based, event-based, and barrier-based EALs and ICs.

Symptom-based EALs and ICs refers to those indicators that are measurable over some continuous spectrum, such as core temperature, coolant levels, containment pressure, etc. The level of seriousness indicated by these symptoms depends on the degree to which they have exceeded technical specifications, the other symptoms or events that are occurring contemporaneously, and the capability of the licensed operators to gain control and bring the indicator back to safe levels.

Event-based EALs and ICs refer to occurrences with potential safety significance, such as the failure of a high-pressure safety injection pump, a safety relief valve failure, or a loss of electric power to some part of the plant. The range of seriousness of these “events” is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc. Several categories of emergencies have no instrumentation to indicate a developing problem, or the event may be identified before any other indications are recognized. For emergencies related to the reactor system and safety systems, the ICs shift to an event based scheme as the plant mode moves toward cold shutdown and refueling modes. For non-radiological events, such as FIRE, floods, wind loads, etc. event-based ICs are the norm.

Barrier-based EALs and ICs refer to the level of challenge to principal barriers used to assure containment of radioactive materials contained within a nuclear power plant. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and primary containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers concurrently under challenge. The fission product barrier matrix is a hybrid approach that recognizes that some events may represent a challenge to more than one barrier, and that the containment barrier is weighted less than the reactor coolant system pressure boundary and the fuel clad barriers.

The most common bases for establishing Threshold Values are the technical specifications and setpoints for each plant that have been developed in the design basis calculations and the Final Safety Analysis Report (FSAR). For those conditions that are easily measurable and instrumented, the boundary is likely to be the EAL (observable by plant staff, instrument reading, alarm setpoint, etc.) that indicates entry into a particular emergency class. That radiation level also may be the setpoint that closes the main steam isolation valves (MSIV) and initiates the reactor scram. This same radiation level threshold, depending on plant-specific parameters, also may be the appropriate EAL for a direct entry into an emergency class.

PURPOSE/SCOPE

The purpose of this guideline is to provide the EAL classification Initiating Conditions (IC) matrices utilized in NMP-EP-110, Emergency Classification and Initial Actions, for the Joseph M. Farley Nuclear Power Plant. Theses matrices are utilized in the classification of off-normal events into one of four emergency classification levels.

This guideline provides the IC, Threshold Values (TV), and Basis for each Emergency Action Levels (EALs) grouped by their Recognition Categories. If after reviewing the classification IC matrices, the classification of an event or the determination if a Threshold Value is met or exceeded is unclear, the basis information provides additional clarification for each IC.

There are three considerations related to emergency classes. These are:

(1) The potential impact on radiological safety, either as now known or as can be reasonably projected;

(2) How far the plant is beyond its predefined design, safety, and operating envelopes; and

(3) Whether or not conditions that threaten health are expected to be confined to within the site boundary.

Although the majority of the EALs provide very specific thresholds, the Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the EAL threshold is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded. While this is particularly prudent at the higher emergency classes (as the early classification may provide for more effective implementation of protective measures), it is nonetheless applicable to all emergency classes. At all times, when conditions present themselves that are not explicitly provided in the EAL scheme, the

Emergency Director has discretion to declare an event based on his knowledge of the emergency classes and judgment of the situation or condition. Specific EALs (HU5, HA6, HS3, & HG2) are provided within the scheme to allow these discretionary classifications.

The classification procedure is written to classify events based on meeting the IC and a TV for an EAL considering each Unit independently. The IC Matrices are human factored to read from top to bottom (i.e., General Emergency down to Notification of Unusual Event) within a category or subcategory to eliminate the higher classifications before reaching a lower classification. This arrangement lessens the possibility of under-classifying a condition. During events, the ICs and TVs are monitored and if conditions meet another higher EAL, that higher emergency classification is declared and appropriate notifications made. The Notifications are made on a site basis, not a Unit (1 or 2) basis. If both Units are in concurrent classifications, the highest classification must be used for the notification and the other Unit events noted on the SNC Emergency Notification form.

The SNC policy is that once an emergency classification is made, it cannot be downgraded to a lower classification. Termination criteria contained in procedure NMP-EP-110, Emergency Classification and Initial Actions shall be completed for an event to be terminated. At termination, on an event specific basis, the site can either enter normal operating conditions or enter a recovery condition with a recovery organization established for turnover from the ERO.

APPLICABILITY

This procedure applies to emergency classification determinations and associated initial responses. This procedure will be utilized for actual emergencies, emergency drills/exercises, or training as required.

REFERENCES

Farley Nuclear Plant Emergency Plan

NUREG 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73

Farley Unit 1 & 2 Technical Specifications (TS)

SNC Letter Dated December 30, 2005; NL-05-2236, Southern Nuclear Operating Company Emergency Plans Transition to NEI 99-01 Emergency Action Level Scheme

SNC Letter Dated October 13, 2006; NL-06-2177, Transition to NEI 99-01 Emergency Action Levels Response to Request for Additional Information

SNC Letter Dated April 1, 2007; NL-07-0522, Transition to NEI 99-01 Emergency Action Levels - Response to Request for Additional Information;

NRC Letter Dated April 30, 2007; LC # 14580 - Emergency Action Level Revisions for Southern Nuclear Operating Company, Inc. (Includes Safety Evaluation By The Office of Nuclear Reactor Regulation Related to Proposed Revisions to the Emergency Action Levels for The Joseph M Farley Nuclear Plant, Unit Nos. 1 and 2 (FNP).

DEFINITIONS

ALERT**:** Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of Hostile Action. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

CIVIL DISTURBANCE: An organized demonstration by an individual or group of unexpected, unidentified, or unauthorized people within the Owner Controlled Area (OCA) which is used to promote a political or social issue or belief.

COMMITTED DOSE EQUIVALENT (CDE): Means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

COMMITTED EFFECTIVE DOSE EQUIVALENT (CEDE): The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs of tissues.

CONFINEMENT BOUNDARY: The barrier(s) between areas containing radioactive substances and the environment.

CONTAINMENT CLOSURE: Per FNP-1-STP-18.4, “Containment Integrity Verification and Closure”.

CONTIGUOUS: Being in actual contact; touching along a boundary or at a point.

CREDIBLE THREAT: A threat is considered credible when (1) physical evidence supporting the threat exists, or (2) information independent from the actual threat message exists that support the threat, or (3) a specific group or organization claims responsibility for the threat, or (4) a message (written or verbal) is received that contains specific information about plant locations, systems or device description an average person would most likely not know. The determination of credibility should be made by the Shift Manager with input from the Shift Captain or their designated representatives.

DEEP DOSE EQUIVALENT (DDE): Which applies to external whole body exposure, is the dose equivalent at a tissue depth of 1 cm.

EXPLOSION: Rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

GENERAL EMERGENCY**:** Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity or Hostile Action that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

HOSTILE ACTION: An act toward an NPP or its personnel that includes the use of violent force to destroy equipment, takes hostages, and /or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non- terrorism-based Threshold Values should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

IMMINENT: Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

NORMAL PLANT OPERATIONS: Activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONS.

NOTIFICATION OF UNUSUAL EVENT (NOUE) - Events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

OWNER CONTROLED AREA: The area which normally encompasses all controlled areas within the FNP site boundary but outside security protected area fence.

PROTECTED AREA: The area which normally encompasses all controlled areas within the security protected area fence.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

SIGNIFICANT TRANSIENT: An UNPLANNED event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations greater than 10%.

SITE AREA EMERGENCY: Events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or Hostile Actions that result in intentional damage or malicious acts; 1) toward site personnel or equipment that could lead to the likely failure of or; 2) that prevent effective access to, equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

STRIKE ACTION: A work stoppage within the PROTECTED AREA by a body of workers to enforce compliance with demands. The STRIKE ACTION must threaten to interrupt NORMAL PLANT OPERATIONs.

TOTAL EFFECTIVE DOSE EQUIVALENT (TEDE): The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

UNPLANNED: A parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: An indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

VISIBLE DAMAGE: Damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

VITAL AREA: Any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

ACRONYMS

Acronyms, when used within an IC, Threshold Value or the basis, are defined within the body of the document. With this method, the user has a direct reference to the acronym's usage without having to go to this section to determine its particular contextual meaning.

RECORDS

No Records Generated

Farley Nuclear Plant

EMERGENCY ACTION LEVELS

INITIATING CONDITIONS, THRESHOLD VALUES, AND BASIS

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this Attachment is to provide additional guidance and clarification to the EAL classification Initiating Conditions (IC) matrices in Attachments 1-3 of this procedure. They are utilized in the classification of off-normal events into one of four emergency classification levels.

This attachment provides the IC, Threshold Values (TV), and Basis for each Emergency Action Levels (EALs) grouped by their Recognition Categories. If after reviewing the classification IC matrices, the classification of an event or the determination if a Threshold Value is met or exceeded is unclear, the basis information provides additional clarification for each IC.

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The classification procedure is written to classify events based on meeting the IC and a TV for an EAL considering each Unit independently. The IC Matrices are human factored to read from top to bottom (i.e., General Emergency down to Notification of Unusual Event) within a category or subcategory to eliminate the higher classifications before reaching a lower classification. This arrangement lessens the possibility of under-classifying a condition. During events, the ICs and TVs are monitored and if conditions meet another higher EAL, that higher emergency classification is declared and appropriate notifications made. The Notifications are made on a site basis, not a Unit (1 or 2) basis. If both Units are in concurrent classifications, the highest classification must be used for the notification and the other Unit events noted on the SNC Emergency Notification form.

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2.0 Background

FNP must respond to a formal set of threshold conditions that require plant personnel to take specific actions with regard to notifying state and local governments and the public when certain off-normal indicators or events are recognized. Emergency classes are defined in 10 CFR 50. The levels of response and conditions leading to those responses are defined in a joint NRC/FEMA guidelines contained in Appendix 1 of NUREG-0654/FEMA-REP-1, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," October 1980. The nuclear industry has developed NEI 99-01, Revision 4, a set of generic EAL guidelines and supporting basis, to use to develop the site specific EALs, their Threshold Values and Basis. This generic guidance is intended to clearly define conditions that represent increasing risk to the public and can give consistent classifications when applied at different sites. It is a NRC endorsed acceptable alternative to the guidance in NUREG-0654.

This information is presented by Recognition Category:

- **R** - Abnormal **R**ad Levels/Radiological Effluent
- **C** - **C**old Shutdown./ Refueling System Malfunction
- **E** - **E**vents Related to Independent Spent Fuel Storage Installations (ISFSI)
- **F** - **F**ission Product Barrier Degradation
- **H** - **H**azards and Other Conditions Affecting Plant Safety
- **S** - **S**ystem Malfunction

In this Attachment each of the EALs in Recognition Categories **R**, **C**, **D**, **E**, **H**, and **S** are structured in the following way:

- Recognition Category - As listed above.
- Emergency Class - NUE, Alert, Site Area Emergency or General Emergency.
- Initiating Condition - Symptom or Event-Based, Generic Identification and Title.
- Operating Mode Applicability - Power Operation, Startup, Hot Shutdown, Cold Shutdown, Refueling, Defueled, or All.
- Threshold Value(s) corresponding to the IC.
- Basis information for plant-specific readings and factors that may relate to changing the generic IC or EAL to a different emergency class, such as for Loss of All AC Power.

For Recognition Category **F**, the EAL information is presented in a matrix format. The method was chosen to clearly show the synergism among the EALs and to support more accurate dynamic assessments. For category **F**, the EALs are arranged by fission product barrier. Classifications are based on various combinations of barrier challenges.

2.0 Background (cont.)

Emergency classes were established by the Nuclear Regulatory Commission (NRC), for grouping off-normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time-sensitive onsite and off-site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency.

With the emergency classes defined, the Initiating Conditions and Threshold Values that must be met for each EAL to be placed under the emergency class can be determined. There are two basic approaches to determining these EALs. EALs and emergency class boundaries coincide for those continuously measurable, instrumented ICs, such as radioactivity, core temperature, coolant levels, etc. For these ICs, the EAL will be the threshold reading that most closely corresponds to the emergency class description using the best available information. For discrete (discontinuous) events, the approach will have to be somewhat different. Typically, in this category are internal and external hazards such as fire or earthquake. The purpose for including hazards in EALs is to assure that station personnel and offsite emergency response organizations are prepared to deal with consequential damage these hazards may cause. If, indeed, hazards have caused damage to safety functions or fission product barriers, this should be confirmed by symptoms or by observation of such failures. Of course, security events must reflect potential for increasing security threat levels.

The EALs and ICs can be grouped in one of several schemes. The classification scheme incorporates <u>symptom-based</u>, <u>event-based</u>, and <u>barrier-based</u> EALs and ICs.

<u>Symptom-based EALs and ICs</u> refers to those indicators that are measurable over some continuous spectrum, such as core temperature, coolant levels, containment pressure, etc. The level of seriousness indicated by these symptoms depends on the degree to which they have exceeded technical specifications, the other symptoms or events that are occurring contemporaneously, and the capability of the licensed operators to gain control and bring the indicator back to safe levels.

<u>Event-based EALs and ICs</u> refer to occurrences with potential safety significance, such as the failure of a high-pressure safety injection pump, a safety relief valve failure, or a loss of electric power to some part of the plant. The range of seriousness of these “events” is dependent on the location, number of contemporaneous events, remaining plant safety margin, etc. Several categories of emergencies have no instrumentation to indicate a developing problem, or the event may be identified before any other indications are recognized. For emergencies related to the reactor system and safety systems, the ICs shift to an event based scheme as the plant mode moves toward cold shutdown and refueling modes. For non-radiological events, such as FIRE, floods, wind loads, etc. event-based ICs are the norm.

2.0 Background (cont.)

Barrier-based EALs and ICs refer to the level of challenge to principal barriers used to assure containment of radioactive materials contained within a nuclear power plant. For radioactive materials that are contained within the reactor core, these barriers are: fuel cladding, reactor coolant system pressure boundary, and primary containment. The level of challenge to these barriers encompasses the extent of damage (loss or potential loss) and the number of barriers concurrently under challenge. The fission product barrier matrix is a hybrid approach that recognizes that some events may represent a challenge to more than one barrier, and that the containment barrier is weighted less than the reactor coolant system pressure boundary and the fuel clad barriers.

The most common bases for establishing Threshold Values are the technical specifications and setpoints for each plant that have been developed in the design basis calculations and the Final Safety Analysis Report (FSAR). For those conditions that are easily measurable and instrumented, the boundary is likely to be the EAL (observable by plant staff, instrument reading, alarm setpoint, etc.) that indicates entry into a particular emergency class. That radiation level also may be the setpoint that closes the main steam isolation valves (MSIV) and initiates the reactor scram. This same radiation level threshold, depending on plant-specific parameters, also may be the appropriate EAL for a direct entry into an emergency class.

3.0 MODE DESCRIPTION

Farley Units 1 and 2 Technical Specifications Table 1.1-1 provides the following mode definitions:

Mode	Title	Average RCS Temperature (°F)
1	Power Operation	NA
2	Startup	NA
3	Hot Standby	Greater than 350
4	Hot Shutdown (a)	350 > Tavg > 200
5	Cold Shutdown(a)	$\leq$ 200
6	Refueling	NA

(a) All reactor vessel head closure bolts fully tensioned.
(b) One or more reactor vessel head closure bolts less than fully tensioned

These modes are used throughout the Farley EALs with no modifications from NEI 99-01. For the condition when a unit is defueled, the Initiating Conditions designated as Mode Condition “ALL” or “Defueled” are applicable.

4.0 FNP EALS - INITIATING CONDITIONS, THRESHOLD VALUES AND BASIS

4.1 Category R - Abnormal Radiological

RG1

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1,000 mR TEDE OR 5,000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3)

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

NOTE: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following radiation monitors that exceeds OR expected to exceed the reading shown for 15 minutes OR longer:

Steam jet Air Ejector RE-15C	130 μCi/cc (130 R/hr)
Plant Vent Stack RE-29B (NG)	0.9 μCi/cc
Steam Generator Relief RE-60A,B,C	0.5 μCi/cc (0.5 R/hr)
TDAFW Steam Exhaust RE-60D	11 μCi/cc (11 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 1,000 mR TEDE **OR** 5,000 mR thyroid CDE at **OR** beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 1,000 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 5,000 mR for one hour of inhalation, at **OR** beyond site boundary.

Basis:

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed the EPA Protective Action Guides (PAGs). Public protective actions will be necessary. Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public and likely involve fuel damage. While these failures are addressed by other ICs, this IC provides appropriate diversity and addresses events which may not be able to be classified on the basis of plant status alone. It is important to note that, for the more severe accidents, the release may be unmonitored or there may be large uncertainties associated with the source term and/or meteorology.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The monitor reading Threshold Values are determined using a dose assessment method that back calculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used are the same as those used for determining the monitor reading Threshold Values in ICs RU1 and RA1. This protocol will maintain intervals between the Threshold Values for the four classifications. Since doses are generally not monitored in real-time, a release duration of one hour is assumed, and that the Threshold Values are based on a site boundary (or beyond) dose of 1000 mR/hour whole body or 5000 mR/hour thyroid, whichever is more limiting.

Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold Values are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made the dose assessment results override the monitor reading Threshold Values. Classification should not be delayed pending the results of these dose assessments.

RS1

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE **OR** 500 mR Thyroid CDE for the Actual or Projected Duration of the Release.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3)

NOTE: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

NOTE: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following radiation monitors that exceeds **OR** is expected to exceed the reading shown for 15 minutes **OR** longer:

Steam jet Air Ejector RE-15C	13 μCi/cc (13 R/hr)
Plant Vent Stack RE-29B (NG)	0.09 μCi/cc
Steam Generator Relief RE-60A,B,C	0.05 μCi/cc (0.05 R/hr)
TDAFW Steam Exhaust RE-60D	1.1 μCi/cc (1.1 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE **OR** 500 mR thyroid CDE at **OR** beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at **OR** beyond the site boundary.

Basis:

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses radioactivity releases that result in doses at or beyond the site boundary that exceed a small fraction of the EPA Protective Action Guides (PAGs). Releases of this magnitude are associated with the failure of plant systems needed for the protection of the public.

The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

The monitor reading Threshold Values are determined using a dose assessment method that back calculates from the dose values specified in the IC. The meteorology and source term (noble gases, particulates, and halogens) used is the same as those used for determining the monitor reading Threshold Values in ICs RU1 and RA1. This protocol maintains intervals between the Threshold Values for the four classifications. Since doses are generally not monitored in real-time, a release duration of one hour is assumed, and that the Threshold Values be based on a site boundary (or beyond) dose of 100 mR/hour whole body or 500 mR/hour thyroid, whichever is more limiting.

The release rates which result in site boundary doses of 100 mR TEDE are in excess of the range of the monitors listed in RU1 and RA1.

Since dose assessment is based on actual meteorology, whereas the monitor reading Threshold Values are not, the results from these assessments may indicate that the classification is not warranted, or may indicate that a higher classification is warranted. For this reason, emergency implementing procedures should call for the timely performance of dose assessments using actual meteorology and release information. If the results of these dose assessments are available when the classification is made (*e.g.*, initiated at a lower classification level), the dose assessment results override the monitor reading Threshold Values. Classification should not be delayed pending the results of these dose assessments.

RA1

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3)

NOTE:	**The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.**

1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes **OR** longer.

Monitor		200 X Setpoint Value
Liquid Radwaste Effluent Line (RE-18)		200 x release permit setpoint (planned release) Greater than **OR** equal to 1×10^{6} cpm (no planned release)
Steam Generator Blowdown Effluent Line (RE-23B)		2.8×10^{5} cpm
Steam Jet Air Ejector RE-15A		3.5×10^{4} cpm
Plant Vent Gas	RE-14	200 x release permit setpoint (planned release) Greater than **OR** equal to 1×10^{6} cpm (no planned release)
	RE-22	4.0×10^{4} cpm
	RE-29B (NG)	8.9×10^{-2} µCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes **OR** longer:

Main Steam Atmos Relief (R60A,B,C)	1.4×10^{1} R/hr
TDAFW Exhaust (R60D)	1.4×10^{1} R/hr

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations **OR** release rates in excess of 200 times Technical Specification 5.5.4.b as confirmed by ODCM , with a release duration of 15 minutes **OR** longer.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses a potential or actual decline in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. **The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.** Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 15 minutes.

Threshold Value #1 addresses radioactivity releases that for whatever reason cause effluent radiation monitor readings that exceed two hundred times the alarm setpoint established by the radioactivity discharge permit. This alarm setpoint may be associated with a planned batch release, or a continuous release path.

Threshold Value #2 is similar to Threshold Value #1, but is intended to address effluent or accident radiation monitors on non-routine release pathways for which a discharge permit would not normally be prepared.

Threshold Value #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

Threshold Values #1 and #2 directly correlate with the IC since annual average meteorology is required to be used in showing compliance with the RECP and is used in calculating the alarm setpoints.

Due to the uncertainty associated with meteorology, emergency implementing procedures should call for the timely performance of dose assessments using actual (real-time) meteorology in the event of a gaseous radioactivity release of this magnitude. The results of these assessments should be compared to the ICs RS1 and RG1 to determine if the event classification should be escalated.

RA2

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Damage to Irradiated Fuel **OR** Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2)

1. UNPLANNED VALID alarm on any of the following radiation monitors:

Drumming Station RE-0008
Containment Purge Ventilation Monitor RE-24A **OR** B
Spent Fuel Pool Ventilation Monitor RE-25A **OR** B
Spent Fuel Pool Area Radiation Monitor RE-5

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:

Report of personnel during fuel assembly movements.	
Spent Fuel Pool Storage	Less than El 129
Transfer Canal	Less than El 116
Reactor Core Elevation	Less than El 118

Basis:

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator’s operability, the condition’s existence, or the report’s accuracy is removed. Implicit in this definition is the need for timely assessment.

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC addresses specific events that have resulted, or may result, in unexpected rises in radiation dose rates within plant buildings, and may be a precursor to a radioactivity release to the environment. These events represent a loss of control over radioactive material and represent a degradation in the level of safety of the plant.

Threshold Value #1 addresses radiation monitor indications of fuel uncovery and/or fuel damage. Raised readings on ventilation monitors may be indication of a radioactivity release from the fuel, confirming that damage has occurred. Raised background at the monitor due to water level lowering may mask raised ventilation exhaust airborne activity and needs to be considered. Application of these Initiating Conditions requires understanding of the actual radiological conditions present in the vicinity of the monitor.

In Threshold Value #2, indications include water level and personnel reports. Visual observation will be the primary indicator for spent fuel pool and fuel movement activities. Personnel report during fuel assembly movements is included to ensure that reports of actual or potential fuel uncovery is classified. If available, video cameras may allow remote observation. Depending on available level indication, the declaration threshold may need to be based on indications of water makeup rate or lowering in refueling water storage tank level.

RA3

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:

Control Room radiation monitor	RE-1A

2. UNPLANNED VALID radiation readings greater than 1000 mR/hr (1 R/hr) values in the following areas requiring infrequent access to maintain plant safety functions.

RadioChemistry Lab Area RE-3	Electrical And Piping Penetration Rooms
Charging Pump Room Area RE-4	VCT Valve Room
Sample Room Area RE-6	Seal Water HX Room
Lower Equipment Room	CCW HX Room
Main Steam Valve Room	Turbine Building Air Compressor Area
4160 Volt ESF Bus Rooms	DC Switchgear Rooms
Control Rod Drive Room	Valve Box 1, 2, 3 and 4
Diesel Building	Service Water Intake Structure
Hot shutdown Panels	

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses raised radiation levels that impede necessary access to operating stations, or other areas containing equipment that must be operated manually or that requires local monitoring, in order to maintain safe operation or perform a safe shutdown. It is this impaired ability to operate the plant that results in the actual or potential substantial degradation of the level of safety of the plant. The cause and/or magnitude of the rise in radiation levels is not a concern of this IC. The Emergency Director must consider the source or cause of the raised radiation levels and determine if any other IC may be involved.

This IC is not meant to apply to anticipated temporary rises due to planned events.

The area requiring continuous occupancy is the control room and the central alarm station. The Central Alarm Station is in the Control Room envelope. The value of 15mR/hr is derived from the GDC 19 value of 5 Rem in 30 days with adjustment for expected occupancy times.

For areas requiring infrequent access, the 1000 mR/hr (1 R/hr) (Locked High Rad Area) is based on radiation levels which result in exposure control measures intended to maintain doses within normal occupational exposure guidelines and limits (i.e., 10 CFR 20), and in doing so, will impede necessary access. As used here, *impede,* includes hindering or interfering provided that the interference or delay is sufficient to significantly threaten the safe operation of the plant.

RU1

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3)

NOTE: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes **OR** longer.

Monitor	2 X Setpoint Value
Liquid Radwaste Effluent Line RE-18	2 x release permit setpoint (planned release) 1.6×10^{4} cpm (no planned release)
Steam Generator Blowdown Effluent Line RE-23B	2.8×10^{3} cpm
Steam Jet Air Ejector RE-15	3.5×10^{2}cpm
Plant Vent Gas RE-14 RE-22 RE-29B (NG)	2 x release permit setpoint (planned release) 3.2×10^{4} cpm (no planned release) 4.0×10^{2} cpm 8.9×10^{-4} μCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes **OR** longer:

Main Steam Atmos Relief R60A,B,C	1.4×10^{-1} R/hr
TDAFW Exhaust R60D	1.4×10^{-1} R/hr

3. Confirmed sample analyses for gaseous **OR** liquid releases indicates concentrations **OR** release rates, with a release duration of 60 minutes **OR** longer, in excess of two times Technical Specification 5.5.4.b, as confirmed by the ODCM.

Basis:

UNPLANNED, as used in this context, includes any release for which a radioactivity discharge permit was not prepared, or a release that exceeds the conditions (e.g., minimum dilution flow, maximum discharge flow, alarm setpoints, etc.) on the applicable permit. The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. Also, if an ongoing release is detected and the starting time for that release is unknown, the Emergency Director should, in the absence of data to the contrary, assume that the release has exceeded 60 minutes.

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses a potential or actual decline in the level of safety of the plant as indicated by a radiological release that exceeds regulatory commitments for an extended period of time. Nuclear power plants incorporate features intended to control the release of radioactive effluents to the environment. Further, there are administrative controls established to prevent unintentional releases, or control and monitor intentional releases. These controls are located in the Offsite Dose Calculation Manual (ODCM), Ref 2. The occurrence of extended, uncontrolled radioactive releases to the environment is indicative of a degradation in these features and/or controls.

Threshold Value #1 addresses radioactivity releases, that for whatever reason, cause effluent radiation monitor readings to exceed two times the Technical Specification limit and releases are not terminated within 60 minutes. This alarm setpoint may be associated with a planned batch release, or a continuous release path. In either case, the setpoint is established by the ODCM to warn of a release that is not in compliance with the TS 5.5.4. Indexing the Threshold Value to the ODCM setpoints in this manner ensures that the Threshold Value will never be less than the setpoint established by a specific discharge permit.

Threshold Value #2 is intended for effluent monitoring on non-routine release pathways for which a discharge permit would not normally be prepared. The ODCM establishes a methodology for determining effluent radiation monitor setpoints. The ODCM specifies default source terms and, for gaseous releases, prescribes the use of pre-determined annual average meteorology in the most limiting downwind sector for showing compliance with the regulatory commitments. These monitor reading Threshold Values are determined using this methodology.

Threshold Value #3 addresses uncontrolled releases that are detected by sample analyses, particularly on unmonitored pathways, e.g., spills of radioactive liquids into storm drains, heat exchanger leakage in river water systems, etc.

RU2

Initiating Condition (Back to Cold IC p.108)(Back to Hot IC p.107)

Unexpected Rise in Plant Radiation.

Operating Mode Applicability: All

Threshold Values: (1.a **AND** b. **OR** 2)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, **OR** fuel transfer canal with all irradiated fuel assemblies remaining covered by water.

Personnel report of low water level
Annunciator EH2 “SFP LVL HI/LO”
Personnel report of cavitation **OR** low discharge pressure for SFP Pump Discharge Pressure **AND/OR** RHR Pump Discharge Pressure

AND

b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:

RE-0005 in the fuel building
RE-0002 in containment
RE-0027A **OR** B in containment

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1,000 over normal* levels.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

VALID: an indication, report, or condition, is considered to be VALID when it is verified by (1) an instrument channel check, or (2) indications on related or redundant indicators, or (3) by direct observation by plant personnel, such that doubt related to the indicator's operability, the condition's existence, or the report's accuracy is removed. Implicit in this definition is the need for timely assessment.

This IC addresses raised radiation levels as a result of water level lowering above the RPV flange or events that have resulted, or may result, in unexpected rises in radiation dose rates within plant buildings. These radiation rises represent a loss of control over radioactive material and may represent a potential degradation in the level of safety of the plant.

Classification as a NOUE is warranted as a precursor to a more serious event. Indications include instrumentation such as water level and local area radiation monitors, equipment parameters and personnel (e.g., refueling crew) reports. If available, video cameras may allow remote observation. Depending on available level instrumentation, the declaration threshold may need to be based on indications of water makeup rate or lowering in refueling water storage tank level.

Threshold Value 1a: Personnel report of low water level is the primary indicator.

Threshold Value 1b limits:
FSAR 9.1.3.1.2 indicates that 12 feet of water above the stored fuel yields a dose rate of about 2.5 mR/hr. These monitors do not directly see the fuel, but see the dose reflected from the ceiling or containment dome, A conservative reduction on reflection is 10^{-2} . Thus when on-scale, the lowest reading monitor (RE-0005, 10^4 mR/hr, Table 3.3-6) equivalent reading at the pool surface would be 10^6 mR/hr. Then a rise from a general area dose rate of 2.5 mR/hr represents log (10^6/2.5) = 5.6 1/10th thicknesses. Assuming a 1/10th thickness is 2 feet, the monitor reading rise is equivalent to a water level lowering of 5.6 x 2 = 11.2 feet. Thus the fuel would have at least 6 inches of water cover.

Threshold Value #2 addresses UNPLANNED VALID Direct Area Radiation Monitor readings indicating rises in in-plant radiation levels that represent a degradation in the control of radioactive material, and represent a potential degradation in the level of safety of the plant. "Direct Radiation Area Monitors" include installed Area Radiation Monitors (ARMs) but does not include effluent monitors (which are addressed in separate EALs) or process monitors.

4.2 Category F - Fission Product Barrier

Fission Product Barrier Degradation

NOUE		ALERT		SITE AREA EMERGENCY		GENERAL EMERGENCY
FU1	ANY Loss or ANY Potential Loss of Containment *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FA1**	ANY Loss or ANY Potential Loss of **EITHER** Fuel Clad **OR** RCS *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FS1**	Loss or Potential Loss of ANY Two Barriers *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*	**FG1** Loss of ANY Two Barriers **AND** Loss or Potential Loss of Third Barrier *Op. Modes: Power Operation, Hot Standby, Startup, Hot Shutdown*

NOTES

1. The logic used for these initiating conditions reflects the following considerations:

- The Fuel Clad Barrier and the RCS Barrier are weighted more heavily than the Containment Barrier. NOUE ICs associated with RCS and Fuel Clad Barriers are addressed under System Malfunction ICs.

- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from the threshold for a General Emergency. For example, if Fuel Clad and RCS Barrier “Loss” Threshold Values existed, that, in addition to offsite dose assessments, would require continual assessments of radioactive inventory and containment integrity. Alternatively, if both Fuel Clad and RCS Barrier “Potential Loss” Threshold Values existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.

- The ability to escalate to higher emergency classes as an event deteriorates must be maintained. For example, RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Farley Fission Product Barrier Evaluation

FUEL CLAD BARRIER Threshold Values: (Back to FPB p.106)

The Fuel Clad Barrier is the zircalloy or stainless steel tubes that contain the fuel pellets.

1. Critical Safety Function Status

NOTE: Heat Sink CSF should not be considered –RED if total AFW flow is less than 395 gpm due to operator action.

RED path indicates an extreme challenge to the safety function. ORANGE path indicates a severe challenge to the safety function.

Core Cooling - ORANGE indicates subcooling has been lost and that some clad damage may occur. Heat Sink - RED indicates the ultimate heat sink function is under extreme challenge and thus these two items indicate potential loss of the Fuel Clad Barrier.

Core Cooling - RED indicates significant superheating and core uncovery and is considered to indicate loss of the Fuel Clad Barrier.

2. Primary Coolant Activity Level

The 300 μCi/gm I-131 equivalent. Assessment by the NUMARC EAL Task Force indicates that this amount of coolant activity is well above that expected for iodine spikes and corresponds to less than 5% fuel clad damage. This amount of radioactivity indicates significant clad damage and thus the Fuel Clad Barrier is considered lost.

There is no equivalent "Potential Loss" Threshold Value for this item.

3. Core Exit Thermocouple Readings

Core Exit Thermocouple Readings are included in addition to the Critical Safety Functions to include conditions when the CSFs may not be in use.

The "Loss" Threshold Value of 1200 degrees F corresponds to significant superheating of the coolant. This value corresponds to the temperature reading that indicats core cooling - RED in Fuel Clad Barrier Threshold Value #1.

The "Potential Loss" Threshold Value of 700 degrees F corresponds to loss of subcooling. This value corresponds to the temperature reading that indicates core cooling - ORANGE in Fuel Clad Barrier Threshold Value #1.

4. Reactor Vessel Water Level

There is no "Loss" Threshold Value corresponding to this item because it is better covered by the other Fuel Clad Barrier "Loss" Threshold Values.
The 0% RVLIS value for the "Potential Loss" Threshold Value corresponds to the top of the active fuel. The "Potential Loss" Threshold Value is defined by the Core Cooling - ORANGE path.

5. Containment Radiation Monitoring

The greater than 600 R/hr reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 μCi/gm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss Threshold Value #4. Thus, this Threshold Value indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

There is no "Potential Loss" Threshold Value associated with this item.

7. Emergency Director Judgment

This Threshold Value addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier is incorporated in this Threshold Value as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

RCS BARRIER Threshold Values: (Back to FPB p.17)

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. Critical Safety Function Status

There is no "Loss" Threshold Value associated with this item.

NOTE:	**Heat Sink CSF should not be considered –RED if total AFW flow is less than 395 gpm due to operator action.**

This Threshold Value uses the Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. An RCS Integrity RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

2. RCS Leak Rate

The "Loss" Threshold Value addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

The "Potential Loss" Threshold Value is based on the inability to maintain normal liquid inventory within the Reactor Coolant System (RCS) by normal operation of the Chemical and Volume Control System which is considered as one centrifugal charging pump discharging to the charging header. A second charging pump being required is indicative of a substantial RCS leak providing the 120 GPM value.

3. SG Tube Rupture

This Threshold Value is intended to address the full spectrum of Steam Generator (SG) tube rupture events in conjunction with Containment Barrier "Loss" Threshold Value #4 and Fuel Clad Barrier Threshold Values. The "Loss" Threshold Value addresses RUPTURED SG(s) for which the leakage is large enough to cause actuation of ECCS (SI). This is consistent to the RCS Barrier “Potential Loss” Threshold Value #2. This condition is described by EEP-3.0 entered. By itself, this Threshold Value will result in the declaration of an Alert. However, if the SG is also FAULTED (i.e., two barriers failed), the declaration escalates to a Site Area Emergency per Containment Barrier "Loss" Threshold Value #4.

There is no "Potential Loss" Threshold Value.

4. Containment Radiation Monitoring

The RE-2 greater than 1.0 R/hr and RE-7 greater than 500 mR/hr threshold is a value which indicates the release of reactor coolant to the containment. The reading is calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with normal operating concentrations (i.e., within T/S) into the containment atmosphere. This value is less than that specified for Fuel Clad Barrier Threshold Value #5. Thus, this Threshold Value would be indicative of a RCS leak only. If the radiation monitor reading rise to that specified by Fuel Clad Barrier Threshold Value #5, fuel damage would also be indicated.

There is no "Potential Loss" Threshold Value associated with this item.

5. Other Indications

There is no "Loss" Threshold Value associated with this item.

An unexplained level rise in the containment sump, Reactor Coolant Drain Tank or the Waste Holdup Tank could indicate a RCS leak and is therefore included as a Potential Loss of the RCS Barrier. Rises in the containment sump levels, Reactor Coolant Drain Tank or the Waste Holdup Tank that are anticipated as part of a planned evolution (or have already been identified by other leak assessment measures and accounted for in other threshold values) do not meet this threshold. Likewise, rises in containment sump levels, Reactor Coolant Drain Tank or the Waste Holdup Tank that can be readily attributed to a source other than the RCS would also not meet the threshold. Sump and tank level increases must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

6. Emergency Director Judgment

This Threshold Value addresses any other factors that are to be used by the Emergency Director in determining whether the RCS barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this Threshold Value as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

CONTAINMENT BARRIER Threshold Values: (Back to FPB p.176)

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Critical Safety Function Status

There is no "Loss" Threshold Value associated with this item.

This Threshold Value uses Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. Containment RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings and/or sampling results, and thus represents a potential loss of containment. Conditions leading to a containment RED path result from RCS barrier and/or Fuel Clad Barrier Loss. Thus, this Threshold Value is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

2. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure rise indicates a loss of containment integrity. Containment pressure and sump levels should rise as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity.

The 54 PSIG for potential loss of containment is based on the containment design pressure. Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit (greater than 6%) curve exists. The indications of potential loss under this Threshold Value corresponds to some of those leading to the RED path in Threshold Value #1 above and may be declared. As described above, this Threshold Value is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

The CTMT CSF ORANGE condition represents a potential loss of containment in that the containment heat removal/depressurization system (e.g., containment sprays, fan coolers, etc., but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint at which the equipment was supposed to have actuated.

3. Core Exit Thermocouples

There is no "Loss" Threshold Value associated with this item.

In this Threshold Value, the function restoration procedures are those emergency operating procedures that address the recovery of the core cooling critical safety functions. The procedure is considered effective if the temperature is decreasing or if the vessel water level is increasing. For units using the CSF status trees a direct correlation to those status trees can be made if the effectiveness of the restoration procedures is also evaluated as stated below.

Severe accident analyses (e.g., NUREG-1150) have concluded that function restoration procedures can arrest core degradation within the reactor vessel in a significant fraction of the core damage scenarios, and that the likelihood of containment failure is very small in these events. Given this, it is appropriate to provide a reasonable period to allow function restoration procedures to arrest the core melt sequence. Whether or not the procedures will be effective should be apparent within 15 minutes. The Emergency Director should make the declaration as soon as it is determined that the procedures have been, or will be ineffective. The reactor vessel level chosen should be consistent with the emergency response guides applicable to the facility.

The conditions in this potential loss Threshold Value represent an imminent core melt sequence which, if not corrected, could lead to vessel failure and an raised potential for containment failure. In conjunction with the Core Cooling and Heat Sink criteria in the Fuel and RCS barrier columns, this Threshold Value would result in the declaration of a General Emergency -- loss of two barriers and the potential loss of a third. If the function restoration procedures are ineffective, there is no "success" path.

4. SG Secondary Side Release With Primary To Secondary Leakage

This “loss” Threshold Value recognizes that SG tube leakage can represent a bypass of the containment barrier as well as a loss of the RCS barrier. The “loss” Threshold Value addresses the condition in which a RUPTURED steam generator is also FAULTED. This condition represents a bypass of the RCS and containment barriers. In conjunction with RCS Barrier “loss” Threshold Value #3, this would always result in the declaration of a Site Area Emergency.

The other leakage “loss” Threshold Value addresses SG tube leaks that exceed 10 gpm in conjunction with a nonisolable release path to the environment from the affected steam generator. The threshold for establishing the nonisolable secondary side release is intended to be a prolonged release of radioactivity from the RUPTURED steam generator directly to the environment. This could be expected to occur when the main condenser is unavailable to accept the contaminated steam (i.e., SGTR with concurrent loss of offsite power and the RUPTURED steam generator is required for plant cooldown or a stuck open relief valve). If the main condenser is available, there may be releases via air ejectors, gland seal exhausters, and other similar controlled, and often monitored, pathways. These pathways do not meet the intent of a nonisolable release path to the environment. These minor releases are assessed using Abnormal Rad Levels / Radiological Effluent ICs.

5. Containment Isolation Valve Status After Containment Isolation

This Threshold Value addresses incomplete containment isolation that allows direct release to the environment. It represents a loss of the containment barrier.

The use of the modifier “direct” in defining the release path discriminates against release paths through interfacing liquid systems. The existence of an in–line charcoal filter does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

There is no "Potential Loss" Threshold Value associated with this item.

6. Significant Radioactive Inventory in Containment

There is no "Loss" Threshold Value associated with this item.

The greater than 8,000 R/hr value indicates significant fuel damage well in excess of the Threshold Values associated with both loss of Fuel Clad and loss of RCS Barriers. A major release of radioactivity requiring offsite protective actions from core damage is not possible unless a major failure of fuel cladding allows radioactive material to be released from the core into the reactor coolant.

Regardless of whether containment is challenged, this amount of activity in containment, if released, could have such severe consequences that it is prudent to treat this as a potential loss of containment, such that a General Emergency declaration is warranted. NUREG-1228, "Source Estimations During Incident Response to Severe Nuclear Power Plant Accidents," indicates that such conditions do not exist when the amount of clad damage is less than 20%. A radiation monitor reading corresponding to 20% fuel clad damage is specified here.

7. Other Indications

Leakage from the Containment would be routed through various ventilation systems where the specific monitors would indicate a release. R10, R14, R21, **OR** R22 Alarms would indicate a breach in containment.

8. Emergency Director Judgment

This Threshold Value addresses any other factors that are to be used by the Emergency Director in determining whether the Containment barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this Threshold Value as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost.

4.3 Category S - System Malfunctions - Hot Matrix

SG1

Initiating Condition (Back to Hot IC p. 107)

Prolonged Loss of All Offsite Power **AND** Prolonged Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1 **AND EITHER** 2 **OR** 3)

1. Loss of ALL AC power indicated by:

   a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G for greater than 15 minutes.

   AND

   b. Failure of emergency diesel generators to supply power to emergency busses.

   AND EITHER

2. Restoration of at least one 4160V ESF bus, 1(2)F **OR** 1(2)G, within 4 hr. of time of loss is **NOT** likely.

   OR

3. Fuel Clad Barrier Evaluation indicates continuing degradation (Loss or Potential Loss) due to core cooling.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will lead to loss of fuel clad, RCS, and containment. The 4 hours to restore AC power is based on a site blackout coping analysis. Appropriate allowance for offsite emergency response including evacuation of surrounding areas should be considered. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response.

This IC is specified to assure that in the unlikely event of a prolonged station blackout, timely recognition of the seriousness of the event occurs and that declaration of a General Emergency occurs as early as is appropriate, based on a reasonable assessment of the event trajectory.

The likelihood of restoring at least one emergency bus should be based on a realistic appraisal of the situation since a delay in an upgrade decision based on only a chance of mitigating the event could result in a loss of valuable time in preparing and implementing public protective actions.

In addition, under these conditions, fission product barrier monitoring capability may be degraded. Although it may be difficult to predict when power can be restored, it is necessary to give the Emergency Director a reasonable idea of how quickly (s)he may need to declare a General Emergency based on two major considerations:

1. Are there any present indications that core cooling is already degraded to the point that Loss or Potential Loss of Fission Product Barriers is imminent?

2. If there are no present indications of such core cooling degradation, how likely is it that power can be restored in time to assure that a loss of two barriers with a potential loss of the third barrier can be prevented?

Thus, indication of continuing core cooling degradation must be based on Fission Product Barrier monitoring with particular emphasis on Emergency Director judgment as it relates to imminent Loss or Potential Loss of fission product barriers and degraded ability to monitor fission product barriers.

SG2

Initiating Condition (Back to Hot IC p. 107)

Failure of the Reactor Protection System to Complete an Automatic Trip **AND** Manual Trip was NOT Successful **AND** there is Indication of an Extreme Challenge to the Ability to Cool the Core.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)

Threshold Value: (1 **AND EITHER** a **OR** b)

NOTE:	**A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.**

NOTE:	**Heat Sink CSF should not be considered RED if total AFW flow is less than 395 gpm due to operator action.**

NOTE	**Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.**

1. Indications exist that a reactor protection system setpoint was exceeded and automatic trip did not occur, and a manual trip did not result in the reactor being made subcritical.

   AND EITHER

   a. Core Cooling CSF - RED

   OR

   b. Heat Sink CSF - RED

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console is required to trip the reactor. Under the conditions of this IC and its associated Threshold Values, the efforts to bring the reactor subcritical have been unsuccessful and, as a result, the reactor is producing more heat than the maximum decay heat load for which the safety systems were designed. Although there are capabilities away from the reactor control console, such as emergency boration, the continuing temperature rise indicates that these capabilities are not effective. This situation could be a precursor for a core melt sequence. This Threshold Value equates to a Subcriticality RED condition.

The extreme challenge to the ability to cool the core is intended to mean that the core exit temperatures are at or approaching 1200 degrees F or that the reactor vessel water level is below the top of active fuel. This Threshold Value equates to a Core Cooling RED condition.

Another consideration is the inability to initially remove heat during the early stages of this sequence. If emergency feedwater flow is insufficient to remove the amount of heat required by design from at least one steam generator, an extreme challenge should be considered to exist. This Threshold Value equates to a Heat Sink RED condition.

In the event either of these challenges exist at a time that the reactor has not been brought below the power associated with the safety system design (typically 3 to 5% power) a core melt sequence exists. In this situation, core degradation can occur rapidly. For this reason, the General Emergency declaration is intended to be anticipatory of the fission product barrier matrix declaration to permit maximum offsite intervention time.

SS1

Initiating Condition (Back to Hot IC p. 107)

Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:
Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1a **AND** 1b **AND** 1c)

1. Loss of all AC power indicated by:
   a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G for greater than 15 minutes.

      AND

   b. Failure of emergency diesel generators to supply power to emergency busses.

      AND

   c. Restoration of at least one 4160V ESF bus, F **OR** G, has **NOT** occurred within 15 minutes of time of loss of all AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal and the Ultimate Heat Sink. Prolonged loss of all AC power will cause core uncovering and loss of containment integrity, thus this event can escalate to a General Emergency. The 15 minute time duration is selected to exclude transient or momentary power losses.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads are not operable on the energized bus then the bus should not be considered operable.

SS2

Initiating Condition (Back to Hot IC p. 107)

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded **AND** Manual Trip Was NOT Successful.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)

Threshold Value: (1)

NOTE: A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.

NOTE Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.

1. Indications exist that a reactor protection system setpoint was exceeded and automatic trip did not occur, and a manual trip did not result in the reactor being made subcritical (NOTE).

Basis:

Automatic and manual trip are not considered successful if action away from the reactor control console was required to trip the reactor.

The Reactor should be considered subcritical when reactor power level has been reduced to less than 5% power and SUR is negative.

Under these continued power generation conditions, the reactor may be producing more heat than the maximum decay heat load for which the safety systems are designed. A Site Area Emergency is indicated because conditions exist that may lead to imminent loss or potential loss of both fuel clad and RCS. Although this IC may be viewed as redundant to the Fission Product Barrier Degradation IC, its inclusion is necessary to better assure timely recognition and emergency response. Escalation of this event to a General Emergency would be via Fission Product Barrier Degradation or Emergency Director Judgment ICs. A manual reactor trip is considered to be a trip input to the automatic Reactor Protection System.

SS3

Initiating Condition (Back to Hot IC p. 107)

Loss of All Vital DC Power.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1)

1. Loss of all Vital DC power to 125 VDC Bus A **AND** B indicated by bus voltage indications less than 105 VDC for greater than 15 minutes.

Basis:

Loss of all DC power compromises ability to monitor and control plant safety functions. Prolonged loss of all DC power will cause core uncovering and loss of containment integrity when there is significant decay heat and sensible heat in the reactor system.

105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads.

SS4

Initiating Condition (Back to Hot IC p. 107)

Complete Loss of Heat Removal Capability.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1a **AND** 1b)

NOTE:	**Heat Sink CSF should not be considered –RED if total AFW flow is less than 395 gpm due to operator action.**

1. Complete Loss of Heat Removal Capability as indicated by:

   a. Core Cooling CSF - ORANGE

   AND

   b. Heat Sink CSF - RED

Basis:

This Threshold Value addresses complete loss of functions, including ultimate heat sink (NSCW), required for hot shutdown with the reactor at pressure and temperature. Reactivity control is addressed in other Threshold Values.

Under these conditions, there is an actual major failure of a system intended for protection of the public. Thus, declaration of a Site Area Emergency is warranted.

SS6

Initiating Condition (Back to Hot IC p. 107)

Inability to Monitor a SIGNIFICANT TRANSIENT in Progress.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1a **AND** 1b **AND** 1c **AND** 1d)

1. a. A SIGNIFICANT TRANSIENT in progress.

   AND

   b. Loss of most **OR** all (approximately 75% of the MCB) annunciators **OR** indicators associated with safety systems.

   AND

   c. Compensatory non-alarming indications are **NOT** available.

   AND

   d. Indications needed to monitor the Critical Safety Function Status Tree parameters are **NOT** available.

Basis:

SIGNIFICANT TRANSIENT: is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations greater than 10%.

This IC and its associated Threshold Value are intended to recognize the inability of the control room staff to monitor the plant response to a transient. A Site Area Emergency is considered to exist if the control room staff cannot monitor safety functions needed for protection of the public.

The annunciators for this Threshold Value are limited to include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other Threshold Values.

"Compensatory non-alarming indications" in this context includes computer based information such as SPDS.

The indications needed to monitor safety functions necessary for protection of the public must include control room indications, computer generated indications and dedicated annunciation capability. The specific indications are those used to determine such functions as the ability to shut down the reactor, maintain the core cooled, to maintain the reactor coolant system intact, and to maintain containment intact.

"Planned" and “UNPLANNED” actions are not differentiated since the loss of instrumentation of this magnitude is of such significance during a transient that the cause of the loss is not an ameliorating factor.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is a greater risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide augmented monitoring of system operation.

SA2

Initiating Condition (Back to Hot IC p. 107)

Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded **AND** Manual Trip Was Successful.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)

Threshold Value: (1)

NOTE:	**A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.**

NOTE	**Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.**

1. Indication(s) exist that a reactor protection setpoint was exceeded and an automatic trip did not occur, and a manual trip resulted in the reactor being subcritical.

Basis:

This condition indicates failure of the automatic protection system to trip the reactor. This condition is more than a potential degradation of a safety system in that a front line automatic protection system did not function in response to a plant transient and thus the plant safety has been compromised, and design limits of the fuel may have been exceeded. An Alert is indicated because conditions exist that lead to potential loss of fuel clad or RCS. Reactor protection system setpoint being exceeded, rather than limiting safety system setpoint being exceeded, is specified here because failure of the automatic protection system is the issue. A manual reactor trip is considered to be a trip input to the automatic Reactor Protection System or de-energizing the MG sets should initiate a manual trip.

The Reactor should be considered subcritical when reactor power level has been reduced to less than 5% power and SUR is negative.

SA4

Initiating Condition (Back to Hot IC p. 107)

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in the Control Room With **EITHER** (1) a SIGNIFICANT TRANSIENT in Progress, **OR** (2) Compensatory Non-Alarming Indicators are Unavailable.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1 **AND EITHER** a **OR** b)

1. UNPLANNED loss of most **OR** all (approximately 75% of the MCB) annunciators **OR** indicators associated with safety systems for greater than 15 minutes.

AND EITHER

a. A SIGNIFICANT TRANSIENT is in progress.

OR

b. Compensatory non-alarming indications are **NOT** available.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

SIGNIFICANT TRANSIENT: is an UNPLANNED event involving one or more of the following: (1) automatic turbine runback greater than 25% thermal reactor power, (2) electrical load rejection greater than 25% full electrical load, (3) Reactor Trip, (4) Safety Injection Activation, or (5) thermal power oscillations greater than 10%.

This IC and its associated Threshold Values are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment during a transient. Recognition of the availability of computer based indication equipment is considered. A "planned" loss of annunciators or indicators includes scheduled maintenance and testing activities.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is a greater risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions. It is also not intended that the Shift Supervisor be tasked with making a judgment decision as to whether additional personnel are required to provide augmented monitoring of system operation.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this Threshold Value due to difficulty associated with assessment of plant conditions.

The annunciators or indicators for this Threshold Value include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other Threshold Values.

"Compensatory non-alarming indications" in this context includes computer based information such as SPDS. This should include all computer systems available for this use depending on specific plant design and subsequent retrofits. If both a major portion of the annunciation system and all computer monitoring are unavailable, the Alert is required.

SA5

Initiating Condition (Back to Hot IC p. 107)

AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value:

(1a **AND** 1b)

1. a. AC power capability to 4160V ESF busses 1(2)F **AND** 1(2)G reduced to a single power source for greater than 15 minutes.

   AND

   b. ANY additional single failure will result in station blackout.

Basis:

This IC and the associated Threshold Values are intended to provide an escalation from IC SU1, "Loss of All Offsite Power To Essential Busses for Greater Than 15 Minutes." The condition indicated by this IC is the degradation of the offsite and onsite power systems such that any additional single failure would result in a station blackout. This condition could occur due to a loss of offsite power with a concurrent failure of one emergency generator to supply power to its emergency busses. Another related condition could be the loss of all offsite power and loss of onsite emergency diesels with only one train of emergency busses being backfed from the SAT, or the loss of onsite emergency diesels with only one train of emergency busses being backfed from offsite power. The subsequent loss of this single power source would escalate the event to a Site Area Emergency in accordance with IC SS1, "Loss of All Offsite and Loss of All Onsite AC Power to Essential Busses".

The Threshold Values allow credit for operation of installed design features, such as cross-ties or swing diesels, provided that abnormal or emergency operating procedures address their use.

SU1

Initiating Condition (Back to Hot IC p. 107)

Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1a **AND** 1b)

1. a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G for greater than 15 minutes

   AND

   b. Emergency diesel generators supplying power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G.

Basis:

Prolonged loss of Offsite AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

SU2

Initiating Condition (Back to Hot IC p. 107)

Inability to Reach Required Shutdown Within Technical Specification Limits.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: (1)

1. Plant is **NOT** brought to required operating mode within FNP Technical Specifications LCO Action Statement Time limit.

Basis:

Limiting Conditions of Operation (LCOs) require the plant to be brought to a required shutdown mode when the Technical Specification required configuration cannot be restored. Depending on the circumstances, this may or may not be an emergency or precursor to a more severe condition. In any case, the initiation of plant shutdown required by the site Technical Specifications requires a four hour report under 10 CFR 50.72 (b) Non-emergency events. The plant is within its safety envelope when being shut down within the allowable action statement time in the Technical Specifications. An immediate NOUE is required when the plant is not brought to the required operating mode within the allowable action statement time in the Technical Specifications. Declaration of a NOUE is based on the time at which the LCO-specified action statement time period elapses under the site Technical Specifications and is not related to how long a condition may have existed. Other required Technical Specification shutdowns that involve precursors to more serious events are addressed by other System Malfunction, Hazards, or Fission Product Barrier Degradation ICs.

SU3

Initiating Condition (Back to Hot IC p. 107)

UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value:

(1)

1. UNPLANNED loss of most **OR** all (approximately 75% of the MCB annunciators) **OR** indicators associated with safety systems for greater than 15 minutes.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC and its associated Threshold Value are intended to recognize the difficulty associated with monitoring changing plant conditions without the use of a major portion of the annunciation or indication equipment.

Recognition of the availability of computer based indication equipment is considered.

Quantification of "Most" is arbitrary, however, it is estimated that if approximately 75% of the safety system annunciators or indicators are lost, there is a greater risk that a degraded plant condition could go undetected. It is not intended that plant personnel perform a detailed count of the instrumentation lost but use the value as a judgment threshold for determining the severity of the plant conditions.

It is further recognized that most plant designs provide redundant safety system indication powered from separate uninterruptible power supplies. While failure of a large portion of annunciators is more likely than a failure of a large portion of indications, the concern is included in this Threshold Value due to difficulty associated with assessment of plant conditions.

The annunciators or indicators for this Threshold Value include those identified in the Abnormal Operating Procedures, in the Emergency Operating Procedures, and in other Threshold Values. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

SU4

Initiating Condition (Back to Hot IC p. 107)

Fuel Clad Degradation.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Values: (1)

1. RCS coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits as indicated by ANY of the following:

Dose Equivalent I-131 greater than 0.5 μCi/gm for greater than 48 hours
Dose Equivalent I-131 greater than Technical Specification figure 3.4.16-1. **IF** less than 20% power, **THEN** use the Dose Equivalent I-131 20% power limit on Technical Specification figure 3.4.16-1
RCS gross specific activity greater than 100/Ē μCi/gm.

Basis:

This IC is included as a NOUE because it is considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems.

The Threshold Value addresses coolant samples exceeding coolant technical specifications for iodine spike. Escalation of this IC to the Alert level is via the Fission Product Barrier Degradation Monitoring ICs. Though the referenced Technical Specification limits are mode dependent, it is appropriate that the Threshold Value's be applicable in all modes, as they indicate a potential degradation in the level of safety of the plant.

SU5

Initiating Condition (Back to Hot IC p. 107)

RCS Leakage.

Operating Mode Applicability: Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Values: (1 **OR** 2)

1. RCS Unidentified **OR** pressure boundary leakage greater than 10 gpm.

2. RCS Identified leakage greater than 25 gpm.

Basis:

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. The 10 gpm value for the unidentified and pressure boundary leakage was selected as it is observable with normal control room indications. Lesser values must generally be determined through time-consuming surveillance tests (e.g., mass balances). The Threshold Value for identified leakage is set at a higher value due to the lesser significance of identified leakage in comparison to unidentified or pressure boundary leakage.

SU6

Initiating Condition (Back to Hot IC p. 107)

UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities.

Operating Mode Applicability:

Power Operation (Mode 1)
Startup (Mode 2)
Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Values: (1 **OR** 2)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:

In plant telephones
Public Address System
Plant radio systems

2. UNPLANNED loss of ALL of the following off-site communications capability:

ENN (Emergency Notification Network)
ENS (Emergency Notification System)
Commercial phones (Radio, PBX, Satellite, Wireless)
VOIP (Voice Over Internet Protocol)
OPX (Off Premise Extension)

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

The purpose of this IC and its associated Threshold Values is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant conditions. This Threshold Value is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being used to make communications possible. The list for onsite communications loss encompasses the loss of all means of routine communications. The list for offsite communications loss encompasses the loss of all means of communications with offsite authorities.

SU8

Initiating Condition (Back to Hot IC p. 107)

Inadvertent Criticality.

OPERATING MODE APPLICABILITY Hot Standby (Mode 3)
Hot Shutdown (Mode 4)

Threshold Value: **(1)**

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC addresses inadvertent criticality events. While the primary concern of this IC is criticality events that occur in Cold Shutdown or Refueling modes (NUREG 1449, Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States), the IC is applicable in other modes in which inadvertent criticalities are possible. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification. This IC excludes inadvertent criticalities that occur during planned reactivity changes associated with reactor startups.

This condition is identified using the startup rate monitor. The term “sustained“ is used in order to allow exclusion of expected short term positive startup rates from planned control rod movements. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

4.4 CATEGORY H - HAZARDS AND OTHER CONDITIONS AFFECTING PLANT SAFETY

HG1

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

HOSTILE ACTION Resulting in Loss Of Physical Control of the Facility.

Operating Mode Applicability: All

Threshold Value: (1 **OR** 2)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

Basis:

HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

VITAL AREAS: any areas, normally within the PROTECTED AREA, that contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.

IMMINENT: Mitigation actions have been ineffective, additional actions are not expected to be successful, and trended information indicates that the event or condition will occur.

This IC encompasses conditions under which a HOSTILE ACTION has resulted in a loss of physical control of VITAL AREAs (containing vital equipment or controls of vital equipment) resulting in the loss of one or more safety functions and control of the equipment required to maintain the safety function(s) cannot be transferred to and operated from another location. These safety functions are reactivity control, RCS inventory, and secondary heat removal. If control of the plant equipment necessary to maintain safety functions can be transferred to another location, then the above initiating condition is not met.

Threshold Value 2 should also address loss of physical control of spent fuel pool cooling systems if IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool. If the calculated SFP “time to boil” is 2 hours or less, spent fuel damage is likely.

HG2

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency.

Operating Mode Applicability: All

Threshold Value: (1)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity **OR** HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Basis:

HOSTILE ACTION: An act toward an NPP or its personnel that includes the use of violent force to destroy equipment, takes hostages, and /or intimidates the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non- terrorism-based Threshold Values should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

This Threshold Value is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the General Emergency class.

HS2

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Control Room Evacuation Has Been Initiated **AND** Plant Control Cannot Be Established.

Operating Mode Applicability: All

Threshold Value: (1a **AND** 1b)

1. a. Control Room evacuation has been initiated.

   AND

   b. Control of the plant can **NOT** be established per AOP-28.0, Control Room Inaccessibility, within 15 minutes.

Basis:

Expeditious transfer of safety systems has not occurred but fission product barrier damage may not yet be indicated. The intent of this IC is to capture those events where control of the plant cannot be reestablished in a timely manner. The time for transfer is based on analysis or assessments as to how quickly control must be reestablished without core uncovering and/or core damage. The determination of whether or not control is established at the remote shutdown panel is based on Emergency Director (ED) judgment. The ED is expected to make a reasonable, informed judgment within the time for transfer that the operators have control of the plant.

The intent of the Threshold Value is to establish control of important plant equipment and knowledge of important plant parameters in a timely manner. Primary emphasis should be placed on those components and instruments that supply protection for and information about safety functions. These safety functions are reactivity control, RCS inventory, and secondary heat removal.

HS3

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency.

Operating Mode Applicability: All

Threshold Value: (1)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which involve actual **OR** likely major failures of plant functions needed for protection of the public **OR** HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

Basis:

HOSTILE ACTION: An act toward an NPP or its personnel that includes the use of violent force to destroy equipment, take hostages, and /or intimidate the licensee to achieve an end. This includes attack by air, land, or water using weapons, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based Threshold Values should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

This Threshold Value is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the emergency class description for Site Area Emergency.

HS4

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

HOSTILE ACTION within the PROTECTED AREA

Operating Mode Applicability: **All**

Threshold Value: (1)

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the site security force.

Basis:

HOSTILE ACTION: An act toward an NPP or its personnel that includes the use of violent force to destroy equipment, take hostages, and /or intimidate the licensee to achieve an end. This includes attack by air, land, or water using weapons, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based Threshold Values should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

PROJECTILE: An object directed toward a NPP that could cause concern for its continued operability, reliability, or personnel safety.

This class of security events represents an escalated threat to plant safety above that contained in the Alert IC in that a HOSTILE ACTION has progressed from the Owner Controlled Area to the Protected Area.

This EAL addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the PROTECTED AREA. Those events are adequately addressed by other EALs.

Escalation of this emergency classification level, if appropriate, would be based on actual plant status after impact or progression of attack.

HA1

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Natural and Destructive Phenomena Affecting the Plant VITAL AREA.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3 **OR** 4 **OR** 5 **OR** 6)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5 indicates Seismic event greater than Operating Basis Earthquake (OBE).

2. Tornado **OR** high winds greater than 115 mph within the PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures/equipment **OR** the Control Room has indication of degraded performance of any listed systems:

Containment	Auxiliary Building
Service Water Intake Structure (SWIS)	Service Water Pond
Refueling Water Storage Tank (RWST)	Condensate Storage Tank (CST)
Diesel Generator Building	

3. Vehicle crash within PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures **OR** equipment therein **OR** Control Room has indication of degraded performance of those systems:

Containment	Auxiliary Building
Service Water Intake Structure (SWIS)	Service Water Pond
Refueling Water Storage Tank (RWST)	Condensate Storage Tank (CST)
Diesel Generator Building	

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to **OR** penetration of any of the following plant areas containing safety-related equipment, their controls **OR** their power supplies.

Containment	Auxiliary Building
Refueling Water Storage Tank (RWST)	Condensate Storage Tank (CST)
Diesel Generator Building	Control Room

5. Uncontrolled flooding in the following areas that results in degraded safety system performance as indicated in the control room **OR** that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate **OR** monitor safety equipment.

Service Water Intake Structure (SWIS)
Auxiliary Building
Turbine Building Basement

6. Sustained hurricane winds greater than 74 mph onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems.

Basis:

PROTECTED AREA: the area which normally encompasses all controlled areas within the security protected area fence.

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

The Threshold Values in this IC escalate from the NOUE Threshold Values in HU1 in that the occurrence of the event has resulted in VISIBLE DAMAGE to plant structures or areas containing equipment necessary for a safe shutdown, or has caused damage to the safety systems in those structures evidenced by control indications of degraded system response or performance. The occurrence of VISIBLE DAMAGE and/or degraded system response is intended to discriminate against lesser events. The initial "report" should not be interpreted as mandating a lengthy damage assessment prior to classification. No attempt is made in this Threshold Value to assess the actual magnitude of the damage. The significance here is not that a particular system or structure was damaged, but rather, that the event was of sufficient magnitude to cause this degradation.

Threshold Value #1 is based on the OBE earthquake FSAR design basis. Seismic events of this magnitude can result in a plant VITAL AREA being subjected to forces beyond design limits, and thus damage may be assumed to have occurred to plant safety systems.

Threshold Value #2 is based on the FSAR design basis. Wind loads greater than 115 mph can cause damage to safety functions and is read from the plant meteorological tower.

Threshold Value #3 addresses crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

Threshold Value #4 addresses the threat to safety related equipment imposed by missiles generated by main turbine rotating component failures. This list of areas include areas containing safety-related equipment, their controls, and their power supplies. This Threshold Value is, therefore, consistent with the definition of an ALERT in that if missiles have damaged or penetrated areas containing safety-related equipment the potential exists for substantial degradation of the level of safety of the plant.

Threshold Value #5 addresses the effect of internal flooding that has resulted in degraded performance of systems affected by the flooding, or has created industrial safety hazards (e.g., electrical shock) that preclude necessary access to operate or monitor safety equipment. The inability to operate or monitor safety equipment represents a potential for substantial degradation of the level of safety of the plant. This flooding may have been caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. TS-PSA-001, Table 3.3.6-4 provides that the SWIS is vulnerable due to failure of the cooling water lines or discharge expansion joints. The Auxiliary Building is vulnerable due to failure of the Service Water piping. The Turbine Building basement is vulnerable to Circulating Water line breaks. The areas include those areas that contain systems required for safe shutdown of the plant, that are not designed to be wetted or submerged.

Threshold Value #6 covers site-specific phenomena of a hurricane. The Threshold Value is based on damage attributable to the wind.

HA2

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

FIRE **OR** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Threshold Value: (1)

1. FIRE **OR** EXPLOSION **AND** affected system parameter indications show degraded performance **OR** plant personnel report VISIBLE DAMAGE to permanent structures **OR** safety related equipment in any of the following VITAL AREAs.

:

Containment	Auxiliary Building
Service Water Intake Structure (SWIS)	Service Water Pond
Storage Pond Dam and Dike	Pond Spillway Structure
Refueling Water Storage Tank (RWST)	Condensate Storage Tank (CST)
Diesel Generator Building	Control Room

Basis:

FIRE: is combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

VITAL AREA: any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

The areas listed contain functions and systems required for the safe shutdown of the plant to determine if the FIRE or EXPLOSION is potentially affecting any redundant trains of safety systems.

This Threshold Value addresses a FIRE / EXPLOSION and not the degradation in performance of affected systems. The reference to damage of systems is used to identify the magnitude of the FIRE / EXPLOSION and to discriminate against minor FIREs / EXPLOSIONs. The significance here is not that a safety system was degraded but the fact that the FIRE / EXPLOSION was large enough to cause damage to these systems. Thus, the designation of a single train was intentional and is appropriate when the FIRE / EXPLOSION is large enough to affect more than one component.

The inclusion of a "report of VISIBLE DAMAGE" should not be interpreted as mandating a lengthy damage assessment prior to classification. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency Director also needs to consider any security aspects of the EXPLOSIONs, if applicable.

HA3

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Release of Toxic, Asphyxiant or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2)

1. Report **OR** detection of toxic **OR** asphyxiant gas within **OR** contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

2. Report **OR** detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within **OR** contiguous to a VITAL AREA.

Basis:

VITAL AREA: any area, normally within the PROTECTED AREA, which contains equipment, systems, components, or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation.

IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH): A condition that either poses an immediate threat to life and health or an immediate threat of severe exposure to contaminants which are likely to have adverse delayed effects on health.

LOWER FLAMMABILITY LIMIT (LFL): The minimum concentration of a combustible substance that is capable of propagating a flame through a homogenous mixture of the combustible and a gaseous oxidizer.

CONTIGUOUS: being in actual contact: touching along a boundary or at a point.

This IC is based on gases that affect the safe operation of the plant. This IC applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas (i.e., service water pump house). The intent of this IC is not to include buildings (e.g., warehouses) or other areas that are not contiguous or immediately adjacent to plant VITAL AREAs.

Threshold Value #1 is met if measurement of toxic gas concentration results in an atmosphere that is IDLH within a VITAL AREA or any area or building contiguous to VITAL AREA. Exposure to an IDLH atmosphere will result in immediate harm to unprotected personnel, and would preclude access to any such affected areas.

Threshold Value #2 is met when the flammable gas concentration in a VITAL AREA or any building or area contiguous to a VITAL AREA exceed the LOWER FLAMMABILITY LIMIT. This Threshold Value addresses concentrations at which gases can ignite/support combustion. An uncontrolled release of flammable gasses within a facility structure has the potential to affect safe operation of the plant by limiting either operator or equipment operations due to the potential for ignition and resulting equipment damage/personnel injury. Once it has been determined that an uncontrolled release is occurring, then sampling must be done to determine if the concentration of the released gas is within this range.

In accordance with NMP-FLS-005, Confined Space Procedure, examples of hazardous atmospheres include but are not limited to the following:

- Oxygen concentration less than 19.5% or greater than 23.5%
- Flammable gas concentration at greater than 10% of the Lower Flammable Limit (LFL) or Lower Explosive Limit (LEL)
- Air contaminate concentration in excess of the permissible Exposure Limits (PEL) as published by the Occupational Safety and Health Administration (OSHA) or a Threshold Limit Value (TLV) published by the American Conference of Governmental Industrial Hygienist (ACGIH)
- Any other atmospheric condition that is immediately dangerous to life and health

HA4

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the security shift supervision.

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

These EALs address the contingency for a very rapid progression of events, such as that experienced on September 11, 2001. They are not premised solely on the potential for a radiological release. Rather the issue includes the need for rapid assistance due to the possibility for significant and indeterminate damage from additional air, land or water attack elements.

The fact that the site is under serious attack or is an identified attack target with minimal time available for further preparation or additional assistance to arrive requires a heightened state of readiness and implementation of protective measures that can be effective (such as on-site evacuation, dispersal, or sheltering).

Threshold 1 addresses the potential for a very rapid progression of events due to a HOSTILE ACTION. It is not intended to address incidents that are accidental events or acts of civil disobedience, such as small aircraft impact, hunters, or physical disputes between employees within the OCA. Those events are adequately addressed by other EALs.

Threshold 2 addresses the immediacy of an expected threat arrival or impact on the site within a relatively short time.

The intent of this EAL is to ensure that notifications for the airliner attack threat are made in a timely manner and that Offsite Response Organizations (ORO) and plant personnel are at a state of heightened awareness regarding the credible threat. Airliner is meant to be a large aircraft with the potential for causing significant damage to the plant.

This EAL is met when a plant receives information regarding an airliner attack threat from NRC and the airliner is within 30 minutes of the plant. Only the plant to which the specific threat is made need declare the Alert.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

HA5

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Control Room Evacuation Has Been Initiated.

Operating Mode Applicability: All

Threshold Value: (1)

1. Entry into AOP-28.0, Control Room Inaccessibility, for Control Room evacuation.

Basis:

With the control room evacuated, additional support, monitoring and direction through the Technical Support Center and/or other emergency response facility is necessary. Inability to establish plant control from outside the control room will escalate this event to a Site Area Emergency.

HA6

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert.

Operating Mode Applicability: All

Threshold Value: (1)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which involve actual **OR** likely potential substantial degradation of the level of safety of the plant **OR** a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Basis:

HOSTILE ACTION - An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take hostages, and /or intimidate the licensee to achieve an end. This includes attack by air, land, or water using weapons, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based Threshold Values should be used to address such activities, (e.g., violent acts between individuals in the owner controlled area).

This Threshold Value is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the Alert emergency class.

HU1

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Natural and Destructive Phenomena Affecting the PROTECTED AREA.

Operating Mode Applicability: All

Threshold Value: (1 **OR** 2 **OR** 3 **OR** 4 **OR** 5 **OR** 6 **OR** 7)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5.

2. Report by plant personnel of tornado **OR** high winds greater than 115 mph striking within the PROTECTED AREA boundary.

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structures **OR** equipment.

5. Report of turbine failure resulting in casing penetration **OR** damage to turbine **OR** generator seals.

6. Uncontrolled flooding in the following areas:

Service Water Intake Structure (SWIS)
Auxiliary Building
Turbine Building Basement

7. Sustained hurricane force winds greater than 74 mph forecast to be at the plant site in the next four hours in accordance with FNP-0-AOP-21.0.

Basis:

PROTECTED AREA: the area which normally encompasses all controlled areas within the security protected area fence.

EXPLOSION: is a rapid, violent, unconfined combustion, or catastrophic failure of pressurized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components.

VISIBLE DAMAGE: is damage to equipment or structure that is readily observable without measurements, testing, or analysis. Damage is sufficient to cause concern regarding the continued operability or reliability of affected safety structure, system, or component. Example damage includes: deformation due to heat or impact, denting, penetration, rupture, cracking, paint blistering. Surface blemishes (e.g., paint chipping, scratches) should not be included.

These ICs are categorized on the basis of the occurrence of an event of sufficient magnitude to be of concern to plant operators. Areas identified in the Threshold Values define the location of the event based on the potential for damage to equipment contained therein.

Threshold Value #1 - As defined in the EPRI-sponsored "Guidelines for Nuclear Plant Response to an Earthquake", dated October 1989, a *"felt earthquake"* is:

An earthquake of sufficient intensity such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic switches of the plant are activated. For most plants with seismic instrumentation, the seismic switches are set at an acceleration of about 0.01g.

Threshold Value #2 is based on the assumption that a tornado striking (touching down) or high winds within the PROTECTED AREA may have potentially damaged plant structures containing functions or systems required for safe shutdown of the plant. The high wind 115 mph value is based on FSAR design basis. If such damage is confirmed visually or by other in-plant indications, the event may be escalated to Alert.

Threshold Value #3 addresses crashes of vehicle types large enough to cause significant damage to plant structures containing functions and systems required for safe shutdown of the plant.

For Threshold Value #4 only those EXPLOSIONs of sufficient force to damage permanent structures or equipment within the PROTECTED AREA should be considered. No attempt is made in this Threshold Value to assess the actual magnitude of the damage. The occurrence of the EXPLOSION with reports of evidence of damage is sufficient for declaration. The Emergency director also needs to consider any security aspects of the EXPLOSION, if applicable.

Threshold Value #5 addresses main turbine rotating component failures of sufficient magnitude to cause observable damage to the turbine casing or to the seals of the turbine generator. Of major concern is the potential for leakage of combustible fluids (lubricating oils) and gases (hydrogen cooling) to the plant environs. Generator seal damage observed after generator purge does not meet the intent of this Threshold Value because it did not impact normal operation of the plant. This Threshold Value is consistent with the definition of a NOUE while maintaining the anticipatory nature desired and recognizing the risk to non-safety related equipment.

Threshold Value #6 addresses the effect of flooding caused by internal events such as component failures, equipment misalignment, or outage activity mishaps. The areas included are those areas that contain systems required for safe shutdown of the plant that are not designed to be wetted or submerged. TS-PSA-001, Table 3.3.6-4 provides that the SWIS is vulnerable due to failure of the cooling water lines or discharge expansion joints. The Auxiliary Building is vulnerable due to failure of the Service Water piping. The Turbine Building basement is vulnerable to Circulating Water line breaks causing loss of Service Air. Escalation of the emergency classification is based on the damage caused or by access restrictions that prevent necessary plant operations or systems monitoring.

Threshold Value #7 covers site-specific phenomena of the hurricane based on the severe weather mitigation procedure.

HU2

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

Threshold Value: (1)

1. FIRE in buildings **OR** areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification **OR** control room alarm unless disproved by personnel observation within 15 minutes of the alarm:

Containment
Service Water Intake Structure (SWIS)
Auxiliary Building
Refueling Water Storage Tank (RWST)
Diesel Generator Building
Condensate Storage Tank (CST)
Control Room

Basis:

FIRE: is combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

CONTIGUOUS: being in actual contact: touching along a boundary or at a point.

PROTECTED AREA: the area which normally encompasses all controlled areas within the security protected area fence.

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, *Detection* is visual observation and report by plant personnel or sensor alarm indication.

The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room or other nearby site-specific location to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene. In other words, a personnel report from the scene may be used to disprove a sensor alarm if received within 15 minutes of the alarm, but shall not be required to verify the alarm.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished (e.g., smoldering waste paper basket). The site-specific list should be limited and applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas.

HU3

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Release of Toxic, Asphyxiant, or Flammable Gases Deemed Detrimental to Normal Operation of the Plant.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2)

1. Report **OR** detection of toxic, asphyxiant, **OR** flammable gas that has **OR** could enter the Owner Controlled Area in amounts greater than life threatening **OR** flammable concentrations that can affect NORMAL PLANT OPERATIONS.

2. Report by Local, County, **OR** State Officials for evacuation **OR** sheltering of site personnel based on an offsite toxic, asphyxiant, **OR** flammable gas event.

Basis:

NORMAL PLANT OPERATIONS: activities at the plant site associated with routine testing, maintenance, or equipment operations, in accordance with normal operating or administrative procedures. Entry into abnormal or emergency operating procedures, or deviation from normal security or radiological controls posture, is a departure from NORMAL PLANT OPERATIONs.

This IC is based on the existence of uncontrolled releases of toxic, asphyxiant, or flammable gas that may enter the Owner Controlled Area and affect normal plant operations. It is intended that releases of toxic or flammable gases are of sufficient quantity, and the release point of such gases is such that normal plant operations would be affected. Offsite events are included through a warning by local officials as the resultant affect on NORMAL PLANT OPERATIONS would be the same. This would preclude small or incidental releases, or releases that do not impact structures needed for plant operation. The Threshold Values are not intended to require significant assessment or quantification. The IC assumes an uncontrolled process that has the potential to affect plant operations, or personnel safety.

In accordance with NMP-FLS-005, Confined Space Procedure, examples of hazardous atmospheres include but are not limited to the following:

- Oxygen concentration less than 19.5% or greater than 23.5%
- Flammable gas concentration at greater than 10% of the Lower Flammable Limit (LFL) or Lower Explosive Limit (LEL)
- Air contaminate concentration in excess of the permissible Exposure Limits (PEL) as published by the Occupational Safety and Health Administration (OSHA) or a Threshold Limit Value (TLV) published by the American Conference of Governmental Industrial Hygienist (ACGIH)
- Any other atmospheric condition that is immediately dangerous to life and health

HU4

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant.

Operating Mode Applicability: All

Threshold Values: (1 **OR** 2 **OR** 3)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

2. A CREDIBLE FNP security THREAT notification.

3. A validated notification from NRC providing information of an aircraft threat.

Basis:

Note: Timely and accurate communication between Security Shift Supervision and the Control Room is crucial for the implementation of effective Security EALs.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

CREDIBLE THREAT: A threat is considered credible through use of FNP-0-SP-37, Threat Assessment and Security Force Protection Recommendations.

Security events which do not represent a potential degradation in the level of safety of the plant are reported under 10 CFR 73.71 or in some cases under 10 CFR 50.72. Security events assessed as HOSTILE ACTIONS are classifiable under HA4, HS4 and HG1.

A higher initial classification could be made based upon the nature and timing of the security threat and potential consequences. The licensee shall consider upgrading the emergency response status and emergency classification level in accordance with the site's Safeguards Contingency Plan and Emergency Plan.

In Threshold #1 reference is made to security shift supervision because these individuals are the designated personnel on-site qualified and trained to confirm that a security event is occurring or has occurred. Training on security event classification confirmation is closely controlled due to the strict secrecy controls placed on the plant FNP Safeguards Contingency Plan.

This threshold is based on site specific security plans. Site specific Safeguards Contingency Plans are based on guidance provided by NEI 03-12.

The intent of Threshold Value 2 is to ensure that appropriate notifications for the security threat are made in a timely manner. This includes information of a CREDIBLE THREAT. Only the plant to which the specific threat is made need declare the Notification of an Unusual Event.

The intent of Threshold Value 3 is to ensure that notifications for the security threat are made in a timely manner and that Offsite Response Organizations and plant personnel are at a state of heightened awareness regarding the credible threat. It is not the intent of this EAL to replace existing non-hostile related EALs involving aircraft.

This Threshold Value is met when a plant receives information regarding an aircraft threat from NRC. Validation is performed by calling the NRC or by other approved methods of authentication. Only the plant to which the specific threat is made need declare the Unusual Event.

The NRC Headquarters Operations Officer (HOO) will communicate to the licensee if the threat involves an airliner (airliner is meant to be a large aircraft with the potential for causing significant damage to the plant). The status and size of the plane may be provided by NORAD through the NRC.

Escalation to Alert emergency classification level would be via HA4 would be appropriate if the threat involves an airliner within 30 minutes of the plant.

A higher initial classification could be made based upon the nature and timing of the threat and potential consequences. The Emergency Director shall consider upgrading the emergency response status and emergency classification in accordance with the FNP Safeguards Contingency Plan and Emergency Plan implementing Procedures.

HU5

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE.

Operating Mode Applicability: All

Threshold Value: (1)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which indicate a potential degradation of the level of safety of the plant **OR** indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response **OR** monitoring are expected unless further degradation of safety systems occurs.

Basis:

This Threshold Value is intended to address unanticipated conditions not addressed explicitly elsewhere but that warrant declaration of an emergency because conditions exist which are believed by the Emergency Director to fall under the NOUE emergency class.

From a broad perspective, one area that may warrant Emergency Director judgment is related to likely or actual breakdown of site-specific event mitigating actions. Examples to consider include inadequate emergency response procedures, transient response either unexpected or not understood, failure or unavailability of emergency systems during an accident in excess of that assumed in accident analysis, or insufficient availability of equipment and/or support personnel.

4.5 Category E - ISFSI Events

E-HU1

Initiating Condition (Back to Cold IC p. 108) (Back to Hot IC p. 107)

Damage to a loaded cask CONFINEMENT BOUNDARY.

Operating Mode Applicability: Not applicable

Threshold Value: (1)

1. Damage to a loaded dry fuel storage cask CONFINEMENT BOUNDARY due to natural phenomena events, accident conditions **OR** any condition in the opinion of the Emergency Director that affects **OR** causes a loss of loaded dry fuel storage cask CONFINEMENT BOUNDARY.

Basis:

CONFINEMENT BOUNDARY: is the barrier(s) between areas containing radioactive substances and the environment.

A NOUE in this IC is categorized on the basis of the occurrence of an event of sufficient magnitude that a loaded cask CONFINEMENT BOUNDARY is damaged or violated. This includes classification based on a loaded fuel storage cask CONFINEMENT BOUNDARY loss leading to the degradation of the fuel during storage or posing an operational safety problem with respect to its removal from storage.

Any condition, which, in the judgment of the Emergency Director, is a potential degradation in the level of safety of the ISFSI. Emergency Director judgment is to be based on known conditions and the expected response to mitigating activities within a short time period.

4.6 Category C - Cold Shutdown System Malfunctions

CG1

Initiating Condition (Back to Cold IC p. 108)

Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV.

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values: (1 **AND** 2 **AND** 3)

1. Loss of RPV inventory as indicated by ANY of the following:

Unexplained Containment sump level rise
Unexplained Reactor Coolant Drain Tank (RCDT) level rise
Unexplained Waste Holdup Tank (WHT) level rise

AND

2. RPV Level:

   a. Less than elevation 118' (Top of Active Fuel) for greater than 30 minutes.

   OR

   b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery for greater than 30 minutes as evidenced by ANY of the following:

Incore Seal Table R7 $\geq$ 3 mR/hr
Erratic Source Range Monitor Indication

AND

3. Containment challenged as indicated by ANY of the following:

Explosive mixture inside containment	greater than **OR** equal to 6% H_2
Pressure	greater than **OR** equal to 5 psig **WITH** CONTAINMENT CLOSURE established
	greater than **OR** equal to 54 psig **WITH** Tech Spec containment integrity intact
CONTAINMENT CLOSURE **NOT** established	

Basis:

CONTAINMENT CLOSURE: per FNP-1-STP-18.4, “Containment Integrity Verification and Closure”.

For Threshold Value 1 in the cold shutdown mode, normal RCS level and RPV level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

For Threshold Value 1 in the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed to assure that the ability to monitor level will not be interrupted. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. For both cold shutdown and refueling modes sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Threshold Value 2 represents the inability to restore and maintain RPV level to above the top of active fuel. Fuel damage is probable if RPV level cannot be restored, as available decay heat will cause boiling, further reducing the RPV level.

Analysis in appropriate references indicates that core damage may occur within an hour following continued core uncovery therefore, conservatively, 30 minutes was chosen.

The GE is declared on the occurrence of the loss or imminent loss of function of <u>all three</u> barriers. Based on the above discussion, RCS barrier failure resulting in core uncovery for 30 minutes or more may cause fuel clad failure. With the CONTAINMENT breached or challenged then the potential for unmonitored fission product release to the environment is high. This represents a direct path for radioactive inventory to be released to the environment. This is consistent with the definition of a GE.

In the context of Threshold Value 3, CONTAINMENT CLOSURE is the action taken to secure containment and its associated structures, systems, and components as a functional barrier to fission product release under existing plant conditions. If the closure is re-established prior to exceeding the temperature or level thresholds of the RCS Barrier and Fuel Clad Barrier Threshold Values, escalation to GE would not occur.

The site-specific pressure at which CONTAINMENT is considered challenged may change based on the condition of the CONTAINMENT. If the Unit is in the cold shutdown mode and the CONTAINMENT is fully intact then the site-specific setpoint is the CONTAINMENT design pressure (54 psig). This is consistent with typical owner’s groups Emergency Response Procedures. With CONTAINMENT CLOSURE established intentionally by the plant staff in preparations for inspection, maintenance, or refueling the setpoint is based on the penetration seals design of 5 psig.

In the early stages of a core uncovery event, it is unlikely that hydrogen buildup due to a core uncovery could result in an explosive mixture of dissolved gasses in CONTAINMENT. However, CONTAINMENT monitoring and/or sampling should be performed to verify this assumption and a General Emergency declared if it is determined that an explosive mixture exists.

CS1

Initiating Condition (Back to Cold IC p. 108)

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability.

Operating Mode Applicability: Cold Shutdown (Mode 5)

Threshold Values: (1 **OR** 2)

1. Loss of Reactor Pressure Vessel (RPV) inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE **NOT** established as indicated by:

   a. RPV level less than 121' (6" below Bottom ID of RCS loop).

   OR

   b. RPV level **CANNOT** be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by unexplained level rise in any of the following:

Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

2. Loss of RPV inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE established as indicated by:

   a. RPV level less than 118' (Top of Active Fuel (TOAF)).

   OR

   b. RPV level **CANNOT** be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by ANY of the following:

Containment sump level rise
Unexplained Reactor Coolant Drain Tank (RCDT) level rise
Unexplained Waste Holdup Tank (WHT) level rise
Erratic Source Range monitor indication

Basis:

CONTAINMENT CLOSURE: per FNP-1-STP-18.4, Containment Integrity Verification and Closure".

Under the conditions specified by this IC, continued lowering in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach, pressure boundary leakage, or continued boiling in the RPV.

In the cold shutdown mode, normal RCS level and reactor vessel level indication systems (RVLIS) will normally be available. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

The 30 minute duration allows sufficient time for actions to be performed to recover needed cooling equipment. The effluent release path is not expected with closure established.

CS2

Initiating Condition (Back to Cold IC p. 108)

Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV.

Operating Mode Applicability: Refueling (Mode 6)

Threshold Values: (1 **OR** 2)

1. **WITH** CONTAINMENT CLOSURE **NOT** established:

   a. RPV level less than elevation 121' (6" below Bottom ID of RCS loop).

   OR

   b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery as evidenced by ANY of the following:

Incore Seal Table R7 $\geq$ 3 mR/hr
Erratic Source Range Monitor Indication

2. **WITH** CONTAINMENT CLOSURE established

   a. RPV level less than elevation 118' (Top of Active Fuel).

   OR

   b. RPV level **CANNOT** be monitored **WITH** Indication of core uncovery as evidenced by ANY of the following:

Containment High Range Radiation Monitor R27 A **OR** B $\geq$100 R/hr
Incore Seal Table R7 $\geq$ 3 mR/hr
Erratic Source Range Monitor Indication

Basis:

CONTAINMENT CLOSURE: per FNP-1-STP-18.4, Containment Integrity Verification and Closure".

Under the conditions specified by this IC, continued lowering in RPV level is indicative of a loss of inventory control. Inventory loss may be due to an RPV breach or continued boiling in the RPV.

As water level in the RPV lowers, the dose rate above the core will rise. The dose rate due to this core shine should result in up-scaled Containment High Range Monitor indication and possible alarm. Additionally, post-TMI studies indicated that the installed nuclear instrumentation will operate erratically when the core is uncovered and that this should be used as a tool for making such determinations.

For Threshold Value 2 in the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed (including the ability to monitor level visually) to assure that the ability to monitor level will not be interrupted.

This effluent release is not expected with closure established.

CA1

Initiating Condition (Back to Cold IC p. 108)

Loss of RCS Inventory.

Operating Mode Applicability: Cold Shutdown (Mode 5)

Threshold Values: (1 **OR** 2)

1. Loss of RCS inventory as indicated by RPV level less than 121’ 6” (bottom ID of RCS loop)

2. a. RCS level **CANNOT** be monitored for greater than 15 minutes

   AND

   b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in ANY of the following:

Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

Basis:

These Threshold Values serve as precursors to a loss of ability to adequately cool the fuel. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level lowering and potential core uncovery.

The Bottom ID of the RCS Loop Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems has occurred. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In the cold shutdown mode, normal RCS level and RPV level instrumentation systems will normally be available. However, if all level indication were to be lost during a loss of RCS inventory event, the operators would need to determine that RPV inventory loss was occurring by observing sump and tank level changes. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration for the loss of level indication was chosen because it is half of the CS1 Site Area Emergency Threshold Value duration.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

CA2

Initiating Condition (Back to Cold IC p. 108)

Loss of RPV Inventory with Irradiated Fuel in the RPV.

Operating Mode Applicability: Refueling (Mode 6)

Threshold Values: (1 **OR** 2)

1. Loss of inventory as indicated by RPV level less than 121' 6" (bottom ID of RCS loop)

2. a. RPV level **CANNOT** be monitored for greater than 15 minutes

   AND

   b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in ANY of the following:

Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

Basis:

These Threshold Values serve as precursors to a loss of heat removal. The magnitude of this loss of water indicates that makeup systems have not been effective and may not be capable of preventing further RPV level lowering and potential core uncovery. This condition will result in a minimum classification of Alert.

The Bottom ID of the RCS Loop Setpoint was chosen because at this level remote RCS level indication may be lost and loss of suction to decay heat removal systems may occur. The inability to restore and maintain level after reaching this setpoint would therefore be indicative of a failure of the RCS barrier.

In the refueling mode, normal means of RPV level indication may not be available. Redundant means of RPV level indication will be normally installed to assure that the ability to monitor level will not be interrupted. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage. The 15-minute duration allows CA2 to be an effective precursor to CS2.

The difference between CA1 and CA2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

CA3

Initiating Condition (Back to Cold IC p. 108)

Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses.

Operating Mode Applicability:

Cold Shutdown (Mode 5)
Refueling (Mode 6)
Defueled

Threshold Value:

1. a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G

   AND

   b. Failure of emergency diesel generators to supply power to emergency busses.

   AND

   c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

Basis:

Loss of all AC power compromises all plant safety systems requiring electric power including RHR, ECCS, Containment Heat Removal, Spent Fuel Heat Removal and the Ultimate Heat Sink. When in cold shutdown, refueling, or defueled mode the event can be classified as an Alert, because of the significantly reduced decay heat, lower temperature and pressure, increasing the time to restore one of the emergency busses. Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

Consideration should be given to operable loads necessary to remove decay heat or provide Reactor Vessel makeup capability when evaluating loss of AC power to essential busses. Even though an essential bus may be energized, if necessary loads (i.e., loads that if lost would inhibit decay heat removal capability or Reactor Vessel makeup capability) are not operable on the energized bus then the bus should not be considered operable.

CA4

Initiating Condition (Back to Cold IC p. 108)

Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV.

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values: (1 **OR** 2 **OR** 3)

1. An UNPLANNED event results in RCS temperature exceeding 200°F with:
   a. CONTAINMENT CLOSURE **NOT** established.

      AND

   b. RCS integrity **NOT** established

NOTE 1	**The Emergency Director should not wait until the indicated time of Threshold Values 2 or 3 has elapsed, but should declare the event as soon as it is determined that the duration has or will likely be exceeded.**
NOTE 2	**If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then Threshold Values 2 and 3 are not applicable**

2. An UNPLANNED event results in RCS temperature exceeding 200°F for greater than 20 minutes (Note) with:

   a. CONTAINMENT CLOSURE established.

      AND

   b. RCS integrity **NOT** established.

      OR

   c. RCS inventory reduced.

3. An UNPLANNED event results in:

   a. RCS temperature exceeding 200°F for greater than 60 minutes (Note).

      OR

   b. RCS pressure increasing greater than 10 psig.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

CONTAINMENT CLOSURE: per FNP-1-STP-18.4, “Containment Integrity Verification and Closure”.

Threshold Value 1 addresses complete loss of functions required for core cooling during refueling and cold shutdown modes when neither CONTAINMENT CLOSURE nor RCS integrity are established. RCS integrity is in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). No delay time is allowed for Threshold Value1 because the evaporated reactor coolant that may be released into the Containment during this heatup condition could also be directly released to the environment.

Threshold Value 2 addresses the complete loss of functions required for core cooling for greater than 20 minutes during refueling and cold shutdown modes when CONTAINMENT CLOSURE is established but RCS integrity is not established or RCS inventory is reduced. As in Threshold Value 1, RCS integrity should be assumed to be in place when the RCS pressure boundary is in its normal condition for the cold shutdown mode of operation (e.g., no freeze seals or nozzle dams). The allowed 20 minute time frame was included to allow operator action to restore the heat removal function, if possible.

Threshold Value 3 addresses complete loss of functions required for core cooling for greater than 60 minutes during refueling and cold shutdown modes when RCS integrity is established. The status of CONTAINMENT CLOSURE in this Threshold Value is immaterial given that the RCS is providing a high pressure barrier to fission product release to the environment. The 60 minute time frame should allow sufficient time to restore cooling without there being a substantial degradation in plant safety. The 10 psig pressure rise covers situations where, due to high decay heat loads, the time provided to restore temperature control, should be less than 60 minutes.

The Emergency Director must remain alert to events or conditions that lead to the conclusion that exceeding the Threshold Value is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

CU1

Initiating Condition (Back to Cold IC p. 108)

RCS Leakage.

Operating Mode Applicability: Cold Shutdown (Mode 5)

Threshold Values:

1. Unable to establish or maintain pressurizer level greater than 15%.

Basis:

This IC is included as a NOUE because it is considered to be a potential degradation of the level of safety of the plant. The inability to establish and maintain level is indicative of loss of RCS inventory. Prolonged loss of RCS Inventory may result in escalation to the Alert level via either IC CA1 (Loss of RCS) or CA4 (Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV).

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling mode applicability. In cold shutdown the RCS will normally be intact and RCS inventory and level monitoring means such as Pressurizer level indication and makeup volume control tank levels are normally available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

CU2

Initiating Condition (Back to Cold IC p. 108)

UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the RPV.

Operating Mode Applicability: Refueling (Mode 6)

Threshold Values: (1 **OR** 2)

1. UNPLANNED RCS level lowering below 129' (RPV flange) for greater than **OR** equal to 15 minutes.

2. a. RPV level **CANNOT** be monitored.

   AND

   b. A possible loss of RPV inventory may be occurring as indicated by unexplained level rise in ANY of the following:

Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as result, is considered to be a potential degradation of the level of safety of the plant. Refueling evolutions that lower RCS water level below the RPV flange are carefully planned and procedurally controlled. An UNPLANNED event that results in water level decreasing below the RPV flange warrants declaration of a NOUE due to the reduced RCS inventory that is available to keep the core covered. The allowance of 15 minutes was chosen because it is reasonable to assume that level can be restored within this time frame using any of the redundant means of refill that should be available.

The difference between CU1 and CU2 deals with the RCS conditions that exist between cold shutdown and refueling modes. In cold shutdown the RCS will normally be intact and standard RCS inventory and level monitoring means are available. In the refueling mode the RCS is not intact and RPV level and inventory are monitored by different means.

In the refueling mode, normal means of core temperature indication and RCS level indication may not be available. Redundant means of RPV level indication will normally be installed to assure that the ability to monitor level will not be interrupted. Sump and tank level rises must be evaluated against other potential sources of leakage such as cooling water sources inside the containment to ensure they are indicative of RCS leakage.

Threshold Value 1 involves a lowering in RCS level below the top of the RPV flange that continues for 15 minutes due to an UNPLANNED event.

CU3

Initiating Condition (Back to Cold IC p. 108)

Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes.

Operating Mode Applicability:

Cold Shutdown (Mode 5)
Refueling (Mode 6)

NOTE: A NOUE should not be made for pre-planned testing such as SI/LOSP testing.

Threshold Value:

1. a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G for greater than 15 minutes.

   AND

   b. At least one emergency diesel generator supplying power to **EITHER** 4160V ESF buss 1(2)F **OR** 1(2)G.

Basis:

Prolonged loss of AC power reduces required redundancy and potentially degrades the level of safety of the plant by rendering the plant more vulnerable to a complete Loss of AC Power (e.g., Station Blackout). Fifteen minutes was selected as a threshold to exclude transient or momentary power losses.

CU4

Initiating Condition (Back to Cold IC p. 108)

UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV.

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values: (1 **OR** 2)

NOTE: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed the Threshold Value.

1. An UNPLANNED event results in RCS temperature exceeding 200°F.

2. Loss of all RCS temperature **AND** RPV level indication for greater than 15 minutes.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC is included as a NOUE because it may be a precursor of more serious conditions and, as a result, is considered to be a potential degradation of the level of safety of the plant. In cold shutdown the ability to remove decay heat relies primarily on forced cooling flow. Operation of the systems that provide this forced cooling may be jeopardized due to the unlikely loss of electrical power or RCS inventory. Since the RCS usually remains intact in the cold shutdown mode a large inventory of water is available to keep the core covered. In cold shutdown the decay heat available to raise RCS temperature during a loss of inventory or heat removal event may be significantly greater than in the refueling mode.

During refueling the level in the RPV will normally be maintained above the RPV flange. Refueling evolutions that lower water level below the RPV flange are carefully planned and procedurally controlled. Loss of forced decay heat removal at reduced inventory may result in more rapid rises in RCS/RPV temperatures depending on the time since shutdown.

Unlike the cold shutdown mode, normal means of core temperature indication and RCS level indication may not be available in the refueling mode. Redundant means of RPV level indication are therefore procedurally installed to assure that the ability to monitor level will not be interrupted. However, if all level and temperature indication were to be lost in either the cold shutdown of refueling modes, Threshold Value 2 would result in declaration of a NOUE if either temperature or level indication cannot be restored within 15 minutes from the loss of both means of indication.

The Emergency Director must remain attentive to events or conditions that lead to the conclusion that exceeding the Threshold Value is imminent. If, in the judgment of the Emergency Director, an imminent situation is at hand, the classification should be made as if the threshold has been exceeded.

CU6

Initiating Condition (Back to Cold IC p. 108)

UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities.

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values: (1 **OR** 2)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:

In plant telephones
Public address system
Plant radio systems

2. UNPLANNED loss of ALL of the following off-site communications capability:

ENN (Emergency Notification Network)
ENS (Emergency Notification System)
Commercial phones (Radio, PBX, Satellite, Wireless)
VOIP (Voice Over Internet Protocol)
OPX (Off Premise Extension)

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

The purpose of this IC and its associated Threshold Values is to recognize a loss of communications capability that either defeats the plant operations staff ability to perform routine tasks necessary for plant operations or the ability to communicate problems with offsite authorities.

The availability of one method of ordinary offsite communications is sufficient to inform state and local authorities of plant problems. This Threshold Value is intended to be used only when extraordinary means (e.g., relaying of information from radio transmissions, individuals being sent to offsite locations, etc.) are being utilized to make communications possible.

CU7

Initiating Condition (Back to Cold IC p. 108)

UNPLANNED Loss of Required DC Power for Greater than 15 Minutes.

Operating Mode Applicability:

Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values:

(1a **AND** 1b)

1. a. UNPLANNED loss of Vital DC power to 125 VDC Bus A **AND** B indicated by bus voltage indications less than 105 VDC.

   AND

   b. Failure to restore power to at least one DC bus within 15 minutes from the time of loss.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

The purpose of this IC and its associated Threshold Values is to recognize a loss of DC power compromising the ability to monitor and control the removal of decay heat during Cold Shutdown or Refueling operations. This Threshold Value is intended to be anticipatory in as much as the operating crew may not have necessary indication and control of equipment needed to respond to the loss.

105 VDC bus voltage is based on the minimum bus voltage necessary for the operation of safety related equipment. This voltage value incorporates a margin of at least 15 minutes of operation before the onset of inability to operate those loads.

CU8

Initiating Condition (Back to Cold IC p.108)

Inadvertent Criticality.

Operating Mode Applicability: Cold Shutdown (Mode 5)
Refueling (Mode 6)

Threshold Values: (1)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

Basis:

UNPLANNED: a parameter change or an event that is not the result of an intended evolution and requires corrective or mitigative actions.

This IC addresses criticality events that occur in Cold Shutdown or Refueling modes such as fuel mis-loading events and inadvertent dilution events. This IC indicates a potential degradation of the level of safety of the plant, warranting a NOUE classification.

The term "sustained" is used in order to allow exclusion of expected short term positive startup rates from planned fuel bundle or control rod movements during core alterations. These short term positive startup rates are the result of the rise in neutron population due to subcritical multiplication.

Figure 1

	FARLEY NUCLEAR PLANT **Figure 1 – Fission Product Barrier Evaluation**		NMP-EP-110- GL01 Rev 8.0
General Emergency	**Site Area Emergency**	**Alert**	**Unusual Event**
FG1 Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier	FS1 Loss or Potential Loss of ANY Two Barriers	FA1 ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS	FU1 ANY Loss or ANY Potential Loss of Containment

Fuel Clad Barrier

Loss	Potential Loss
1. Critical Safety Function Status (p.33) Core-Cooling RED	**1. Critical Safety Function Status** (p.33) Core Cooling-ORANGE **OR** Heat Sink-RED
2. Primary Coolant Activity Level (p. 33) Indications of RCS Coolant Activity greater than 300 µCi/gm Dose Equivalent I-131	2. Primary Coolant Activity Level(p. 33) Not Applicable
3. Core Exit Thermocouple Readings (p. 33) 5th Hottest CETC greater than 1200°F	**3. Core Exit Thermocouple Readings** (p. 33) 5th Hottest CETC greater than 700°F
4. Reactor Vessel Water Level (p. 33) Not Applicable	**4. Reactor Vessel Water Level** (p. 33) RVLS Plenum LEVEL less than 0%
5. Containment Radiation Monitoring (p. 33) Containment Radiation Monitor RE-27 A **OR** B greater than 600 R/hr	5. Containment Radiation Monitoring (p. 33) Not Applicable
6. Other Indications (p.34) Not applicable	6. Other Indications (p. 34) Not applicable
7. Emergency Director Judgment (p. 34) Judgment by the ED that the Fuel Clad Barrier is lost. Consider conditions not addressed and inability to determine the status of the Fuel Clad Barrier	**7. Emergency Director Judgment** (p. 34) Judgment by the ED that the Fuel Clad Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the Fuel Clad Barrier.

RCS Barrier

Loss	Potential Loss
1. Critical Safety Function Status (p. 34) Not Applicable	**1. Critical Safety Function Status** (p. 34) RCS Integrity-RED **OR** Heat Sink-RED
2. RCS Leak Rate (p. 34) RCS subcooling less than 16°F {less than 45° F Adverse} due to an RCS leak greater than Charging / RHR capacity	**2. RCS Leak Rate**(p. 35) Non-isolable RCS leak (including SG tube Leakage) greater than120 GPM.
3. SG Tube Rupture (p. 35) EEP-3.0 entered due to SG tube rupture resulting in an ECCS actuation	3. SG Tube Rupture (p. 35) Not Applicable
4. Containment Radiation Monitoring (p. 35) CTMT Rad Monitor RE-2 greater than 1.0 R/hr OR CTMT Radiation Monitor RE-7 greater than 500 mR/hr	4. Containment Radiation Monitoring (p. 35) Not Applicable
5. Other Indications (p. 35) Not applicable	**5. Other Indications** (p. 35) Unexplained level rise in ANY of the following: Containment sump Reactor Coolant Drain Tank (RCDT) Waste Holdup Tank (WHT)
6. Emergency Director Judgment (p. 35) Judgment by the ED that the RCS Barrier is lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier	**6. Emergency Director Judgment**(p. 35) Judgment by the ED that the RCS Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier.

Containment Barrier

Loss	Potential Loss
1. Critical Safety Function Status (p. 36) Not Applicable	**1. Critical Safety Function Status** (p. 36) Containment-RED
2. Containment Pressure (p. 36) Rapid unexplained CTMT pressure lowering following initial pressure rise **OR** Intersystem LOCA indicated by CTMT pressure or sump level response not consistent with a loss of primary or secondary coolant	2. Containment Pressure (p. 36) CTMT pressure greater than 54 psig and rising **OR** CTMT hydrogen concentration greater than 6% **OR** CTMT CSF - ORANGE **AND** Less than the following minimum operable equipment: One CTMT fan cooler **AND** One train of CTMT spray
3. Core Exit Thermocouple Reading (p. 36) Not applicable	**3. Core Exit Thermocouple Reading** (p. 36) CORE COOLING CSF - RED **OR** - ORANGE for greater than 15min **AND** RVLS LEVEL less than 0%
4. SG Secondary Side Release with Primary to Secondary Leakage (p. 37) RUPTURED S/G is also FAULTED outside of containment **OR** Primary-to-Secondary leakrate greater than 10 gpm with nonisolable steam release from affected S/G to the environment	4. SG Secondary Side Release with P-to-S Leakage (p. 37) Not applicable
5. CTMT Isolation Valves Status After CTMT Isolation (p. 37) CTMT isolation valves **OR** dampers **NOT** closed **AND** downstream pathway to the environment exists after Containment Isolation	5. CTMT Isolation Valves Status After CTMT Isolation (p. 37) Not Applicable
6. Significant Radioactive Inventory in Containment (p. 37) Not Applicable	**6. Significant Radioactive Inventory in Containment** (p. 37) CTMT Rad monitor RE-27 A OR B greater than 8000 R/hr
7. Other Indications (p. 38) Pathway to the environment exists based on VALID RE-10, RE-14, RE-21, **OR** RE-22 Alarms	7. Other Indications (p. 38) Not applicable
8. Emergency Director Judgment (p. 38) Judgment by the ED that the CTMT Barrier is lost. Consider conditions not addressed and inability to determine the status of the CTMT Barrier	**8. Emergency Director Judgment** (p. 38) Judgment by the ED that the CTMT Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the CTMT Barrier

Figure 2 - Hot Initiating Condition Matrix

Farley Nuclear Plant

Procedure Number: NMP-EP-110-GL01 Rev 8.0

EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 2 - Hot Initiating Condition Matrix - Modes 1, 2, 3 and 4

Column groups: RADIOLOGICAL - ALL MODES (EFFLUENTS; RAD LEVELS) | FISSION PRODUCT BARRIERS - Modes 1 -2 -3 - 4 | SYSTEM MALFUNCTIONS MODES 1 - 2 - 3 - 4 (AC / DC POWER; RX and CORE; ALARMS / COMMUNICATIONS) | HAZARDS - ALL MODES (NATURAL DESTRUCTIVE; FIRE/EXPLOSION; TOXIC/FLAMMABLE; SECURITY; CR EVACUATION; ED DISCRETION)

ISFSI

GENERAL EMERGENCY

EFFLUENTS

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1,000 mR TEDE OR 5,000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology (pg. 17)

See Notes

1. VALID reading on any of the following radiation monitors that exceeds OR expected to exceed the reading shown for 15 minutes OR longer:

Steam Jet Air Ejector RE-15C	130 µCi/cc (130 R/hr)
Plant Vent Stack RE-29B (NG)	0.9 µCi/cc
Steam Gen Relief RE-60A,B,C	0.5 µCi/cc (0.5 R/hr)
TDAFW Steam Exh RE-60D	11 µCi/cc (11 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 1,000 mR TEDE OR 5,000 mR thyroid CDE at OR beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 1,000 mR/hr expected to continue for more than one hour; OR analyses of field survey samples indicate thyroid CDE of 5,000 mR for one hour of inhalation, at OR beyond site boundary.

RAD LEVELS

Note: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FISSION PRODUCT BARRIERS

FG1 - Loss of ANY Two Barriers AND Loss or Potential Loss of Third Barrier (pg. 32)

See Fission Product Barrier Matrix

AC / DC POWER

SG1 - Prolonged Loss of All Offsite Power AND Prolonged Loss of All Onsite AC Power to Essential Busses (pg. 40)

1. Loss of ALL AC power indicated by:
a. Loss of offsite power to OR from Start Up Transformers 1(2)A AND 1(2)B resulting in loss of all offsite electrical power to BOTH 4160V ESF busses 1(2)F AND 1(2)G for greater than 15 minutes

AND

b. Failure of emergency diesel generators to supply power to emergency busses.

AND EITHER

2. Restoration of at least one 4160V ESF bus, 1(2)F OR 1(2)G, within 4 hr. of time of loss is NOT likely

OR

3. Fuel Clad Barrier Evaluation indicates continuing degradation (Loss or Potential Loss) due to core cooling

RX and CORE

SG2 - Failure of the Reactor Protection System to Complete an Automatic Trip AND Manual Trip was NOT Successful AND there is Indication of an Extreme Challenge to the Ability to Cool the Core (pg. 42)

NOTE: A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.

NOTE: Heat Sink CSF should not be considered RED if total AFW flow is less than 395 gpm due to operator action.

NOTE: Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.

1. Indications exist that a reactor protection system setpoint was exceeded and automatic trip did not occur, and a manual trip did not result in the reactor being made subcritical. AND EITHER
a. Core Cooling CSF - RED
OR
b. Heat Sink CSF - RED

SECURITY

HG1 – HOSTILE ACTION Resulting in Loss Of Physical Control of the Facility (pg. 62)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

ED DISCRETION

HG2 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency (pg. 63)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity OR HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

EFFLUENTS

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE OR 500 mR Thyroid CDE for the Actual or Projected Duration of the Release. (pg. 19)

See Notes

1. VALID reading on any of the following radiation monitors that exceeds OR is expected to exceed the reading shown for 15 minutes OR longer:

Steam Jet Air Ejector RE-15C	13 µCi/cc (13 R/hr)
Plant Vent Stack RE-29B (NG)	0.09 µCi/cc
Steam Gen Relief RE-60A,B,C	0.05 µCi/cc (0.05 R/hr)
TDAFW Steam Exh RE-60D	1.1 µCi/cc (1.1 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE OR 500 mR thyroid CDE at OR beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; OR analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at OR beyond the site boundary.

RAD LEVELS

Note: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

FISSION PRODUCT BARRIERS

FS1 - Loss or Potential Loss of ANY Two Barriers (pg. 32)

See Fission Product Barrier Matrix

AC / DC POWER

SS1 - Loss of All Offsite Power AND Loss of All Onsite AC Power to Essential Busses (pg. 44)

1. Loss of all AC power indicated by:
a. Loss of offsite power to OR from Start Up Transformers 1(2)A AND 1(2)B resulting in loss of all offsite electrical power to BOTH 4160V ESF busses 1(2)F AND 1(2)G for greater than 15 minutes

AND

b. Failure of emergency diesel generators to supply power to emergency busses.

AND

c. Restoration of at least one 4160V ESF bus, 1(2)F OR 1(2)G, has NOT occurred within 15 minutes of time of loss of all AC power

SS3 - Loss of All Vital DC Power (pg. 46)

1. Loss of all Vital DC power to 125 VDC Bus A AND B indicated by bus voltage indications less than 105 VDC for greater than 15 minutes.

RX and CORE

SS2 - Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded AND Manual Trip Was NOT Successful (pg. 45)

NOTE: A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.

NOTE: Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.

1. Indications exist that a reactor protection system setpoint was exceeded and automatic trip did not occur, and a manual trip did not result in the reactor being made subcritical. (NOTE)

SS4 - Complete Loss of Heat Removal Capability (pg. 47)

NOTE: Heat Sink CSF should not be considered RED if total AFW flow is less than 395 gpm due to operator action.

1. Complete Loss of Heat Removal Capability as indicated by:
a. Core Cooling CSF - ORANGE
AND
b. Heat Sink CSF - RED

ALARMS / COMMUNICATIONS

SS6 - Inability to Monitor a SIGNIFICANT TRANSIENT in Progress (pg. 48)

1. a. A SIGNIFICANT TRANSIENT is in progress
AND
b. Loss of most OR all (approximately 75% of the MCB) annunciators OR indicators associated with safety systems
AND
c. Compensatory non-alarming indications are NOT available
AND
d. Indications needed to monitor the Critical Safety Function Status Tree parameters are NOT available

SECURITY

HS4 – HOSTILE ACTION within the PROTECTED AREA (pg. 66)

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the site security force.

CR EVACUATION

HS2 - Control Room Evacuation Has Been Initiated AND Plant Control Cannot Be Established (pg. 64)

1. a. Control Room evacuation has been initiated
AND
b. Control of the plant can NOT be established per AOP-28.0, Control Room Inaccessibility, within 15 minutes.

ED DISCRETION

HS3 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency (pg. 65)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process OR have occurred which involve actual OR likely major failures of plant functions needed for protection of the public OR HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT

EFFLUENTS

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer (pg. 21)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes OR longer.

Liq RW Eff RE-18	200x LWRP setpoint (planned release) $\geq 1 \times 10^6$ cpm (no planned release)
S/G BD Eff Line RE-23B	2.8×10^5 cpm
SJAE RE-15A	3.5×10^5 cpm
Plant Vent Gas RE-14	200x GWRP setpoint (planned release) $\geq 1 \times 10^6$ cpm (no planned release)
Plant Vent Stack RE-22	4.0×10^5 cpm
Plant Vent Stack RE-29B(NG)	8.9×10^{-3} µCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes OR longer:

MS Atmos Relief (R60A,B,C)	1.4×10^1 R/hr
TDAFW Exhaust (R60D)	1.4×10^1 R/hr

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations OR release rates in excess of 200 times Technical Specification 5.5.4.b as confirmed by the ODCM, with a release duration of 15 minutes OR longer.

RAD LEVELS

RA2 - Damage to Irradiated Fuel OR Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel (pg. 23)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
Drumming Station RE-0008
Containment Purge Ventilation Monitor RE-24A OR B
Spent Fuel Pool Ventilation Monitor RE-25A OR B
Spent Fuel Pool Area Radiation Monitor RE-5

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:

Report of personnel during fuel assembly movements.

Spent Fuel Pool Storage	Less than El 129
Transfer Canal	Less than El 116
Reactor Core Elevation	Less than El 118

RA3 - Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown. (pg. 25)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
Control Room radiation monitor RE-1A

2. UNPLANNED VALID radiation readings greater than 1000mR/hr (1 R/hr) values in the following areas requiring infrequent access to maintain plant safety functions.
RadioChemistry Lab RE-3
Electrical and Piping Penetration Rooms

Charging Pump Room Area RE-4	VCT Valve Room
Sample Room Area RE-6	Seal Water HX Room
Lower Equipment Room	CCW HX Room
Main Steam Valve Room	Hot Shutdown Panels
4160 V ESF Bus Rooms	DC Switchgear Rooms
Control Rod Drive Room	Valve Box 1, 2, 3, and 4
Diesel Building	Service Water Intake Structure
Turbine Building Air Compressor Area	

FISSION PRODUCT BARRIERS

EA1 - ANY Loss or ANY Potential Loss of EITHER Fuel Clad OR RCS (pg. 32)

See Fission Product Barrier Matrix

AC / DC POWER

SA5 - AC power capability to essential busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in station blackout. (pg. 53)

1. a. AC power capability to 4160V ESF busses 1(2)F AND 1(2)G reduced to a single power source for greater than 15 minutes

AND

b. ANY additional single failure will result in station blackout.

RX and CORE

SA2 - Failure of Reactor Protection System Instrumentation to Complete or Initiate an Automatic Reactor Trip Once a Reactor Protection System Setpoint Has Been Exceeded AND Manual Trip Was Successful. (pg. 50)

NOTE: A successful manual trip for purposes of declaration is any action taken from the MCB that rapidly inserts the control rods. This can be accomplished by tripping the reactor using the Reactor Trip switches on the MCB OR by de-energizing both Rod Drive Motor Generator sets from the MCB.

NOTE: Failure of both MCB Rx Trip switches to trip the reactor meets the TV criteria of a setpoint being exceeded with no automatic trip occurring.

1. Indication(s) exist that a reactor protection setpoint was exceeded and an automatic trip did not occur, and a manual trip resulted in the reactor being subcritical.

ALARMS / COMMUNICATIONS

SA4 - UNPLANNED Loss of Most or All Safety System Annunciation or Indication in the Control Room With EITHER (1) a SIGNIFICANT TRANSIENT in Progress, OR (2) Compensatory Non-Alarming Indicators are Unavailable. (pg. 51)

1. UNPLANNED loss of most OR all (approximately 75% of the MCB) annunciators OR indicators associated with safety systems for greater than 15 minutes.

AND EITHER

a. A SIGNIFICANT TRANSIENT is in process
OR
b. Compensatory non-alarming indications are NOT available.

NATURAL DESTRUCTIVE

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA (pg. 67)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5 indicates Seismic event greater than Operating Basis Earthquake (OBE).

2. Tornado OR high winds greater than 115 mph within the PROTECTED AREA boundary AND resulting in VISIBLE DAMAGE to any of the following plant structures/equipment OR the Control Room has indication of degraded performance of any listed systems:

Containment	Auxiliary Building
Service Water Pond	Condensate Storage Tank (CST)
Service Water Intake Structure (SWIS)	
Refueling Water Storage Tank (RWST)	
D/G Building	

3. Vehicle crash within PROTECTED AREA boundary AND resulting in VISIBLE DAMAGE to any of the following plant structures OR equipment therein OR Control Room has indication of degraded performance of those systems:

Containment	Auxiliary Building
Service Water Pond	Condensate Storage Tank (CST)
Service Water Intake Structure (SWIS)	
Refueling Water Storage Tank (RWST)	D/G Building

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to OR penetration of any of the following plant areas containing safety-related equipment, their controls OR their power supplies.

Containment	Auxiliary Building
Condensate Storage Tank (CST)	Control Room
Refueling Water Storage Tank (RWST)	D/G Building

5. Uncontrolled flooding in the following areas that results in degraded safety system performance as indicated in the control room OR that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate OR monitor safety equipment.
Service Water Intake Structure (SWIS)
Auxiliary Building Turbine Building Basement

6. Sustained hurricane winds greater than 74 mph onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems

FIRE/EXPLOSION

HA2 - FIRE OR EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (pg. 69)

1. FIRE OR EXPLOSION AND affected system parameter indications show degraded performance OR plant personnel report VISIBLE DAMAGE to permanent structures OR safety related equipment in any of the following VITAL AREAs.
Containment
Auxiliary Building
Control Room
Storage Pond dam and dike
Service Water Pond
Condensate Storage Tank (CST)
Service Water Intake Structure (SWIS)
Refueling Water Storage Tank (RWST)
Diesel Generator Building
Pond Spillway Structure

TOXIC/FLAMMABLE

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown (pg. 71)

1. Report OR detection of toxic OR asphyxiant gas within OR contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

2. Report OR detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within OR contiguous to a VITAL AREA.

SECURITY

HA4 - HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat. (pg. 73)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the security shift supervision.

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

CR EVACUATION

HA5 - Control Room Evacuation Has Been Initiated (pg. 74)

1. Entry into AOP-28.0, Control Room Inaccessibility, for Control Room evacuation.

ED DISCRETION

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert (pg. 75)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process OR have occurred which involve actual OR likely potential substantial degradation of the level of safety of the plant OR a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

UNUSUAL EVENT

EFFLUENTS

RU1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer (pg. 27)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes OR longer.

Liq RW Eff RE-18	2x LWRP setpoint (planned release) 1.6×10^5 cpm (no planned release)
S/G BD Eff Line RE-23B	2.8×10^3 cpm
SJAE RE-15A	3.5×10^3 cpm
Plant Vent Gas RE-14	2x GWRP setpoint (planned release) 3.2×10^4 cpm (no planned release)
Plant Vent Stack RE-22	4.0×10^3 cpm
Plant Vent Stack RE-29B(NG)	8.9×10^{-5} µCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes OR longer:

MS Atmos Relief (R60A,B,C)	1.4×10^{-1} R/hr
TDAFW Exhaust (R60D)	1.4×10^{-1} R/hr

3. Confirmed sample analyses for gaseous OR liquid releases indicates concentrations OR release rates, with a release duration of 60 minutes OR longer, in excess of two times Technical Specification 5.5.4.b, as confirmed by the ODCM.

RAD LEVELS

RU2 - Unexpected Rise in Plant Radiation (pg. 29)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, OR fuel transfer canal with all irradiated fuel assemblies remaining covered by water.
Personnel report of low water level
Annunciator EH2 "SFP LVL HI/LO"
Personnel report of cavitation OR low discharge pressure for SFP Pump Discharge Pressure AND/OR RHR Pump Discharge Pressure

AND

b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
RE-0005 in the fuel building
RE-0002 in containment
RE-0027A OR B in containment

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

FISSION PRODUCT BARRIERS

FU1 - ANY Loss OR ANY Potential Loss of Containment (pg. 32)

See Fission Product Barrier Matrix

AC / DC POWER

SU1 - Loss of All Offsite Power to Essential Busses for greater than 15 Minutes (pg. 54)

1. a. Loss of offsite power to OR from Start Up Transformers 1(2)A AND 1(2)B resulting in loss of all offsite electrical power to BOTH 4160V ESF busses 1(2)F AND 1(2)G for greater than 15 minutes

AND

b. Emergency diesel generators supplying power to BOTH 4160V ESF busses 1(2)F AND 1(2)G.

RX and CORE

SU2 - Inability to Reach Required Shutdown Within Technical Specification Limits (pg. 55)

1. Plant is NOT brought to required operating mode within FNP Technical Specifications LCO Action Statement Time limit.

SU4 - Fuel Clad Degradation (pg. 57)

1. RCS coolant sample activity value indicating fuel clad degradation greater than Technical Specification allowable limits as indicated by ANY of the following:

Dose Equivalent I-131 greater than 0.5 µCi/gm for greater than 48 hours

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Dose Equivalent I-131 greater than Technical Specification figure 3.4.16-1.
IF less than 20% power, THEN use the Dose Equivalent I-131 20% power limit on Technical Specification figure 3.4.16-1

RCS gross specific activity greater than 100/E µCi/gm.

SU5 - RCS Leakage (pg. 58)

1. RCS Unidentified OR pressure boundary leakage greater than 10 gpm.

2. RCS Identified leakage greater than 25 gpm.

SU8 - Inadvertent Criticality (pg. 60)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

ALARMS / COMMUNICATIONS

SU3 - UNPLANNED Loss of Most or All Safety System Annunciation or Indication in The Control Room for Greater Than 15 Minutes (pg. 56)

1. UNPLANNED loss of most OR all (approximately 75% of the MCB annunciators) OR indicators associated with safety systems for greater than 15 minutes.

SU6 - UNPLANNED Loss of All Onsite OR Offsite Communications Capabilities (pg. 59)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
In plant telephones
Public address system
Plant radio systems

2. UNPLANNED loss of ALL of the following off-site communications capability:
ENN (Emergency Notification Network)
ENS (Emergency Notification System)
Commercial phones (Radio, PBX, Satellite, Wireless)
VOIP (Voice Over Internet Protocol)
OPX (Off Premise Extension)

NATURAL DESTRUCTIVE

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA (pg. 76)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5.

2. Report by plant personnel of tornado OR high winds greater than 115 mph striking within the PROTECTED AREA boundary.

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure OR equipment.

5. Report of turbine failure resulting in casing penetration OR damage to turbine OR generator seals.

6. Uncontrolled flooding in the following areas:
Service Water Intake Structure (SWIS)
Auxiliary Building
Turbine Building Basement

7. Sustained hurricane force winds greater than 74 mph forecast to be at the plant site in the next four hours in accordance with FNP-0-AOP-21.0.

E-HU1 - Damage to a loaded cask CONFINEMENT BOUNDARY (pg. 85)

1. Damage to a loaded dry fuel storage cask CONFINEMENT BOUNDARY due to natural phenomena events, accident conditions OR any condition in the opinion of the Emergency Director that affects OR causes a loss of loaded dry fuel storage cask CONFINEMENT BOUNDARY.

FIRE/EXPLOSION

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection (pg. 78)

1. FIRE in buildings OR areas contiguous to any of the following areas NOT extinguished within 15 minutes of control room notification OR control room alarm unless disproved by personnel observation within 15 minutes of the alarm:

Containment	Auxiliary Building
Condensate Storage Tank (CST)	
Service Water Intake Structure (SWIS)	
Refueling Water Storage Tank (RWST)	
Diesel Generator Building	Control Room

TOXIC/FLAMMABLE

HU3 - Release of Toxic, Asphyxiant, or Flammable Gases Deemed Detrimental to Normal Operation of the Plant (pg. 80)

1. Report OR detection of toxic, asphyxiant, OR flammable gas that has OR could enter the Owner Controlled Area in amounts greater than life threatening OR flammable concentrations that can affect NORMAL PLANT OPERATIONS.

2. Report by Local, County, OR State Officials for evacuation OR sheltering of site personnel based on an offsite toxic, asphyxiant, OR flammable gas event.

SECURITY

HU4 - Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant (pg. 81)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

2. A CREDIBLE FNP security THREAT notification.

3. A validated notification from NRC providing information of an aircraft threat.

ED DISCRETION

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE (pg. 83)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process OR have occurred which indicate a potential degradation of the level of safety of the plant OR indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response OR monitoring are expected unless further degradation of safety systems occurs.

Figure 3 – Cold Initiating Condition Matrix

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EMERGENCY CLASSIFICATION AND IMPLEMENTING INSTRUCTIONS

Figure 3 - Cold Initiating Condition Matrix - Modes 5, 6, and Defueled

Columns: RADIOLOGICAL - ALL MODES (EFFLUENTS; RAD LEVELS) | System Malfunctions Modes 5 - 6 - Defueled (AC / DC POWER; RX and CORE / RCS / HEAT REMOVAL; COMMUNICATIONS) | HAZARDS - ALL MODES (NATURAL DESTRUCTIVE – ISFSI; FIRE/EXPLOSION; TOXIC/FLAMMABLE; SECURITY; CR EVACUATION; ED DISCRETION)

GENERAL EMERGENCY

EFFLUENTS

RG1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 1000 mR TEDE **OR** 5000 mR Thyroid CDE for the Actual or Projected Duration of the Release Using Actual Meteorology. (pg. 17)

See Notes

1. VALID reading on any of the following radiation monitors that exceeds OR expected to exceed the reading shown for 15 minutes OR longer:

Steam jet Air Ejector RE-15C	130 µCi/cc (130 R/hr)
Plant Vent Stack RE-29B (NG)	0.9 µCi/cc
Steam Gen Relief RE-60A,B,C	0.5 µCi/cc (0.5 R/hr)
TDAFW Steam Exh RE-60D	11 µCi/cc (11 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 1,000 mR TEDE **OR** 5,000 mR thyroid CDE at **OR** beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 1,000 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 5,000 mR for one hour of inhalation, at **OR** beyond the site boundary.

RAD LEVELS

Note: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

RX and CORE / RCS / HEAT REMOVAL

CG1 - Loss of RPV Inventory Affecting Fuel Clad Integrity with Containment Challenged with Irradiated Fuel in the RPV. (pg. 87)

1. Loss of RPV inventory as indicated by any of the following:
Unexplained Containment sump level rise
Unexplained Reactor Coolant Drain Tank (RCDT) level rise
Unexplained Waste Holdup Tank (WHT) level rise

AND

2. RPV Level:
a. Less than elevation 118' (Top Of Active Fuel) for greater than 30 minutes
OR
b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery for greater than 30 minutes as evidenced by any of the following:
Incore Seal Table R-7 ≥3 mR/hr
Erratic Source Range Monitor Indication

AND

3. Containment challenged as indicated by any of the following:
Explosive mixture inside containment greater than **OR** equal to 6% H_2

Pressure greater than **OR** equal to 5 psig **WITH** CONTAINMENT CLOSURE established

Pressure greater than **OR** equal to 54 psig **WITH** Tech Spec containment integrity intact

CONTAINMENT CLOSURE **NOT** established

SECURITY

HG1 - HOSTILE ACTION Resulting in Loss Of Physical Control of the Facility. (pg. 62)

1. A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions.

2. A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in pool.

ED DISCRETION

HG2 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of General Emergency. (pg. 63)

Other conditions exist which in the judgment of the Emergency Director indicate that events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity **OR** HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

SITE AREA EMERGENCY

EFFLUENTS

RS1 - Offsite Dose Resulting from an Actual or Imminent Release of Gaseous Radioactivity Exceeds 100 mR TEDE **OR** 500 mR Thyroid CDE for the Actual or Projected Duration of the Release. (pg. 19)

See Notes

1. VALID reading on any of the following radiation monitors that exceeds **OR** is expected to exceed the reading shown for 15 minutes **OR** longer:

Steam jet Air Ejector RE-15C	13 µCi/cc (13 R/hr)
Plant Vent Stack RE-29B (NG)	0.09 µCi/cc
Steam Gen Relief RE-60A,B,C	0.05 µCi/cc (0.05 R/hr)
TDAFW Steam Exh RE-60D	1.1 µCi/cc (1.1 R/hr)

2. Dose assessment using actual meteorology indicates doses greater than 100 mR TEDE **OR** 500 mR thyroid CDE at **OR** beyond the site boundary.

3. Field survey results indicate closed window dose rates exceeding 100 mR/hr expected to continue for more than one hour; **OR** analyses of field survey samples indicate thyroid CDE of 500 mR for one hour of inhalation, at **OR** beyond the site boundary.

RAD LEVELS

Note: If dose assessment results are available at the time of declaration, the classification should be based on Threshold Value #2 instead of Threshold Value #1. While necessary declarations should not be delayed awaiting results, the dose assessment should be initiated / completed in order to determine if the classification should be subsequently escalated.

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

RX and CORE / RCS / HEAT REMOVAL

CS1 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability. (pg. 89) **Mode 5 only**

1. Loss of Reactor Pressure Vessel (RPV) inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE **NOT** established as indicated by:
a. RPV level less than 121' (6" below Bottom ID of RCS loop).
OR
b. RPV level **CANNOT** be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by unexplained level rise in any of the following:
Containment Sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

2. Loss of RPV inventory affecting core decay heat removal capability with CONTAINMENT CLOSURE established as indicated by:
a. RPV level less than 118' (Top of Active Fuel (TOAF)).
OR
b. RPV level **CANNOT** be monitored for greater than 30 minutes with a possible loss of RPV inventory as indicated by any of the following:
Containment sump level rise
Unexplained Reactor Coolant Drain Tank (RCDT) level rise
Unexplained Waste Holdup Tank (WHT) level rise
Erratic Source Range monitor indication

CS2 - Loss of RPV Inventory Affecting Core Decay Heat Removal Capability with Irradiated Fuel in the RPV. (pg. 91) **Mode 6 only**

1. **WITH** CONTAINMENT CLOSURE **NOT** established:
a. RPV level less than elevation 121' (6" below Bottom ID of RCS loop).
OR
b. RPV level **CANNOT** be monitored **WITH** indication of core uncovery as evidenced by ANY of the following:
Incore Seal Table R-7 ≥3 mR/hr
Erratic Source Range Monitor Indication

2. **WITH** CONTAINMENT CLOSURE established
a. RPV level less than elevation 118' (Top of Active Fuel).
OR
b. RPV level **CANNOT** be monitored **WITH** Indication of core uncovery as evidenced by ANY of the following:
Containment High Range Radiation Monitor R27 A **OR** B ≥ 100 R/hr
Incore Seal Table R7 ≥ 3 mR/hr
Erratic Source Range Monitor Indication

SECURITY

HS4 - HOSTILE ACTION within the PROTECTED AREA. (pg. 66)

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by the site security force.

CR EVACUATION

HS2 - Control Room Evacuation Has Been Initiated **AND** Plant Control Cannot Be Established. (pg. 64)

1. a. Control Room evacuation has been initiated
AND
b. Control of the plant can **NOT** be established per AOP-28.0, Control Room Inaccessibility, within 15 minutes.

ED DISCRETION

HS3 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of Site Area Emergency. (pg. 65)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which involve actual **OR** likely major failures of plant functions needed for protection of the public **OR** HOSTILE ACTION that results in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels that exceed EPA Protective Action Guideline exposure levels beyond the site boundary.

ALERT

EFFLUENTS

RA1 - Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds 200 Times the Radiological Effluent Technical Specifications for 15 Minutes or Longer. (pg. 21)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 15 minutes.

1. VALID reading on any of the following effluent monitor that exceeds 200 times the alarm setpoint established by a current radioactivity discharge permit for 15 minutes **OR** longer.

Liq RW Eff RE-18	200x LWRP setpoint (planned release); ≥1 x 10^6 cpm (no planned release)
S/G BD Eff Line RE-23B	2.8 x 10^5 cpm
SJAE RE-15A	3.5 x 10^4 cpm
Plant Vent Gas RE-14	200x GWRP setpoint (planned release); ≥1 x 10^6 cpm (no planned release)
Plant Vent Stack RE-22	4.0 x 10^4 cpm
Plant Vent Stack RE-29B(NG)	8.9 x 10^{-2} µCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 15 minutes **OR** longer:
MS Atmos Relief (R60A,B,C) 1.4 x 10^1 R/hr
TDAFW Exhaust (R60D) 1.4 x 10^1 R/hr

3. Confirmed sample analyses for gaseous or liquid releases indicates concentrations **OR** release rates in excess of 200 times Technical Specification 5.5.4.b as confirmed by the ODCM, with a release duration of 15 minutes **OR** longer.

RAD LEVELS

RA2 - Damage to Irradiated Fuel **OR** Loss of Water Level that Has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. (pg. 23)

1. UNPLANNED VALID alarm on any of the following radiation monitors:
Drumming Station RE-0008
Containment Purge Ventilation Monitor RE-24A **OR** B
Spent Fuel Pool Ventilation Monitor RE-25A **OR** B
Spent Fuel Pool Area Radiation Monitor RE-5

2. Loss of water level that has or will result in the uncovering of irradiated fuel outside the Reactor Vessel as indicated by ANY of the following:

Report of personnel during fuel assembly movements.

Spent Fuel Pool Storage	Less than El 129
Transfer Canal	Less than El 116
Reactor Core Elevation	Less than El 118

RA3 - Release of Radioactive Material or Rises in Radiation Levels Within the Facility That Impedes Operation of Systems Required to Maintain Safe Operations or to Establish or Maintain Cold Shutdown (pg. 25)

1. VALID radiation monitor readings greater than 15 mR/hr in areas requiring continuous occupancy to maintain plant safety functions:
Control Room radiation monitor RE-1A

2. UNPLANNED VALID radiation readings greater than 1000 mR/hr (1 R/hr) values in the following areas requiring infrequent access to maintain plant safety functions.
RadioChemistry Lab RE-3
Electrical and Piping Penetration Rooms

Charging Pump Room Area RE-4	VCT Valve Room
Sample Room Area RE-6	Seal Water HX Room
Lower Equipment Room	CCW HX Room
Main Steam Valve Room	Hot Shutdown Panels
4160 V ESF Bus Rooms	DC Switchgear Rooms
Control Rod Drive Room	Valve Box 1, 2, 3, and 4
Diesel Building	Service Water Intake Structure
Turbine Building Air Compressor Area	

AC / DC POWER

CA3 - Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses. (pg. 95)

1. a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G.
AND
b. Failure of emergency diesel generators to supply power to emergency busses.
AND
c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

RX and CORE / RCS / HEAT REMOVAL

CA1 - Loss of RCS Inventory. (pg. 93) **Mode 5 only**

1. Loss of RCS inventory as indicated by RPV level less than 121' 6" (bottom ID of RCS loop)

2. a. RCS level **CANNOT** be monitored for greater than 15 minutes
AND
b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in ANY of the following:
Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

CA2 - Loss of RPV Inventory with irradiated fuel in the RPV. (pg. 94) **Mode 6 only**

1. Loss of inventory as indicated by RPV level less than 121' 6" (bottom ID of RCS loop)

2. a. RPV level **CANNOT** be monitored for greater than 15 minutes
AND
b. A possible loss of RCS inventory may be occurring as indicated by unexplained level rise in ANY of the following:
Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

CA4 - Inability to Maintain Plant in Cold Shutdown with Irradiated Fuel in the RPV. (pg. 96)

1. An UNPLANNED event results in RCS temperature exceeding 200°F with:
a. CONTAINMENT CLOSURE **NOT** established
AND
b. RCS integrity **NOT** established

Note 1 The Emergency Director should not wait until the indicated time of Threshold Values 2 or 3 has elapsed, but should declare the event as soon as it is determined that the duration has or will likely be exceeded.

Note 2 If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced then Threshold Values 2 and 3 are not applicable.

2. An UNPLANNED event results in RCS temperature exceeding 200F for greater than 20 minutes (Note 2) with:
a. CONTAINMENT CLOSURE established.
AND
b. RCS integrity **NOT** established.
OR
c. RCS inventory reduced.

3. An UNPLANNED event results in:
a. RCS temperature exceeding 200F for greater than 60 minutes (Note 2)
OR
b. RCS pressure increasing greater than 10 psig

NATURAL DESTRUCTIVE

HA1 - Natural and Destructive Phenomena Affecting the Plant VITAL AREA. (pg. 67)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5 indicates Seismic event greater than Operating Basis Earthquake (OBE).

2. Tornado **OR** high winds greater than 115 mph within the PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures/equipment **OR** the Control Room has indication of degraded performance of any listed systems:

Containment	Auxiliary Building
Service Water Pond	Diesel Generator Building
Service Water Intake Structure (SWIS)	
Refueling Water Storage Tank (RWST)	
Condensate Storage Tank (CST)	

3. Vehicle crash within PROTECTED AREA boundary **AND** resulting in VISIBLE DAMAGE to any of the following plant structures **OR** equipment therein **OR** Control Room has indication of degraded performance of those systems:

Containment	Auxiliary Building
Service Water Intake Structure (SWIS)	Service Water Pond
Refueling Water Storage Tank (RWST)	
Condensate Storage Tank (CST)	Diesel Generator Building

4. Turbine failure-generated missiles result in any VISIBLE DAMAGE to **OR** penetration of any of the following plant areas containing safety-related equipment, their controls **OR** their power supplies.

Containment	Auxiliary Building
Condensate Storage Tank (CST)	Control Room
Refueling Water Storage Tank (RWST)	
Diesel Generator Building	

5. Uncontrolled flooding in the following areas that results in degraded safety system performance as indicated in the control room **OR** that creates industrial safety hazards (e.g., electric shock) that precludes access necessary to operate **OR** monitor safety equipment.
Service Water Intake Structure (SWIS)
Auxiliary Building
Turbine Building Basement

6. Sustained hurricane winds greater than 74 mph onsite resulting in VISIBLE DAMAGE to plant structures within the PROTECTED AREA boundary containing equipment necessary for safe shutdown, or has caused damage as evidenced by control room indication of degraded performance of those systems.

FIRE/EXPLOSION

HA2 - FIRE **OR** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown. (pg. 69)

1. FIRE **OR** EXPLOSION **AND** affected system parameter indications show degraded performance **OR** plant personnel report VISIBLE DAMAGE to permanent structures **OR** safety related equipment in any of the following VITAL AREAs:
Containment
Auxiliary Building
Control Room
Storage Pond Dam and Dike
Service Water Pond
Condensate Storage Tank (CST)
Service Water Intake Structure (SWIS)
Refueling Water Storage Tank (RWST)
Diesel Generator Building
Pond Spillway Structure

TOXIC/FLAMMABLE

HA3 - Release of Toxic, Asphyxiant, or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown. (pg. 71)

1. Report **OR** detection of toxic **OR** asphyxiant gas within **OR** contiguous to a VITAL AREA in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH).

2. Report **OR** detection of flammable gases in concentration greater than the LOWER FLAMMABILITY LIMIT within **OR** contiguous to a VITAL AREA.

SECURITY

HA4 - HOSTILE ACTION within the OWNER CONTROLLED AREA or Airborne Attack Threat. (pg. 73)

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLED AREA as reported by the security shift supervision.

2. A validated notification from NRC of an airliner attack threat within 30 minutes of the site.

CR EVACUATION

HA5 - Control Room Evacuation Has Been Initiated. (pg. 74)

1. Entry into AOP-28.0, Control Room Inaccessibility, for Control Room evacuation.

ED DISCRETION

HA6 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of an Alert. (pg. 75)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which involve actual **OR** likely potential substantial degradation of the level of safety of the plant **OR** a security event that involves probable life threatening risk to site personnel or damage to site equipment because of intentional malicious dedicated efforts of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

UNUSUAL EVENT

EFFLUENTS

RU1 – Any UNPLANNED Release of Gaseous or Liquid Radioactivity to the Environment that Exceeds Two Times the Radiological Effluent Technical Specifications for 60 Minutes or Longer. (pg. 27)

Note: The Emergency Director should not wait until 60 minutes has elapsed, but should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes.

1. VALID reading on any effluent monitor that exceeds two times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes **OR** longer.

Liquid RW Eff (RE-18)	2x LWRP setpoint (planned release); 1.6 x 10^4 cpm (no planned release)
S/G BD Eff Line (RE-23B)	2.8 x 10^3 cpm
SJAE RE-15	3.5 x 10^2 cpm
Plant Vent Gas RE-14	2 x GWRP setpoint (planned release); 3.2 x 10^4 cpm (no planned release)
Plant Vent Stack RE-22	4.0 x 10^2 cpm
Plant Vent Stack RE-29B(NG)	8.9 x 10^{-4} µCi/cc

2. VALID reading on any of the following radiation monitors that exceeds the reading shown for 60 minutes **OR** longer:
MS Atmos Relief (R60A,B,C) 1.4 x 10^{-1} R/hr
TDAFW Exhaust (R60D) 1.4 x 10^{-1} R/hr

3. Confirmed sample analyses for gaseous **OR** liquid releases indicates concentrations **OR** release rates, with a release duration of 60 minutes **OR** longer, in excess of two times Technical Specification 5.5.4.b, as confirmed by the ODCM.

RAD LEVELS

RU2 – Unexpected Rise in Plant Radiation. (pg. 29)

1. a. VALID indication of uncontrolled water level lowering in the reactor refueling cavity, spent fuel pool, **OR** fuel transfer canal with all irradiated fuel assemblies remaining covered by water.
Personnel report of low water level
Annunciator EH2 "SFP LVL HI/LO"
Personnel report of cavitation **OR** low discharge pressure for SFP Pump Discharge Pressure **AND/OR** RHR Pump Discharge Pressure
AND
b. UNPLANNED VALID Direct Area Radiation Monitor readings rise on any of the following:
RE-0005 in the fuel building
RE-0002 in containment
RE-0027A **OR** B in containment

2. UNPLANNED VALID Direct Area Radiation Monitor readings rise by a factor of 1000 over normal* levels.

*Normal levels can be considered as the highest reading in the past twenty-four hours excluding the current peak value.

AC / DC POWER

CU3 - Loss of All Offsite Power to Essential Busses for Greater Than 15 Minutes. (pg. 101)

NOTE: A NOUE should not be made for pre-planned testing such as SI/LOSP testing.

1. a. Loss of offsite power to **OR** from Start Up Transformers 1(2)A **AND** 1(2)B resulting in loss of all offsite electrical power to **BOTH** 4160V ESF busses 1(2)F **AND** 1(2)G for greater than 15 minutes.
AND
b. At least one emergency diesel generator supplying power to **EITHER** 4160V ESF buss 1(2)F **OR** 1(2)G.

CU7 - UNPLANNED Loss of Required DC Power for Greater than 15 Minutes. (pg. 104)

1. a. UNPLANNED loss of Vital DC power to 125 VDC Bus A **AND** B indicated by bus voltage indications less than 105 VDC.
AND
b. Failure to restore power to at least one DC bus within 15 minutes from the time of loss.

RX and CORE / RCS / HEAT REMOVAL

CU1 - RCS Leakage (pg. 98) **Mode 5 only**

1. Unable to establish or maintain pressurizer level greater than 15%.

CU2 - UNPLANNED Loss of RCS Inventory with Irradiated Fuel in the RPV. (pg. 99) **Mode 6 only**

1. UNPLANNED RCS level lowering below 129' (RPV flange) for greater than **OR** equal to 15 minutes.

2. a. RPV level **CANNOT** be monitored
AND
b. A possible loss of RPV inventory may be occurring as indicated by unexplained level rise in ANY of the following:
Containment sump
Reactor Coolant Drain Tank (RCDT)
Waste Holdup Tank (WHT)

CU8 - Inadvertent Criticality. (pg. 105)

1. An UNPLANNED sustained positive startup rate observed on nuclear instrumentation.

CU4 - UNPLANNED Loss of Decay Heat Removal Capability with Irradiated Fuel in the RPV. (pg.102)

Note: The Emergency Director should not wait until 15 minutes has elapsed, but should declare the event as soon as it is determined that the duration has or will likely exceed the Threshold Value.

1. An UNPLANNED event results in RCS temperature exceeding 200F.

2. Loss of all RCS temperature **AND** RPV level indication for greater than 15 minutes.

COMMUNICATIONS

CU6 - UNPLANNED Loss of All Onsite **OR** Offsite Communications Capabilities. (pg. 103)

1. UNPLANNED loss of ALL of the following on-site communications capability affecting the ability to perform routine operations:
In plant telephones
Public address system
Plant radio systems

2. UNPLANNED loss of ALL of the following off-site communications capability:
ENN (Emergency Notification Network)
ENS (Emergency Notification System)
Commercial phones (Radio, PBX, Satellite, Wireless)
VOIP (Voice Over Internet Protocol)
OPX (Off Premise Extension)

NATURAL DESTRUCTIVE

HU1 - Natural and Destructive Phenomena Affecting the PROTECTED AREA. (pg. 76)

1. A felt earthquake validated in accordance with FNP-1-ARP-1.12, LOCATION MK5.

2. Report by plant personnel of tornado **OR** high winds greater than 115 mph striking within the PROTECTED AREA boundary.

3. Crash of vehicle, large enough to cause significant damage, into plant structures containing functions or systems required for safe shutdown within the PROTECTED AREA boundary.

4. Report by plant personnel of an unanticipated EXPLOSION within the PROTECTED AREA boundary resulting in VISIBLE DAMAGE to permanent structure **OR** equipment.

5. Report of turbine failure resulting in casing penetration **OR** damage to turbine **OR** generator seals.

6. Uncontrolled flooding in the following areas:
Service Water Intake Structure (SWIS)
Auxiliary Building
Turbine Building Basement

7. Sustained hurricane force winds greater than 74 mph forecast to be at the plant site in the next four hours in accordance with FNP-0-AOP-21.0.

E-HU1 - Damage to a loaded cask CONFINEMENT BOUNDARY. (pg. 85)

1. Damage to a loaded dry fuel storage cask CONFINEMENT BOUNDARY due to natural phenomena events, accident conditions **OR** any condition in the opinion of the Emergency Director that affects **OR** causes a loss of loaded dry fuel storage cask CONFINEMENT BOUNDARY

FIRE/EXPLOSION

HU2 - FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection. (pg. 78)

1. FIRE in buildings **OR** areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification **OR** control room alarm unless disproved by personnel observation within 15 minutes of the alarm:
Containment
Auxiliary Building
Control Room
Condensate Storage Tank (CST)
Service Water Intake Structure (SWIS)
Refueling Water Storage Tank (RWST)
Diesel Generator Building

TOXIC/FLAMMABLE

HU3 - Release of Toxic, Asphyxiant, or Flammable Gases Deemed Detrimental to Normal Operation of the Plant. (pg.80)

1. Report **OR** detection of toxic, asphyxiant, **OR** flammable gas that has **OR** could enter the Owner Controlled Area in amounts greater than life threatening **OR** flammable concentrations that can affect NORMAL PLANT OPERATIONS.

2. Report by Local, County, **OR** State Officials for evacuation **OR** sheltering of site personnel based on an offsite toxic, asphyxiant, **OR** flammable gas event.

SECURITY

HU4 - Confirmed SECURITY CONDITION or Threat Which Indicates a Potential Degradation in the Level of Safety of the Plant. (pg. 81)

1. A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by security shift supervision.

2. A CREDIBLE FNP security THREAT notification.

3. A validated notification from NRC providing information of an aircraft threat.

ED DISCRETION

HU5 - Other Conditions Existing Which in the Judgment of the Emergency Director Warrant Declaration of a NOUE. (pg. 83)

1. Other conditions exist which in the judgment of the Emergency Director indicate that events are in process **OR** have occurred which indicate a potential degradation of the level of safety of the plant **OR** indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response **OR** monitoring are expected unless further degradation of safety systems occurs.