Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Partial or Complete Loss of AC / 6	Group #		1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13) Whether a partial or complete loss of A.C. power has occurred	K/A # 295003		AA2.05
	Importance Rating		4.2

Proposed Question: The plant is shutdown for a Refueling outage. Site electrical power is being provided from the 115 KV. The only deviation from the normal alignment is that disconnect 10017, North-South Bus Disconnect, is currently OPEN. From this condition, circuit breaker 10022, Lighthouse Hill, trips.

Which one of the following identifies the expected procedural response?

RO/SRO

S1

a) AOP-16, Loss of 10300 Bus and AOP-18, Loss of 10500 Bus

b) AOP-17, Loss of 10400 Bus and AOP-19, Loss of 10600 Bus

c) AOP-57, Recovery from Residual Bus Transfer

d) AOP-49A, Station Blackout In Cold Condition

Proposed Answer: b) AOP-17, Loss of 10400 Bus and AOP-19, Loss of 10600 Bus

Explanation (Optional):

Technical Reference(s): OP-44, AOP-17 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71D, EO-1.05.a, 1.06, 1.09 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	Group #	1	1
Knowledge of facility ALARA program. (CFR: 41.12 / 43.4 / 45.9 / 45.10)	K/A # 295001	2.3.2	2.3.2
	Importance Rating	2.5	2.9

Proposed Question: During 100% power, the Shift Manager authorizes an entry into the Steam Affected Area to find the source of a new steam leak. The Operator is given a ten (10) minute limit for the search based on expected dose rates. Just as the Operator enters the Steam Affected Area, an announcement is made that 'A' Recirculation Pump has tripped. The dose rate in the Steam Affected Area drops to ½ of the pre-transient level.

The Operator should......

RO/SRO

1/2

a) Double the search time that was allowed to adequately identify the leak.

b) Double the search time that was allowed to identify additional discrepancies.

c) Leave the area when the ten (10) minute time is expired.

d) Request an additional Operator to assist in the leak identification.

Proposed Answer: c) Leave the area when the ten (10) minute time is expired.

Explanation (Optional):

Technical Reference(s): AP-7.03 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP-7.03, EO-28.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Complete Loss of AC / 6	Group #	1	1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13) Cause of partial or complete loss of A.C. power	K/A # 295003	AA2.01	AA2.01
	Importance Rating	3.4	3.7

Proposed Question: From a normal full power operating condition, a complete and instantaneous loss of bus 10500 occurs.

Which one of the following is a **<u>LIKELY</u>** cause for the occurrence?

RO/SRO

2/3

a) Loss of DC Control Power to bus 10500

b) Activation of the bus 10500 Degraded Bus Voltage timer

c) Ground fault trip of circuit breaker 10514

d) Overcurrent condition on CRD pump A motor

Proposed Answer: c) Ground fault trip of circuit breaker 10514

Explanation (Optional):

Technical Reference(s): AOP-18 (Attach if not previously provided)

ARP-09-8-2-8

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71E, EO-1.05.C, 1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Total Loss of DC Pwr / 6	Group #	1	1
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: (CFR: 41.7 / 45.8) Battery charger	K/A # 295004	AK2.01	AK2.01
	Importance Rating	3.1	3.1

Proposed Question: While at full power, Maintenance disconnects 125 VDC Station Battery 'A' to replace a faulty cell.

WHICH ONE (1) of the following will be the response of the system during a large 125 VDC load emergency starting?

RO/SRO

3/4

a) The charger will trip on high starting currents associated with emergency loads.

b) The charger will supply emergency loads under these conditions for one hour.

c) The charger will supply both normal and emergency loads for four hours.

d) The charger will supply emergency loads for four hours.

Proposed Answer: a) The charger will trip on high starting currents associated with emergency loads.

Explanation (Optional):

Technical Reference(s): OP-43A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B, EO-1.05.A.2, 1.13A (As available)

Question Source: Bank # Duane Arnold 1 INPO # 7209 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/19/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Keep this one & modify

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Total Loss of DC Pwr / 6	Group #	1	1
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: (CFR: 41.7 / 45.8)	K/A # 295004	AK2.01	AK2.01
Battery charger			
	Importance Rating	3.1	3.1

Proposed Question: While at full power, maintenance disconnects 125VDC Station Battery 'A' to replace a faulty cell ~~During power operations an operator opens the battery supply breaker to the 250 volt battery system and only the charger is supplying bus voltage.~~

WHICH ONE (1) of the following will be the response of the system during ~~emergency load starting requiring 250 volt power?~~ 125 large 125VDC load emergency starting

RO/SRO

3/4

a) The charger will trip on high starting currents associated with emergency loads.

b) The charger will supply emergency loads under these conditions for one hour.

c) The charger will supply both normal and emergency loads for four hours.

d) The charger will supply the emergency loads for four hours.

Proposed Answer: a) The charger will trip on high starting currents associated with emergency loads.

Explanation (Optional):

Technical Reference(s): OP-43A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B Obj. 1.05.A.2, 1.13a (As available)

Question Source: Bank # ~~Duane Arnold 1 INPO # 7209~~ (modified to fit)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam ~~1/19/1996~~ ✓

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis ✓

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Main Turbine Generator Trip / 3	Group #	1	1
Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: (CFR: 41.7 / 45.8) Reactor pressure control	K/A # 295005	AK2.07	AK2.07
	Importance Rating	3.6	3.7

Proposed Question: The following conditions exist while performing a shutdown for a refueling outage:

- Reactor power is 27%.
- Recirculation flow is at minimum.
- A Main Turbine Trip occurs.

Turbine Bypass Valves respond as designed.

Which of the following is **CORRECT** concerning the plant and operator response?

a) Turbine Bypass Valves will **NOT** be able to control Reactor pressure. **Unless** available steam drains are opened, it will be necessary to insert a manual Reactor SCRAM.

b) Turbine Bypass Valves will **NOT** be able to control Reactor pressure. It will be necessary to insert Control Rod Cram Groups in accordance with Reactor Analyst Instructions in RAP-7.3.16.

RO/SRO

4/5

c) Turbine Bypass Valves will control Reactor pressure. It will be necessary to close all available steam line drains to assist Reactor pressure control.

d) Turbine Bypass Valves will control Reactor pressure. It will be necessary to operate Turbine Bypass Valves manually.

Proposed Answer: a) Turbine Bypass Valves will **NOT** be able to control Reactor pressure. **Unless** available steam drains are opened, it will be necessary to insert a manual Reactor SCRAM.

Explanation (Optional):

Technical Reference(s): OP-9, AOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-94C, EO-1.10.K (As available)

Question Source: Bank # Duane Arnold 1 INPO Bank # 620 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 5/25/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Main Turbine Generator Trip / 3	Group #	1	1
Knowledge of the interrelations between MAIN TURBINE GENERATOR TRIP and the following: (CFR: 41.7 / 45.8)	K/A # 295005	AK2.07	AK2.07
Reactor pressure control			
	Importance Rating	3.6	3.7

Proposed Question: 'The plant is performing a shutdown for a refueling outage. The following conditions at exist: ~~One Condensate and Feed water pump are in operation.~~ Reactor power is ~~29%~~ 27%. Recirculation / ~~Recirc~~ flow is at minimum. The crew is inserting control rods when a Turbine Trip occurs. The Turbine Bypass Valves respond as designed.

Which of the following is ~~the~~ CORRECT concerning the plant and operator response to this event?

RO/SRO

4/5

a) The Bypass Valves will NOT be able to control Reactor pressure. (Unless additional steam drains are opened,) It will be necessary for operators to insert a manual Reactor Scram.

b) The Bypass Valves will NOT be able to control Reactor pressure. It will be necessary for operators to insert ~~the~~ control rod cram group~~s~~ in accordance with ~~Fast Power Reduction~~. Reactor Analyst Instructions in RAP 7.3.16

c) The Bypass Valves will be able to control Reactor pressure. It will be necessary for operators to ~~open~~ close all available steam line drains to assist Reactor pressure control.

d) The Bypass Valves will be able to control Reactor pressure, However, ~~No~~ operator action will be necessary to operate Bypass Valves manually

Proposed Answer: a) The Bypass Valves will NOT be able to control Reactor pressure. It will be necessary for operators to insert a manual Reactor Scram

Explanation (Optional): 'IPOI-5 directs a manual scram if an auto scram is unavoidable. Turbine trip does not auto scram reactor < 30%, but an auto scram is unavoidable as RPV pressure increases. The SD say the Bypass Valve capacity is 25%; recent studies have show that, with the concurrent loss of feedwater heating, BPV pressure control would be inadequate > 22..5%. An RO should know this.

Distracter B- NOT be able" is correct but cram rod insertion is not. Fast power reduction allows the cram group to lower power to < 75% load line, after that a scram is required. At 29% power the cram rods would already be inserted.

Distracter C- "Be able " is incorrect; Opening MSL drains will help but most drains would already be open and this action will not make scram unavoidable.

Distracter D- "Be able " is incorrect and a manual scram is necessary.

Technical Reference(s): OP-9, AOP-~~6~~ 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP94C Obj. 1.10.K (As available)

Question Source: Bank # Duane Arnold 1 INPO Bank # 620 (mod to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 5/25/1999

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
SCRAM / 1	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to SCRAM :	K/A # 295006	AK1.03	AK1.03
(CFR: 41.8 to 41.10)			
Reactivity control			
	Importance Rating	3.7	4.0

Proposed Question: The Control Room Supervisor orders you to insert a manual scram because power is unexpectedly rising. Which of the following responses indicates that the scram has successfully controlled reactivity under all conditions?

RO/SRO

5/6

a) Reactor power dropping rapidly through the IRM and SRM ranges.

b) 6 rods indicate position 02, remaining rods indicate position 00.

c) 1 rod indicates 48, 1 rod at 10, remaining rods indicate position 00.

d) Annunciators, 09-5-1-13, RPS A MAN SCRAM and 09-5-1-14, RPS B MAN SCRAM are in alarm.

Proposed Answer: b) 6 rods indicate position 02, remaining rods indicate position 00.

Explanation (Optional):

Technical Reference(s): AOP-1, EP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP, EO-2.01, EOP2LP, EO-1.07 (As available)

Question Source: Bank # Dresden 2 INPO Bank # 6558 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 3/11/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

d) Annunciators 09-5-1-13, RPS A MAN SCRAM and 09-5-1-14, RPS B MAN SCRAM in alarm.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
SCRAM / 1	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to SCRAM :	K/A # 295006	AK1.03	AK1.03
(CFR: 41.8 to 41.10)			
Reactivity control	Importance Rating	3.7	4.0

Proposed Question: The ~~Unit Supervisor~~ Control Room Supervisor orders you to insert a manual scram because power is unexpectedly increasing. Which of the following responses indicates that the scram has successfully controlled reactivity under all conditions?

RO/SRO 5/6

a) Reactor power dropping rapidly. ~~through the IRM and SRM ranges~~
b) 6 rods indicate position 02, remaining rods indicate position 00.
c) 1 rod indicates 48, 1 rod at 10, remaining rods indicate position 00.
d) ~~Panel 902-5 CHANNEL A RX SCRAM and CHANNEL B RX SCRAM alarms lit and Scram Solenoids Group indicating lights A1, A2, B1, and B2 all lit.~~

Proposed Answer: b) 6 rods indicate position 02, remaining rods indicate position 00.

Explanation (Optional):

Technical Reference(s): AOP-1, EP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: LP-AOP Obj. 2.01 (As available)

Question Source: Bank # Dresden 2 INPO Bank # 6558

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 3/11/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Control Room Abandonment / 7	Group #	1	1
Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8) Control room HVAC	K/A # 295016	AK2.03	AK2.03
	Importance Rating	2.9	3.1

Proposed Question: A Control Room HVAC fire has resulted in a significant amount of smoke in the Control Room requiring Control Room Evacuation.

Which of the following describes the conditions under which plant control may be returned to the Control Room as directed by AOP-43, PLANT SHUTDOWN FROM OUTSIDE THE CONTROL ROOM?

RO/SRO

6/7

a) The fire is verified to be extinguished by the Fire Brigade Leader **BEFORE** Remote Shutdown Panel actions have been commenced.

b) No Control Room damage exists as determined by the County Fire Control Coordinator and Remote Shutdown Panel actions have been completed.

c) The Shift Manager has authorized transferring control to the Control Room and Remote Shutdown Panel to Control Room turnover procedures are completed.

d) The Emergency Director and Security Manager have determined the Control Room is functional and habitable.

Proposed Answer: c) The Shift Manager has authorized transferring control to the Control Room and Remote Shutdown Panel to Control Room turnover procedures are completed.

Explanation (Optional):

Technical Reference(s): AOP-43 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO-1.03.A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Total Loss of CCW / 8	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : (CFR: 41.8 to 41.10) Effects on component/system operations	K/A # 295018	AK1.01	AK1.01
	Importance Rating	3.5	3.6

Proposed Question: The plant is operating at 90% power with one Reactor Building Closed Loop Cooling (RBCLC) pump tagged out of service. An electrical problem causes the two running RBCLC pumps to trip.

Operators have the ability to restore cooling via Emergency Service Water to **EACH** of the following **EXCEPT**:

a) RWCU Non- Regenerative Heat Exchanger

RO/SRO b) Drywell Cooling Assemblies

7/8 c) Recirculation Pump Seal Coolers

d) Drywell Equipment Sump Cooler

Proposed Answer: a) RWCU Non- Regenerative Heat Exchanger

Explanation (Optional):

Technical Reference(s): AOP-11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15, EO-1.09, SDLP-46B, EO-1.06.B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Partial or Total Loss of Inst. Air / 8	Group #	1	1
Knowledge of refueling administrative requirements. (CFR: 43.5 / 45.13)	K/A # 295019	2.2.26	2.2.26
	Importance Rating	2.5	3.7

Proposed Question: The plant is in a refueling outage. The Refuel Bridge is over the core supporting 'In-Vessel' inspections. Thirty, (30) minutes after a complete loss of Instrument Air occurs, an NPO calls from the refuel floor to report that Spent Fuel Pool level has risen several inches over the last hour.

Which of the below is the probable cause?

RO/SRO

8/9

a) RWCU Blowdown Flow Control Valve (12FCV-55) failed closed.

b) Main Steam Line Plugs have depressurized.

c) In-service Fuel Pool Filter/ Demineralizer has isolated.

d) Feedwater Low Flow Control Valve loss of air control signal.

Proposed Answer: a) RWCU Blowdown Flow Control Valve (12FCV-55) failed closed.

Explanation (Optional):

Technical Reference(s): AOP-12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO-1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4,7,10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Loss of Shutdown Cooling / 4	Group #	1	1
Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.7 / 45.6) Alternate heat removal methods	K/A # 295021	AA1.04	AA1.04
	Importance Rating	3.7	3.7

Proposed Question: A loss of shutdown cooling has occurred. The cavity is flooded and the spent fuel pool gates are removed. The current decay heat load of the core and spent fuel pool is $1.8x10^7$ BTU/hr.

Which decay heat removal lineup listed below would provide sufficient decay heat removal?

RO/SRO 9/10

a) RWCU in blowdown mode.
b) Fuel pool cooling system.
c) Decay heat removal system.
d) RWCU in recirculation mode.

Proposed Answer: c) Decay heat removal system.

Explanation (Optional):

Technical Reference(s): AOP-30 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: AOP-30, Attachment 3

Learning Objective: SDLP-10, EO 1.15.a (As available)

Question Source: Bank # JAF LOR 20004206B02C Rev.2

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

ATTACHMENT 3

ALTERNATE COOLING METHODS

METHOD	APPROXIMATE HEAT REMOVAL CAPACITY (BTU/hr)	LIMITATIONS
Decay Heat Removal	3.00E7	• Gates removed between cavity and spent fuel pool
Fuel Pool Cooling	3.30E6	• Gates removed between cavity and spent fuel pool • RBC must be available • SW must be available
Fuel Pool Cooling Assist	2.40E7	• RHR must be available • RHRSW must be available • Gates removed between cavity and spent fuel pool
RWCU Blowdown Mode • 1 pump running • 125 gpm blowdown flow • 125 gpm makeup flow	2.06E6	• No isolation signal present • Makeup source must be available (see list below) • Main Condenser or Radwaste must be available
RWCU Recirc Mode	1.70E6	No isolation signal present
RWCU Blowdown Mode • gravity drain • 50 gpm blowdown flow • 50 gpm makeup flow	1.00E6	• No isolation signal present • Makeup source must be available (see list below) • Main Condenser or Radwaste must be available

Makeup Sources

- Condensate transfer keep-full using Core Spray or RHR
- Control Rod Drive System
- Condensate/Feedwater
- Condensate transfer to skimmer surge tanks (gates removed)
- Condensate transfer to fuel pool using DHR (gates removed)
- Condensate transfer using service box connections on the refuel floor (gates removed)
- Fire Protection System water from local fire hose stations or outside sources
- RHR service water cross-tie
- Fire Water Crosstie

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Refueling Acc Cooling Mode / 8	Group #	1	1
Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : (CFR: 41.5 / 45.6)	K/A # 295023	AK3.02	AK3.02
Interlocks associated with fuel handling equipment			
	Importance Rating	3.4	3.8

Proposed Question: Which one (1) of the following will result in a control rod block during Refuel Floor activities in an outage?

RO/SRO

10/11

a) Mode Switch in START/HOT STBY, Fuel Grapple loaded and Refuel Bridge near or over the Spent Fuel Pool.

b) Mode Switch in REFUEL, Fuel Grapple not full up and Refuel Bridge near or over the Spent Fuel Pool.

c) Mode Switch in REFUEL, a single control rod is **not** full in and selection of any other control rod.

d) Mode Switch in START/HOT STBY, all control rods are full in and selection of any control rod.

Proposed Answer: c) Mode Switch in REFUEL, a single control rod is **not** full in and selection of any other control rod.

Explanation (Optional):

Technical Reference(s): ST-20F (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-08B, EO-1.02, 1.05.B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Drywell Pressure / 5	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : (CFR: 41.8 to 41.10) Drywell integrity: Plant-Specific	K/A # 295024	EK1.01	EK1.01
	Importance Rating	4.1	4.2

Proposed Question:

Following a LOCA, with severe complications, the RHRSW crosstie is being used for RPV makeup. The primary containment water level is 80 feet and rising. Primary containment pressure is 35 psig and rising.

The Primary Containment shall be vented before:

RO/SRO

11/12

a) Torus pressure exceeds 56 psig.

b) Torus pressure reaches 62 psig.

c) Torus pressure reaches 77 psig.

d) Torus pressure reaches 85 psig.

Proposed Answer: d) Torus pressure reaches 85 psig.

Explanation (Optional):

Technical Reference(s): EOP-4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP's

Learning Objective: MIT-301.11E, EO 4.07 (As available)

Question Source: Bank # JAF LOR 20005213B05C Rev.3

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Reactor Pressure / 3	Group #	1	1
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6) ARI/RPT/ATWS: Plant-Specific	K/A # 295025	EA1.07	EA1.07
	Importance Rating	4.1	4.1

Proposed Question: Which **ONE** of the following describes the effect a reactor vessel pressure signal of 1170 psig will have on the reactor recirculation pumps and alternate rod insertion (ARI) system?

The Recirculation motor/generator...

RO/SRO

12/13

a) drive motor breakers will trip and the ARI solenoid valves will energize.

b) generator field breakers will trip and the ARI solenoid valves will energize.

c) drive motor breakers will trip and the ARI solenoid valves will de-energize.

d) generator field breakers will trip and the ARI solenoid valves will de-energize.

Proposed Answer: a) drive motor breakers will trip and the ARI solenoid valves will energize.

Explanation (Optional):

Technical Reference(s): ITS-3.3.4.1/SR-3.3.4.1.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02H EO 1.05.C.2, SDLP-03C EO1.05.C.2 (As available)

Question Source: Bank # Quad Cities 1 INPO Bank # 16832 (Modified for JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 3/16/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Reactor Pressure / 3	Group #	1	1
Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.7 / 45.6)	K/A # 295025	EA1.07	EA1.07
ARI/RPT/ATWS: Plant-Specific			
	Importance Rating	4.1	4.1

Proposed Question: Which ONE of the following describes the effect a reactor vessel pressure signal of ~~1250~~ 1170 psig will have on the reactor recirculation pumps and alternate rod insertion (ARI) system?

The ~~recirc pump~~... Recirculation motor/generator . . .

RO/SRO

12/13

a) drive motor breakers will trip and the ARI solenoid valves will energize.

b) generator field breakers will trip and the ARI solenoid valves will energize.

c) drive motor breakers will trip and the ARI solenoid valves will de-energize.

d) generator field breakers will trip and the ARI solenoid valves will de-energize.

Proposed Answer: ~~b) generator field breaker will trip and the ARI solenoid valves will energize.~~

Explanation (Optional): a) drive motor breakers will trip and the ARI solenoid valves will energize

Technical Reference(s): EOP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: ~~MIT-301.11d EO 1.07~~ (As available)

SDLP-02H 1.05.e.2

SDLP-03C 1.05.C.2

Question Source: Bank # Quad Cities 1 INPO Bank # 16832

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 3/16/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Control Room Abandonment / 7	Group #		1
Ability to perform specific system and integrated plant procedures during different modes of plant operation. (CFR: 45.2 / 45.6) Link to 10CFR-55.43(b)(6)	K/A # 295016		2.1.23
	Importance Rating		4.0

Proposed Question: The Plant was operating at 100% full power with no systems out of service; all equipment was in a normal lineup. Subsequently a fire has occurred in the Control Room and the Control Room was evacuated without performing any AOP-43, Plant Shutdown from Outside the Control Room, actions.

Which of the following AOP-43 actions will ensure the Reactor is shutdown?

RO/SRO

S14

a) Trip RWR MG Set 'A' & 'B' Generator Field Breakers.

b) Trip RWR MG Set 'A' & 'B' Drive Motor Breakers.

c) Isolate and vent the scram air header at Reactor Building 272' Southwest.

d) Place RPS MG Set 'A' Generator Output Breaker in OFF.

Proposed Answer: c) Isolate and vent the scram air header at Reactor Building 272' Southwest.

Explanation (Optional):

Technical Reference(s): AOP-43 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO-1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6,7,10

55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Partial or Total Loss of Inst. Air / 8	Group #		1
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :(CFR: 41.10 / 43.5 / 45.13)	K/A # 295019		AA2.01
Instrument air system pressure			
	Importance Rating		3.6

Proposed Question:

The plant is operating at 100% power with Control Room Annunciators inoperable. During panel walkdowns, the SNO-2 reports that all Air Compressors are running. The SNO-1 reports that Scram Air Header pressure is 60 psig and lowering and one (1) Rod Drift light is lit on the Full Core Display.

Which of the following is the correct response?

a) Enter AOP-27, Control Rod Drift, and manually SCRAM if a second rod drifts.

RO/SRO

b) Reduce Recirculation Pumps to minimum and enter AOP-8, Loss of Reactor Coolant Flow.

S15

c) Manually SCRAM the Reactor and enter AOP-1, Reactor SCRAM.

d) Trip the Main Turbine and enter AOP-2, Main Turbine Trip Without SCRAM.

Proposed Answer: c) Manually SCRAM the Reactor and enter AOP-1, Reactor SCRAM.

Explanation (Optional):

Technical Reference(s): AOP-12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-39, EO-1.15.A, LPAOP, EO-1.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Reactor Pressure / 3	Group #	1	1
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Reactor water level	K/A # 295025	EA2.06	EA2.06
	Importance Rating	3.7	3.8

Proposed Question: Following a reactor SCRAM and MSIV isolation, HPCI is injecting into the reactor. RPV level on narrow Range is 200" and rising. Reactor pressure is 800 psig and rising.

The HPCI turbine will trip ......

RO/SRO

13/16

a) At a lower indicated NR level at 800 psig than at 1100 psig.
b) At a higher indicated NR level at 800 psig than at 1100 psig.
c) At the same indicated NR level at 800 psig and at 1100 psig.
d) When NR level indication reaches 222.5".

Proposed Answer: a) At a lower indicated NR level at 800 psig than at 1100 psig.

Explanation (Optional):

Technical Reference(s): OP-15, attachment 3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23, EO-1.05.C.1, 1.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Suppression Pool High Water Temp. / 5	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : (CFR: 41.8 to 41.10) Pump NPSH	K/A # 295026	EK1.01	EK1.01
	Importance Rating	3.0	3.4

Proposed Question: The following plant conditions exist:

Torus Pressure-1.0 psig

Torus Level- 11.92 feet

What is the maximum Torus water temperature that two (2) RHR Pumps can operate at 8,000 gpm each without exceeding NPSH limitations?

RO/SRO 14/17

a) 173 ° F

b) 182 ° F

c) 200 ° F

d) 206 ° F

Proposed Answer: d) 200 ° F

Explanation (Optional):

Technical Reference(s): OP-13 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: OP-13A, Attachment # 1

Learning Objective: SDLP-13, EO-1.13.A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

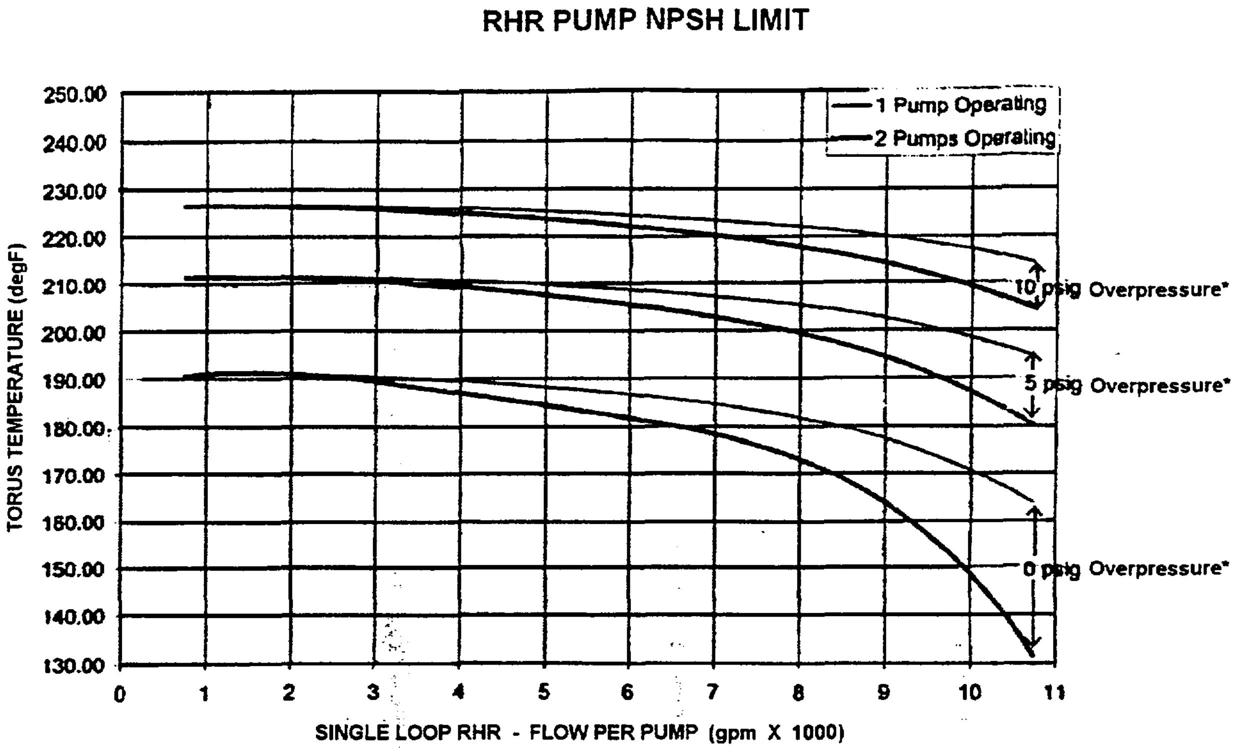
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

RHR NPSH AND VORTEX LIMITS



*Torus Overpressure = Torus Pressure + 0.4 (Torus Water Level - 1.92)

VORTEX LIMIT

Torus Water Level **Greater Than or Equal to** 8.92 feet

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Drywell Temperature / 5	Group #	1	1
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE (CFR: 41.10 / 43.5 / 45.13) Drywell temperature	K/A # 295028	EA2.01	EA2.01
	Importance Rating	4.0	4.1

Proposed Question: EOP-4, PRIMARY CONTAINMENT CONTROL, has been entered due to a valid entry condition. Simultaneously, EPIC power is lost and no Control Room computer screens are available. Which one of the following describes the first preferred indicator(s) to be utilized to determine drywell temperature (Assume normal full power drywell fan configuration)?

RO/SRO

15/18

a) Average of both drywell cooler inlet temperatures from 68TI-100 and 68TI-101 on panel 09-75.

b) Average of drywell cooling assembly air inlet and outlet temperatures from 68TI-100 or 68TI-101 on panel 09-75.

c) Either DW TEMP A 16-1TR-108 or DW TEMP B 16-1TR-107 on panel 09-3.

d) Average of DW TEMP A 16-1TR-108 and DW TEMP B 16-1TR-107 on panel 09-3.

Proposed Answer: d) Average of DW TEMP A 16-1TR-108 and DW TEMP B 16-1TR-107 on panel 09-3.

Explanation (Optional):

Technical Reference(s): EP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EP-1 (excluding section 4.7)

Learning Objective: (As available)

Question Source: Bank # JAF LOR 20005204B06C Rev.1

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

RO 15
SRO 18

ENTERGY NUCLEAR OPERATIONS, INC.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
EOP AND SAOG SUPPORT PROCEDURE

EOP ENTRY AND USE
EP-1
REVISION 6

APPROVED BY: [signature] Acting
RESPONSIBLE PROCEDURE OWNER

DATE 10/23/02

EFFECTIVE DATE: 10/26/02

FIRST ISSUE ☐ FULL REVISION ☐ LIMITED REVISION ☒

INFORMATIONAL USE	TSR
TECHNICAL	

REVISION SUMMARY SHEET

REV. NO.	CHANGE AND REASON FOR CHANGE
6	Revised the Minimum Core Flooding Interval listed on Attachment 1 and the Maximum Core Uncovery Time depicted on Attachment 2, to reflect changes from Cycle 16 core reload. (JD-01-102 and JENG-02-0375)
5	4.2.1.A, 4th line, expanded "Steps E and F" to full step numbers for cross referencing. (4.2.1.E, 4.2.1.F)
	Added para. to Subsection 4.2.3 Operator Actions / Strategies to clarify expectations for use of CS and RHR when in Alternate RPV Water Level Control. (PCR 5, 3/31/00)
	Deleted Subsection 4.4, SPDS (Safety Parameter Display System). Requirements moved to AP-12.06, Rev. 3 (PCR dated 10/5/01, originally assigned to AP-12.03)
	To Subsection 5.1, added plant indications to be used to establish status of reactor shutdown. (PCR 7, 12/21/00)
	4.8.4, 5th line, changed "AOP-39" to "AOP-40" to correct typo. (EC#1, 3/16/00)
	4.8.4, removed the reference to "branch piping." (PCR, dated 1/24/02)
	5.3, 2nd bullet, 2nd IF/THEN statement - add "temperature" after "area" in 4th line for clarification.
	5.3, under Reactor Building Radiation Levels, added another means of monitoring radiation levels is by Radiation Protection survey. (PCR 6, dated 3/31/00; PCR dated 6-28-01)
	5.3, under Reactor Building Radiation Levels, clarified the conditions under which the radiation levels are assumed to be greater than the maximum safe level. (PCR, dated 10/06/01)
	Deleted "7.2 Validation" per AP-02.01.
	Changed "RES" to "Radiation Protection," per ODSO-31.
	Deleted Attachment 1, EPIC SPDS Point Status Log. Moved to AP-12.06, Rev. 3. (PCR dated 10/5/01, against AP-12.03)
	Changed attachment numbers to reflect the deletion of Attachment 1.

TABLE OF CONTENTS

⇓COM7.1.1
1.0 PURPOSE

To provide general guidance for use of EOPs and recognition of EOP entry conditions.

This procedure applies during all plant operating modes, except when reactor coolant temperature is less than 212°F and a reactor startup or shutdown is not in progress.

2.0 PRECAUTIONS

None

3.0 PREREQUISITES

None

4.0 SPECIAL INSTRUCTIONS

4.1 Plant Conditions and Parameters

4.1.1 Monitor the general state of the plant.

4.1.2 Monitor the following parameters using multiple indications:

- Reactor power
- RPV water level
- RPV pressure
- Drywell temperature
- Drywell pressure
- Torus water level
- Torus water temperature
- Containment hydrogen
- Secondary containment temperatures
- Secondary containment radiation levels
- Secondary containment differential pressure
- Crescent area water levels
- Reactor building floor sump levels
- Reactor building ventilation exhaust radiation levels

4.1.3 **IF** local monitoring of a plant parameter is required,
THEN perform the following:

A. Evaluate radiological and environmental conditions to determine accessibility.

B. **IF** access to Reactor Building is required,
THEN follow radiation protection requirements established by Radiation Protection.

4.2 Use of EOPs

4.2.1 EOP Entry, Re-entry, and Exit

A. **IF** an EOP entry condition occurs,
THEN enter that EOP, or re-enter if that EOP has already been entered. Exceptions to this requirement are in Steps 4.2.1.E and 4.2.1.F below.

B. **IF** an EOP has been entered,
THEN determine whether concurrent entry into Emergency Plan is warranted.

C. **WHEN** an operating parameter is trending such that an EOP entry condition is imminent or inevitable, the SM or CRS may enter the applicable EOP.

D. **WHEN** an EOP exit condition is satisfied, or it has been determined that an emergency no longer exists, enter the appropriate operating and/or abnormal operating procedures.

E. **IF** primary containment flooding is or was required,
THEN exit the EOPs and enter the SAOGs. EOPs are not re-entered while in SAOGs, even if an EOP entry condition occurs.

F. Reactor building dP could momentarily meet the EOP-5 entry condition (at or above 0 inches of water) while manually isolating the reactor building during normal plant operation. This is an expected system response and EOP-5 entry is not required provided that dP becomes negative following isolation. Entry into EOP-5 is expected during an automatic isolation or emergency if reactor building dP meets the entry condition.

4.2.2 **Adequate Core Cooling**

Heat removal from the reactor sufficient to prevent rupturing the fuel clad. Submergence is the preferred mechanism for cooling the core. Steam cooling is relied upon only if RPV water level cannot be restored and maintained above TAF, cannot be determined, or must be intentionally lowered below TAF. The covered portion of the core remains cooled by boiling heat transfer which generates the steam that cools the uncovered portion. Steam cooling will maintain the hottest peak clad temperature below:

- Steam Cooling with injection - $< 1500\ ^\circ F$
- Steam Cooling without injection - $< 1800\ ^\circ F$

4.2.3 **Operator Actions/Strategies**

A. **IF** conditions or actions specified by a step are not applicable or cannot be implemented, **THEN** the operator shall proceed to the next step.

B. The SM/CRS may direct isolation of a release path at anytime during an event, provided that isolation of the release path will not conflict with the EOPs.

C. "Anticipation of Emergency RPV Depressurization" is defined as an expectation, based upon evaluation of plant conditions, that an emergency RPV depressurization requirement will soon be reached and cannot be averted by the actions of the EOPs. Before this conclusion can be drawn, the effectiveness of the steps preceding the emergency depressurization requirement must be evaluated. The anticipatory depressurization prescribed by the override requires the MSIVs be open, with the main condenser available and the turbine bypass valves operational (bypassing or defeating the MSIV interlocks is not authorized). Therefore, when performing EOP-2, Alternate RPV Level Control, Anticipation of Emergency RPV Depressurization is not allowed as adequate core cooling exists; this time is best used attempting to align additional injection sources since the MSIVs will automatically close and render the bypass valves inoperable. If the lowering RPV level trend is reversed, the requirement for emergency depressurization will be unnecessary.

D. Anytime the EOPs direct opening 7 ADS valves, this action is performed irrespective of the resulting RPV cooldown rate.

E. **WHEN** performing Section ED of EOP-3, the RPV level band is established based upon whether or not level was previously intentionally lowered. Therefore, if RPV level was intentionally lowered below 110" TAF, this upper limit still applies after the emergency depressurization.

F. **WHEN** performing Steam Cooling, direction is provided to establish a stable or lowering pressure trend. It is preferred to stabilize RPV pressure, to the extent possible, at its existing value. For example: If Steam Cooling is entered and RPV pressure is 500 psig, then RPV pressure should be controlled at or below 500 psig, depending upon the systems available for use. If SRVs are being used for pressure control, a band of 450-500 psig is appropriate. If RPV pressure cannot be stabilized, an alternate approach is to establish a lowering pressure trend. This preserves the assumption of the minimum zero injection RPV water level calculation but will accelerate the rate of inventory loss from the RPV. Therefore, the time that steam cooling can be maintained will be shortened.

G. "All Available Drywell Cooling" is defined as operating 3 of the 4 fans per drywell cooling assembly. Operating all 4 fans/assembly is prohibited as the drywell cooling fan motors could be overloaded causing one or more fans to trip.

H. **WHEN** performing RPV Flooding, the Emergency Response Organization will generate a procedure for recovery. This procedure must restore RPV level instrumentation and should consider filling reference legs, piping integrity, drywell temperature, and power supplies. In addition, this procedure shall ensure the following:

- A method to intentionally lower RPV water level in order to return it on-scale.
- Instrument run temperatures are below 212°F.
- RPV pressure has remained at least 50 psig above torus pressure for the Minimum Core Flooding Interval (Attachment 1), prior to terminating all RPV injection sources.
- EOP-7 Shutdown Flooding shall be entered if RPV water level is not restored within the Maximum Core Uncovery Time Limit (Attachment 2).

I. **WHEN** in Alternate RPV Water Level Control, the direction to maximize injection into the RPV ensures all available normal and ECCS systems are used to restore RPV water level. If RPV pressure remains high (>450 psig), the low pressure ECCS systems cannot be completely lined up for injection because of interlocks. However, those low pressure ECCS systems should be lined up such that maximum flow will be delivered to the RPV as soon as RPV pressure drops below the system's shutoff head and interlock setpoints.

CS systems will be lined up and running on min flow until RPV pressure is below the shutoff head and 450 psig. If previously Terminated and Prevented, CS will be lined up per EP-5, Termination And Prevention Of RPV Injection.

RHR systems may be used for containment control functions as long as injection is prohibited by RPV pressure. However, once RPV pressure is within the shutoff head of the RHR pumps, the RHR system should be realigned to only the LPCI mode until RPV water level is restored. If previously Terminated and Prevented, RHR will be lined up per EP-5.

4.3 Manual Control of Automatic Systems

⇓COM7.1.2

4.3.1 Do not override an automatic initiation of a safety function unless one of the following conditions exist:

- Adequate core cooling is assured by at least two independent indications
- Misoperation in automatic mode is confirmed by at least two independent indications
- Required by EOPs

⇓COM7.1.3

4.3.2 **IF** an operator cannot be dedicated to monitor systems placed in the manual mode, **THEN** frequently check the system for proper operation and system response. The system is considered inoperable.

⇓COM7.1.3

4.3.3 **WHEN** manual operation is no longer required, return systems to automatic or standby mode.

⇓COM7.1.3

4.3.4 Before placing controls in manual for activities which require manual control for an extended period of time, review system response and actions to be taken during potential off-normal events.

4.3.5 **IF** manual control of an automatic system is desired, **THEN** reset the initiation signal, if practicable. This will ensure the system returns to the design setpoint if the system automatically initiates.

4.4 Torus Water Temperature

IF torus water temperature reaches 120°F, **THEN** depressurize RPV to **LESS THAN** 200 psig at normal cooldown rates unless restrained by EOPs.

4.5 Containment Instrument Nitrogen

Use of containment instrument nitrogen for operation of components inside drywell, for example, MSIVs and SRVs, should take precedence over use of instrument air in order to maintain primary containment inerted during degraded plant conditions.

4.6 **RHR and Core Spray Operation**

⇓COM7.1.4

4.6.1 Blockage of ECCS pump suction strainers could occur due to debris in the Torus. Within the latitude provided by EOPs to restore and maintain parameters within specified limits, potential mitigative actions may include:

- Minimizing ECCS flow or removing affected ECCS pumps from service
- Alternating ECCS pumps from one division to another, if available
- Shifting ECCS pump suction to another source, if available
- Operation of alternate injection sources

4.6.2 Whenever RHR is in the LPCI mode, inject into RPV through RHR heat exchangers as soon as possible and establish RHRSW flow.

4.6.3 Secure RHR and core spray pumps that <u>are not</u> needed to support required actions of EOPs.

4.6.4 Diverting low pressure coolant injection to spray the containment <u>should not</u> be done unless adequate core cooling can also be maintained, or as directed per EOPs.

4.6.5 **WHEN** performing both EOP-2 and EOP-4, maintaining adequate core cooling normally takes precedence over maintaining containment parameters. Utilizing RHR flow for LPCI injection, containment spray, or torus cooling, singularly or in combination, is permissible provided continuous LPCI injection is not required for adequate core cooling.

4.6.6 **IF** drywell or torus hydrogen concentration cannot be determined to be below 6%,
AND drywell or torus oxygen concentration cannot be determined to be below 5%,
THEN operate sprays irrespective of adequate core cooling.

CAUTION

Elevated crescent area temperature affects RHR and core spray pump motor winding temperatures and could lead to motor failure.

4.6.7 **IF** an RHR or core spray pump motor winding temperature reaches alarm setpoint,
AND that pump is needed to support required actions of EOPs,
THEN consider reducing pump flow rate or using an alternate system.

4.7 Reactor Shutdown Determination

4.8 **Procedure Use While Performing EOPs**

4.8.1 AOP-1, Reactor Scram, immediate operator actions should be performed concurrently with initial entry into EOP-2.

4.8.2 AOP-1 subsequent operator actions, such as resetting the scram and balance of plant, should be performed concurrently with EOPs to aid in recovery. However, actions taken <u>shall not</u> contradict or subvert actions specified by the EOPs and <u>shall not</u> cause the loss or unavailability of equipment required by the EOPs.

4.8.3 AOP-39, Loss of Coolant, should be performed concurrently with applicable EOPs for events involving a loss of coolant inside the primary containment. However, actions taken per AOP-39 <u>shall not</u> contradict or subvert actions specified by the EOPs and <u>shall not</u> cause the loss or unavailability of equipment required by the EOPs.

4.8.4 AOP-40, Main Steam Line Break, should be performed concurrently with applicable EOPs for events involving a piping break in a main steam line outside the primary containment. However, actions taken per AOP-40 <u>shall not</u> contradict or subvert actions specified by the EOPs and <u>shall not</u> cause the loss or unavailability of equipment required by the EOPs.

4.8.5 Other plant procedures may be used in conjunction with EOPs to enhance emergency response and recovery. However, actions taken per other plant procedures <u>shall not</u> contradict or subvert actions specified by the EOPs and <u>shall not</u> cause the loss or unavailability of equipment required by the EOPs.

5.0 PROCEDURE

Monitor parameters using multiple indications specified by this procedure to determine actual status of parameter.

NOTE: Indications are listed in order of preference under each parameter.

5.1 RPV Control

- **RPV Water Level**
  - SPDS display
  - RX WATER LVL 02-3LI-85A and 02-3LR-85B at panel 09-5
  - Annunciator 09-5-1-31 RPS RX VESSEL LO LVL TRIP
  - RX WTR LVL FUEL ZONE 02-3LR-98 and 02-3LI-91 at panel 09-3
- **RPV Pressure**
  - SPDS display
  - RX VESSEL PRESS 06PI-61A and B, and 06PR-61A and B at panel 09-3
  - Annunciator 09-5-1-22 RPS HI RX PRESS TRIP
- **Reactor Power**
  - SPDS display
  - APRM chart recorders at panel 09-5
  - APRM meters at panel 09-14
  - IRM chart recorders at panel 09-5
  - IRM meters at panel 09-12
- **Reactor Shutdown**
  - Full Core Display FULL-IN green lights at Panel 09-5
  - SPDS Plant Display control rods full-in indication
  - Four rod display notch position indication at Panel 09-5
  - EPIC Full Core Rod Scan
  - EPIC Rods In Monitor Program (RIMP)

5.2 Primary Containment Control

- **Torus Temperature**
  - SPDS display
  - TORUS TEMP A 16-1TR-131A and TORUS TEMP B 16-1TR-131B at panel 09-3
  - Average of bay temperatures obtained individually at 16-1TI-131A and 16-1TI-131B at MAP panel
- **Drywell Temperature**
  - SPDS display
  - Average of temperatures on DW TEMP A 16-1TR-108 and DW TEMP B 16-1TR-107 at panel 09-3
  - Average air inlet and outlet temperature for any drywell cooling assembly that has at least one fan operating (68TI-100 or 68TI-101 at panel 09-75).
- **Drywell Pressure**
  - SPDS display
  - NR PC PRESS 27PI-115A1 and 27PR-115A1, and 27PI-115B1 and 27PR-115B1 at panel 09-3
  - Annunciator 09-5-1-21 RPS HI DW PRESS TRIP
  - WR PC PRESS 27PI-115A2 and 27PR-115A2, and 27PI-115B2 and 27PR-115B2 at panel 09-3
- **Torus Level**
  - SPDS display
  - TORUS LVL 23LI-202A and 23LR-202A, and TORUS LVL 23LI-202B and 23LR-202B at panel 09-3
- **Containment Hydrogen**
  - SPDS display
  - Panel 27PCX-101A and 27PCX-101B in Relay Room
  - Grab samples

5.3 **Secondary Containment Control**

- **Differential Pressure**

  - SPDS display

  - RB DIFF PRESS 01-125DPI-100A and 01-125DPI-100B at panel 09-75

- **Area Temperature High**

  - SPDS display

  - Panels 09-95, 09-96, 09-75, and 09-21, per Table 5-1 of EOP-5

  - **IF** an area <u>does not</u> have remote temperature indication, **THEN** monitor that area locally.

    IF that area is inaccessible,
    AND a primary system is discharging into Secondary Containment,
    THEN assume the area temperature is above the maximum safe level.

- **Reactor Building Radiation Levels**

  - SPDS display

  - ARM at panel 09-11

  - Local monitoring by Radiation Protection survey

  - **IF** the area is accessible
    AND the radiation levels in the area are not available from an ARM (ARM is either out of service or is not installed in the area),
    THEN locally monitor that area with a portable ARM.

    IF the area is inaccessible,
    AND the radiation levels in the area are not available from either the installed ARM or Radiation Protection survey,
    THEN assume the radiation levels in that area are above the maximum safe level.

- **Reactor Building Vent Exhaust**

  - SPDS display

  - RX BLDG VENT RAD MON A 17RIS-452A and RX BLDG VENT RAD MON B 17RIS-452B at panel 09-12

  - REFUEL FLOOR EXH RAD MONITOR 17RM-456A at panel 66HV-3A and REFUEL FLOOR EXH RAD MONITOR 17RM-456B at panel 66HV-3B (Reactor Building 272')

  - Annunciators 09-3-2-29 RX BLDG VENT RAD MON HI and 09-75-1-15 REFUELING FLOOR EXH RAD MON INOP OR HI

  IF Reactor Building is inaccessible,
  THEN assume Refuel Floor exhaust monitor is
  GREATER THAN 10^3 counts per minute.

- **Reactor Building Floor Sump Level**

  - SPDS display

  - Annunciators 25-17-1-1 RB FLR SUMP A LVL HI and 25-17-1-2 RB FLR SUMP B LVL HI at panel 25-17 in Radwaste Control Room

  - Local observation

  IF Reactor Building is inaccessible,
  THEN assume reactor building floor sump level is
  GREATER THAN high alarm setpoint.

- **Crescent Area Water Level**

  - SPDS display

  - Local observation

  IF Reactor Building is inaccessible,
  AND a primary system is discharging into Secondary Containment,
  THEN assume crescent area water level is
  GREATER THAN 18 inches.

5.4 Radioactivity Release Control

Offsite release rates and emergency classification are determined by Site Emergency Plan.

6.0 REFERENCES

6.1 Performance References

6.1.1 AOP-1, Reactor Scram

6.1.2 EOP-2, RPV Control

6.2 Developmental References

6.2.1 ODSO-28, Revision 4, EOP Entry and Use

6.2.2 EOP-2, RPV Control

6.2.3 EOP-3, Failure to Scram

6.2.4 EOP-4, Primary Containment Control

6.2.5 EOP-5, Secondary Containment Control

6.2.6 EOP-6, Radioactivity Release Control

6.2.7 JTS-95-0221, Operability Assessment for DER 95-0740 - 0748; Industry Notification of B-Fill Qualification Limits for Certain ITT-Barton Indicating Switches

6.2.8 JSED-95-0100, Impact of ITT Barton Industry Advisory on the Environmental Qualification of 10DPIS-125A&B, 14FIS-45A&B, and 27PS-110A&B

6.2.9 GE Letter JAB-N8075, dated 11/2/98, MSBWP Results for FitzPatrick Cycle 14 (GE letter 262-98-172 and DRF J11-03359)

6.2.10 JENG-02-0375, dated 10/12/2002, EFFECT OF JD-01-102 (CORE RELOAD) AND DECAY HEAT CURVE CHANGES ON EOP CURVES

7.0 REQUIREMENTS

7.1 Commitments

7.1.1 NRCI-94-03, JAFP-94-0175, ACTS Item 10946. Created EOP Support Procedures (EPs).

7.1.2 NRCN-92-47, Intentional Bypassing of Automatic Actuation of Plant Protective Features (OER 920483, JTS-92-0799)

7.1.3 ACTS Item 5899, incorporate INPO SER 87-34 (OER #870335).

7.1.4 JAFP-94-0228, Response to NRC Bulletin No. 93-02, Supplement 1, Debris Plugging of Emergency Core Cooling Suction Strainers. Added special instruction to alert operators of the potential for ECCS suction strainer clogging and to adjust flow consistent with required needs to mitigate the clogging.

7.2 Validation

Validated per AP-02.02.

8.0 ATTACHMENTS

1. MINIMUM CORE FLOODING INTERVAL

2. MAXIMUM CORE UNCOVERY TIME LIMIT

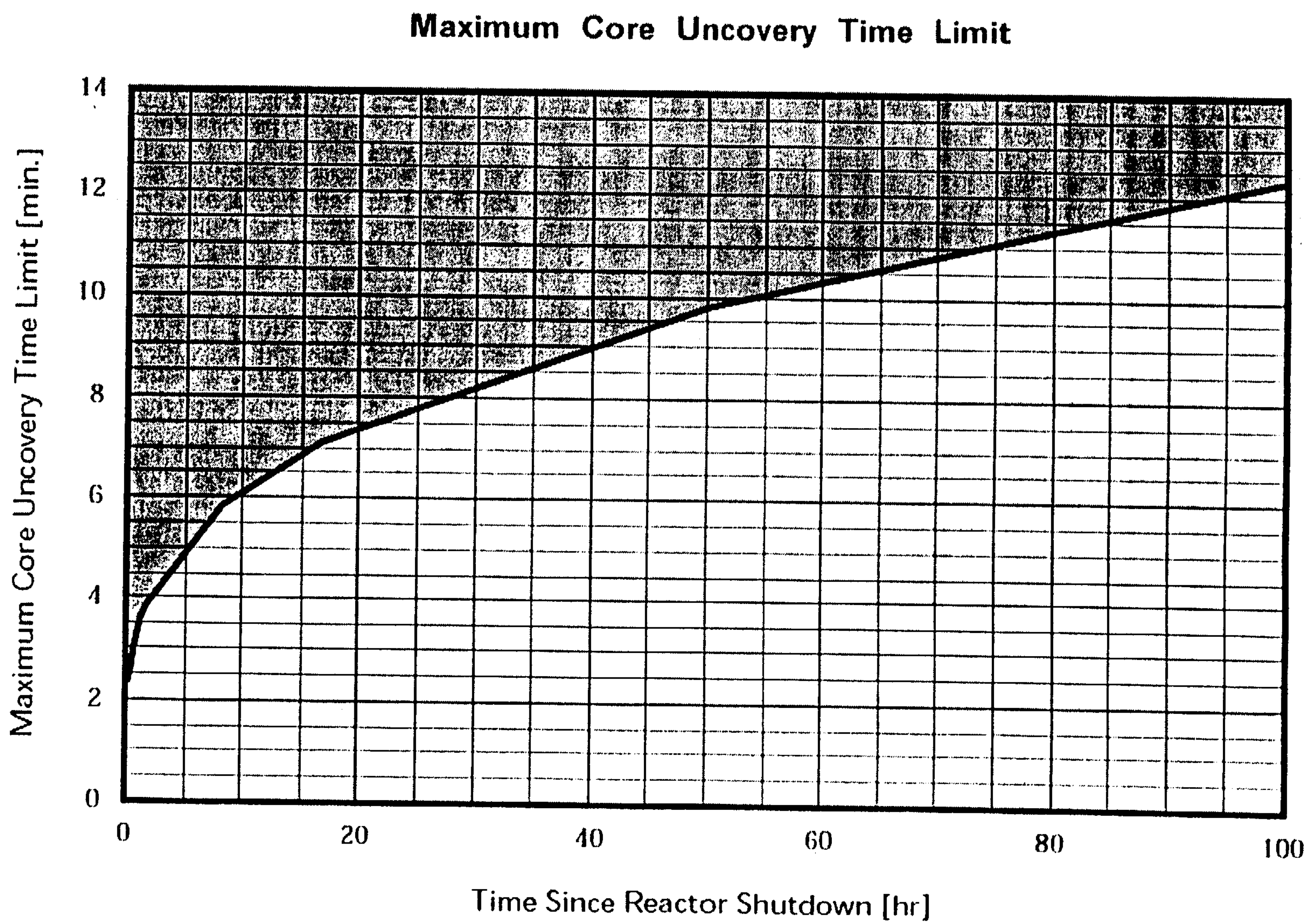
ATTACHMENT 1 Page 1 of 1

MINIMUM CORE FLOODING INTERVAL

Number of Open SRVs	Flooding Interval (min.)
7 or more	22
6	30
5	44

ATTACHMENT 2

MAXIMUM CORE UNCOVERY TIME LIMIT



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Low Suppression Pool Wtr Lvl / 5	Group #	1	1
Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12)	K/A # 295030	2.4.48	2.4.48
	Importance Rating	3.5	3.8

Proposed Question: While experiencing torus water level control problems, an operator opens an ADS valve with torus water level at 5.2 ft.

Opening the SRV under these conditions will result in:

RO/SRO

16/19

a) direct suppression chamber pressurization

b) excessive hydrodynamic loading

c) valve seat damage from the excessive flowrates.

d) drawing water up into the tailpipe.

Proposed Answer: a) direct suppression chamber pressurization

Explanation (Optional):

Technical Reference(s): EOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT 301.11E- EO 4.03 (As available)

Question Source: Bank # Dresden 1INPO # 6483 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/26/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 3

55.43 5

Comments:

Emergency Procedures / Plan

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Low Suppression Pool Wtr Lvl / 5	Group #	1	1
Ability to interpret control room indications to verify the status and operation of system / and understand how operator actions and directives affect plant and system conditions. (CFR: 43.5 / 45.12)	K/A # 295030	2.4.48	2.4.48
	Importance Rating	3.5	3.8

Proposed Question: While experiencing torus water level control problems, ~~the~~ an operator opens an ADS valve ~~while operating in the "unsafe" (cross-hatched) region of the ADS Valve Tail Pipe Level Limit curve.~~ with torus water level at 5.2 ft.

Opening the SRV under these conditions will result in:

RO/SRO 16/19

a) direct suppression chamber pressurization
b) excessive hydrodynamic loading
c) valve seat damage from the excessive flowrates.
d) drawing water up into the tailpipe.

Proposed Answer: ~~b) excessive hydrodynamic loading~~ a) direct suppression chamber pressurization

Explanation (Optional): **NO LINK to 2.4.48 was linked to 2.4.18**

Technical Reference(s): MET 301.11e 4.0.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank # Dresden 1INPO # 6483

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/26/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Reactor Low Water Level / 2	Group #	1	1
Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : (CFR: 41.5 / 45.6) Automatic depressurization system actuation	K/A # 295031	EK3.01	EK3.01
	Importance Rating	3.9	4.2

Proposed Question: From full power and with HPCI inoperable, a SCRAM occurs from a small primary leak in the drywell simultaneous with a loss of offsite power. EDG's start and reenergize vital busses. RCIC initiates, but insufficient injection results in RPV water level continuing to lower.

Which of the following is correct assuming **NO** operator action?

RO/SRO

17/20

a) SRV's should cycle open on their automatic pressure relief setpoints and lower reactor pressure to permit level recovery injection with the Condensate system.

b) SRV's assigned to ADS should open when RPV level lowers to an assigned setpoint to permit level recovery injection with low pressure ECCS.

c) A residual bus transfer will result in automatic start and injection by the Condensate Booster Pumps.

d) SRV's should cycle on their automatic pressure relief setpoints and together with the reduced RCIC injection will provide steam cooling with injection.

Proposed Answer: b) SRV's assigned to ADS should open when RPV level lowers to an assigned setpoint to permit level recovery injection with low pressure ECCS.

Explanation (Optional):

Technical Reference(s): OP-68 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02J, EO-1.01, 1.05.A, 1.05.C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	Group #	1	1
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.10 / 43.5 / 45.13) Reactor water level	K/A # 295037	EA2.02	EA2.02
	Importance Rating	4.1	4.2

Proposed Question: As directed by EOP-3, the current RPV level band is –19 to 110 inches and being controlled at 80-100 inches with Feedwater.

Which of the following is the preferred instrumentation for maintaining the 80-100 inches band?

RO/SRO 18/21

a) Narrow Range.
b) Wide Range.
c) Refuel Zone.
d) Fuel Zone.

Proposed Answer: b) Wide Range.

Explanation (Optional):

Technical Reference(s): SDLP-02B, Table IV (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-02B, EO-1.05.A.3 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Off-site Release Rate / 9	Group #	1	1
Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: (CFR: 41.7 / 45.8) Offgas system	K/A # 295038	EK2.02	EK2.02
	Importance Rating	3.6	3.8

Proposed Question: While operating at full power, a large fuel leak develops.

Which of the following automatic responses from a high radiation signal will occur to limit off-site release rates?

RO/SRO 19/22

a) Condenser Vacuum Pump trip.
b) Off gas System isolation.
c) Off gas Recombiner trip.
d) Reactor SCRAM.

Proposed Answer: b) Off gas System isolation.

Explanation (Optional):

Technical Reference(s): OP-24A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01A, EO-1.05.C.1 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Plant Fire On Site / 8	Group #	1	1
Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm	K/A # 600000	AA1.06	AA1.06
	Importance Rating	3.0	3.0

Proposed Question: With the Fire Protection System in a normal standby lineup, which one (1) of the following Fire Protection Panel Alarms will always be accompanied by the start of one or more Fire Pumps?

RO/SRO
20/23

a) Heat detection actuation in the West Cable Tunnel
b) Heat detection actuation in the North EDG Switchgear Room
c) Ionization detector actuation in the Reactor Building 272' Drywell Entrance
d) Ultraviolet Flame detector in the Recirculation M/G Room

Proposed Answer: a) Heat detection actuation in the West Cable Tunnel

Explanation (Optional):

Technical Reference(s): OP-33 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-76 EO 1.05c (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Loss of Main Condenser Vac / 3	Group #	2	2
Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : (CFR: 41.10 / 43.5 / 45.13) Condenser vacuum/absolute pressure	K/A # 295002	AA2.01	AA2.01
	Importance Rating	2.9	3.1

Proposed Question: Reactor power is 38%, on the APRM's when annunciator 09-6-1-29, CONDENSER VAC LOW, alarms. Condenser vacuum, as read on control room meters, indicates 24.8" and lowering slowly. If vacuum continues to lower, WHICH ONE (1) of the following automatic protective actions would occur first?

RO/SRO 21/24

a) Reactor Feed Pump Turbine Trip
b) Main Turbine Trip
c) Bypass Valve Closure
d) MSIV Closure

Proposed Answer: b) Main Turbine Trip

Explanation (Optional):

Technical Reference(s): OP-9, OP-2A, OP-1, AOP-31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP, EO-1.02 (As available)

Question Source: Bank # Nine Mile Point 1 INPO # 11813 (Modified to JAF)

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 1/20/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

-09-6-1-29, CNDNSR VAC LO,

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Loss of Main Condenser Vac / 3	Group #	2	2
Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : (CFR: 41.10 / 43.5 / 45.13)	K/A # 295002	AA2.01	AA2.01
Condenser vacuum/absolute pressure			
	Importance Rating	2.9	3.1

Proposed Question: Reactor power is 38%, on the APRM's when annunciator ~~A1-3-4, CONDENSER VACCUM BELOW 24"HG~~, alarms. Condenser vacuum, as read on control room meters, indicates 24.8" ~~23.5"~~ and lowering slowly. If vacuum continues to lower, WHICH ONE (1) of the following automatic protective actions would occur first?

RO/SRO 21/24

a) ~~Turbine bypass valves close~~ Reactor Feed Pump Turbine Trip
b) Main Turbine Trip
c) Reactor Scram
d) MSIV Closure

Proposed Answer: b) Main Turbine Trip

Explanation (Optional): ~~No KA for this question identified 295002.AA1.05~~

Technical Reference(s): OP-9, OP-2A, OP-1, AOP 31 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: LEAR 1. 2 (As available)

Question Source: Bank # Nine Mile Point 1 INPO # 11813

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/20/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Reactor Pressure / 3	Group #	2	2
Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: (CFR: 41.7 / 45.8) LPCS	K/A # 295007	AK2.04	AK2.04
	Importance Rating	3.2	3.3

Proposed Question: An Emergency Depressurization is to be performed from 700 psig to permit low pressure ECCS injection into the reactor. The only ECCS available is Core Spray System A. CS Pump A is running on minimum flow and all other components are in a normal standby condition. When SRV's are operated, only one (1) SRV responds. Reactor pressure lowers at approximately 10 psi/minute.

The Core Spray Injection Valve opens when reactor pressure goes below ________.

RPV injection ______________ immediately.

RO/SRO

22/25

a) 450 psig: occurs

b) 450 psig: does **NOT** occur

c) 310 psig: occurs

d) 310 psig: does **NOT** occur

Proposed Answer: b) 450 psig: does **NOT** occur

Explanation (Optional):

Technical Reference(s): OP-14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14, EO-1.13e, 1.14b (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Inadvertent Reactivity Addition / 1	Group #	2	2
Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8) Reactor power	K/A # 295014	AK2.07	AK2.07
	Importance Rating	3.9	3.9

Proposed Question: From normal full power operation, which of the following **WILL** cause reactor power to rise?

RO/SRO

23/26

a) Inadvertently isolating the Reactor Water Cleanup System.

b) Removing local RPS fuses for a control rod hydraulic control unit.

c) Main Condenser Circulating Water Pump Trip.

d) Closing the manual extraction steam valve for Feed Heater 6B.

Proposed Answer: d) Closing the manual extraction steam valve for Feed Heater 6B.

Explanation (Optional): Explanation: The manual extraction steam valve for Feed Heater 6B closing will prevent the heating of the feedwater in the 6B heater, thereby, causing colder feedwater to enter the vessel and drive reactor power up.

Technical Reference(s): AOP-62, AOP-32, OP-3A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP EO 1.02 (As available)

Question Source: Bank # Clinton INPO # 20412 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 7/23/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

a) Inadvertent isolation of the Reactor Water Cleanup System

c) Adjusting T-4 tap changer to lower 4160KV bus voltage by 110 volts

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Inadvertent Reactivity Addition / 1	Group #	2	2
Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: (CFR: 41.7 / 45.8)	K/A # 295014	AK2.07	AK2.07
Reactor power			
	Importance Rating	3.9	3.9

From normal full power conditions

Proposed Question: Which of the following would cause reactor power to ~~go up~~? rise

RO/SRO

23/26

a) ~~RR Flow Control Valve closing.~~ A control Rod Scram Outlet Valve opening. HCU

b) Rod Scram Outlet Valve opening.

c) ~~CD Pump Minimum Flow Valve opening~~

d) 6B Extraction Steam ~~Shutoff~~ Valve closing. Feedwater Heater Non-Return

Proposed Answer: d) 6B Extraction Steam ~~Shutoff~~ Valve closing.

Explanation (Optional): Explanation: The 6B Extraction Steam Shutoff Valve closing will prevent the heating of the feedwater in the 6B heater, thereby, causing colder feedwater to enter the vessel and drive reactor power up. **NO KA AK 2.07exists and is tied- make a tie (was tied to 295014.aa2.01**

Technical Reference(s): AOP-32 ; OP-3A ; AOP-62 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: LP-AOP 1.02 (As available)

Question Source: Bank # Clinton INPO # 20412

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 7/23/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Refueling Acc Cooling Mode / 8	Group #		1
Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS :(CFR: 41.8 to 41.10) Radiation exposure hazards Also 10CFR-55.43(b)(4)	K/A # 295023		AK1.01
	Importance Rating		4.1

Proposed Question: Core Alterations are in progress.

An irradiated fuel bundle being moved from the reactor cavity to the Spent Fuel Pool becomes ungrappled and falls into the reactor vessel downcomer area. (Between the vessel wall and the shroud)

Which of the below describes the person at greatest risk?

RO/SRO

S27

a) Mechanic working on Torus to Drywell Vacuum Breaker.

b) Refuel SRO on the Bridge

c) I&C Technician at SLC Skid.

d) Mechanic working on SRVs

Proposed Answer: d) Mechanic working on SRVs

Explanation (Optional):

Technical Reference(s): RAP-7.1.1.04B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP, RAP-7.1.04B73.03 (As available)

Question Source: Bank # Clinton INPO # 20401 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 7/23/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 9

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Refueling Acc Cooling Mode / 8	Group #		1
Knowledge of the operational implications of the following concepts as they apply to REFUELING ACCIDENTS :(CFR: 41.8 to 41.10) Radiation exposure hazards Also 10CFR 55.43(b)(4)	K/A # 295023		AK1.01
	Importance Rating		4.1

Proposed Question: Core Alterations are in progress.

An irradiated fuel bundle being moved from the reactor cavity to ~~IFTS~~ Spent Fuel Pool becomes ungrappled and falls into the reactor vessel downcomer area. (Between the vessel wall and the shroud)

Which of the the following people would be at greatest risk of radiation overexposure?

RO/SRO
S27

a) Operator ~~in Fuel Building 755' el.~~ at a Drywell to Torus vacuum breaker

b) Refuel SRO on the Bridge

c) ~~IM~~ I & C Technician at SLC Skid.

d) Mechanic working on SRVs

Proposed Answer: d) Mechanic working on SRVs

Explanation (Optional): 'Explanation: Correct Answer - Figure 11 of LP85449 shows the general location of each person. The mechanic would have the greatest risk because he could move into an area with very little shielding between him and the dropped fuel bundle. The other personnel have either large quantities of water or concrete as shielding. **MODIFY to reflect NO IFTS and choose locations- ties to 10CFR 55.43(b)(4) well.**

Technical Reference(s): RAP 7.1.04B (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP RAP 7.1.04B 73.03 (As available)

Question Source: Bank # Clinton INPO # 20401 (mod)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 7/23/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 9

55.43 (b)(4) 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Secondary Containment Area Temperature / 5	Group #	2	2
Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10)	K/A # 295032	2.3.10	2.3.10
	Importance Rating	2.9	3.3

Proposed Question: Which one of the following describes an EOP-5, "Secondary Containment Control," basis for isolating a primary system discharging into the secondary containment?

RO/SRO
24/28

a) To minimize RPV inventory losses.

b) To backup PCIS automatic functions.

c) To terminate rising radiation levels.

d) To ensure Recirculation M/G Room access.

Proposed Answer: c) To terminate rising radiation levels.

Explanation (Optional): Secondary Containment Control does not maintain habitability for all areas. The Max Safe values are based on equipment operability and personnel access necessary for EOP actions. The Recirc MG set room is not one of the areas requiring access.

Technical Reference(s): EOP-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP5LP, EO-1.07 (As available)

Question Source: Bank # Cooper 1 INPO # 302 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 2/12/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43 4

Comments:

T- 2.3.10 ... M/G Room access

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Secondary Containment Area Temperature / 5	Group #	2	2
Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10)	K/A # 295032	2.3.10	2.3.10
	Importance Rating	2.9	3.3

Proposed Question: Which one of the following describes ~~the~~ an EOP-5~~A~~, "Secondary Containment Control," basis for isolating a system discharging into the secondary containment?

primary

RO/SRO

24/28

a) To minimize RPV inventory losses.

b) To backup PCIS automatic functions.

c) To terminate rising ~~temperatures~~, radiation levels, ~~and water levels~~.

d) ~~To maintain the Recirc MG set room accessible to personnel~~.

Proposed Answer: c) To terminate rising ~~temperatures~~, radiation levels, ~~and water levels~~.

Explanation (Optional): Secondary Containment Control does not maintain habitability for all areas. The Max Safe values are based on equipment operability and personnel access necessary for EOP actions. the Recirc MG set room is not one of the areas requiring access. **NO KA Tie to 2.3.10- was tied to 295032.K3.03**

Technical Reference(s): EOP-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: INT 321.NE 1.07 (As available)

Question Source: Bank # Cooper 1 INPO # 302

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 2/12/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High Secondary Containment Area Radiation Levels / 9	Group #	2	2
Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : (CFR: 41.7 / 45.6) Affected systems so as to isolate damaged portions	K/A # 295033	EA1.05	EA1.05
	Importance Rating	3.9	4.0

Proposed Question: During power operation, the Area Radiation Monitor (ARM) for the CRD Removal Hatch area alarms, together with receipt of a Fire Protection System ionization detector alarm in the Southwest Drywell Entrance Area.

Which system(s) should be considered for manual isolation?

RO/SRO

25/29

a) HPCI and RWCU

b) RCIC and Main Steam

c) Main Steam and RWCU

d) HPCI and RCIC

Proposed Answer: d) HPCI and RCIC

Explanation (Optional):

Technical Reference(s): EOP-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP5LP, EO-1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
High Reactor Pressure / 3	Group #		1
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: (CFR: 41.10 / 43.5 / 45.13) Decay heat generation	K/A # 295025		EA2.05
	Importance Rating		3.6

Proposed Question: The plant is starting up after an extended (60 day) outage. At 15% power, a complete loss of EHC results in a manual Reactor SCRAM.

Which of the following describes the expected procedural actions?

RO/SRO

S30

a) Ensure one (1) Bypass Valve opens to control RPV pressure per EOP-2, RPV Control.

b) Startup HPCI and control RPV pressure, 900-1050 psig, per AOP-1, Reactor SCRAM.

c) Reduce Secondary Plant steam loads to control RPV pressure, 900-1050 psig, per AOP-1, Reactor SCRAM.

d) Open one (1) SRV and control RPV pressure, 800-1000 psig, per EOP-2, RPV Control.

Proposed Answer: c) Reduce Secondary Plant steam loads to control RPV pressure, 900-1050 psig, per AOP-1, Reactor SCRAM.

Explanation (Optional):

Technical Reference(s): AOP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP, EO-1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
Secondary Containment High Sump/Area Water Level / 5	Group #	2	2
Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : (CFR: 41.5 / 45.6) Emergency depressurization	K/A # 295036	EK3.01	EK3.01
	Importance Rating	2.6	2.8

Proposed Question: While operating at full power, an earthquake has resulted in the following:

- A severe piping crack between the CST's and the Torus results in a rapid addition of water to the Torus Room and both Crescent Areas.
- A small, un-isolable leak in the RWCU Pump suction piping in the Reactor Building.
- Crescent Area water levels are 19" rising
- Highest Reactor Building Area (RB 300' Southwest) temperature is 103°F

Why must an Emergency Depressurization be performed for these conditions?

RO/SRO

26/31

a) A loss of CST inventory will result in total loss of HPCI and RCIC for inventory control.

b) Operability of equipment located in the Crescents is threatened by Crescent water level rise.

c) Primary Containment integrity is threatened by Torus Room water level rise.

d) Operability of RPV Water Level instruments located on Reactor building 300' is challenged.

Proposed Answer: b) Operability of equipment located in the Crescents is threatened by Crescent water level rise.

Explanation (Optional):

Technical Reference(s): EOP-5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP's

Learning Objective: EOP5LP, EO-1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
High CTMT Hydrogen Conc. / 5	Group #	2	2
Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: (CFR: 41.7 / 45.6) Drywell sprays	K/A # 500000	EA1.06	EA1.06
	Importance Rating	3.3	3.4

Proposed Question: Which of the following requires initiation of Drywell Sprays?

RO/SRO 27/32

	DW H^2	DW O^2	Torus H^2	Torus O^2
a)	6.03 %	5.4 %	6.13 %	3.0 %
b)	6.13 %	3.0 %	6.03 %	5.4 %
c)	5.9 %	6.03 %	6.13 %	3.0 %
d)	3.0 %	6.03 %	6.13 %	5.4 %

Proposed Answer: a) 6.03 % 5.4 % 6.13 % 3.0 %

Explanation (Optional):

Technical Reference(s): EOP-4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP-4A, Primary Containment Gas Control

Learning Objective: EOP4LP, EO-4.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7, 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: Injection Mode	Group #	1	1
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) Suppression pool level	K/A # 203000	A1.05	A1.05
	Importance Rating	3.8	3.7

Proposed Question: A Design Basis LOCA has occurred. ECCS systems are injecting into the reactor. Suppression Pool Level is at 12.8 feet and lowering.

Which one of the following would be the expected response of the Low Pressure Coolant Injection (RHR)?

RO/SRO

28/33

a) The RHR pumps will continue to operate regardless of Suppression Pool Level until the pumps trip on motor overload.

b) The RHR pumps will automatically trip when Suppression Pool Level drops to the RHR Pump Vortex Limit.

c) The RHR pumps will continue to operate regardless of Suppression Pool Level due to automatic bypass of all trip signals.

d) The RHR Pump Suppression Pool Suction valves will automatically close. The RHR pumps will all trip on Interlock.

Proposed Answer: a) The RHR pumps will continue to operate regardless of Suppression Pool Level until the pumps trip on motor overload.

Explanation (Optional):

Technical Reference(s): OP-13A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO-1.10.f (As available)

Question Source: Bank # Grand Gulf 1 INPO # 16342 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/1/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: Injection Mode	Group #	1	1
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI INJECTION MODE (PLANT SPECIFIC) controls including: (CFR: 41.5 / 45.5) Suppression pool level	K/A # 203000	A1.05	A1.05
	Importance Rating	3.8	3.7

Proposed Question: A DBA LOCA has occurred. ECCS systems are injecting into the reactor. Suppression Pool Level is at ~~14.8~~ feet and lowering. ~~Suppression Pool Makeup has failed to actuate.~~ Which one of the following would be the expected response of the Low Pressure Coolant Injection (RHR)?

A leak occurs in the

Level

12.8 feet

RO/SRO

28/33

a) The RHR pumps will continue to operate regardless of Suppression Pool Level until the pumps trip on motor overload.

b) The RHR pumps will ~~ALL~~ automatically trip when Suppression Pool Level drops to ~~14.5 feet~~ ~~which is the vortexing limit~~. Vortex Limit the RHR Pumps

c) The RHR pumps will ~~sequentially trip starting with the 'C' RHR pump on low discharge flow as a result of cavitation~~ continue to operate regardless of suppression pool level due to automatic bypass of all trip signals.

all automatically

d) The RHR pumps will ~~ALL~~ close their Suppression Pool Suction valves ~~and trip~~ the pumps due to NO suction flowpath. will trip

Proposed Answer: a) The RHR pumps will continue to operate regardless of Suppression Pool Level until the pumps trip on motor overload.

Explanation (Optional): **NO KA Exists- Was tied to 203000.K1.02 Modify question to torus vs SP.**

Technical Reference(s): OP-13A (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: SDLP.10 1.10.f (As available)

Question Source: Bank # Grand Gulf 1 INPO # 16342

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/1/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Low Suppression Pool Wtr Lvl / 5	Group #		1
Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL :(CFR: 41.10 / 43.5 / 45.13) Drywell/ suppression chamber differential pressure: Mark-I&II	K/A # 295030		EA2.04
	Importance Rating		3.7

Proposed Question: Two hours into the shift, the SNO reports that Torus water level has dropped from 14.0 ft to 13.91 ft while Drywell to Torus ΔP has dropped from 1.8 psid to 1.6 psid and Torus pressure has remained constant at 0.0 psig. You have confirmed the indications on EPIC-LOG1.

Your actions should be……….

RO/SRO

S34

a) Enter EOP-4, Primary Containment Control, and immediately makeup nitrogen to the Drywell restore ΔP.

b) Enter EOP-4, Primary Containment Control, and immediately makeup water to the Torus to restore Torus level.

c) Enter AOP-9, Loss of Primary Containment Integrity, and dispatch Operators to investigate the cause.

d) Enter AOP-9, Loss of Primary Containment Integrity, and dispatch Operators to locate the leaking Drywell to Torus Vacuum Breaker.

Proposed Answer: c) Enter AOP-9, Loss of Primary Containment Integrity, and dispatch Operators to investigate the cause.

Explanation (Optional):

Technical Reference(s): OP-37 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16B, EO-1.09d (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	Group #		1
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.10 / 43.5 / 45.13) Control rod position	K/A # 295037		EA2.05
	Importance Rating		4.3

Proposed Question: A failure to SCRAM occurs from a full power MSIV closure.

- Control Rods were inserted by draining the SDIV and using several manual SCRAMS.
- It is now believed that the Reactor will remain shutdown under all conditions without boron.

How can this be confirmed AND what actions will result?

RO/SRO

S35

a) Green "Full In" Lamps on Full Core Display. Secure SLC injection and enter EOP-2, RPV Control.

b) EPIC Full Core Rod Scan. Per EOP-2, RPV Control, secure SLC injection.

c) EPIC Solomon Program. Secure SLC injection per EOP-3, Failure to SCRAM, then, enter EOP-2, RPV Control.

d) Rods In Monitoring Program (RIMP). Verify Hot Shutdown Boron Weight and enter EOP-2, RPV Control.

Proposed Answer: a) Green "Full In" Lamps on Full Core Display. Secure SLC injection and enter EOP-2, RPV Control.

Explanation (Optional):

Technical Reference(s): EP-1, AOP-1, EOP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP, EO-1.03, EOP3LP, EO-1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: Injection Mode	Group #	1	1
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Indicating lights and alarms	K/A # 203000	A4.11	A4.11
	Importance Rating	3.7	3.5

Proposed Question: LPCI has automatically initiated due to RPV water level lowering below 59.5". The LPCI inboard and outboard injection valves are open and RPV pressure is 200 psig and lowering. EPIC is unavailable and RHR system flow indications at panel 09-3 are out of service. Which of the following indications could be used to help verify that LPCI is injecting water into the RPV?

	RHR PUMP MTR AMPS	RHR PUMP DISC PRESS	10MOV-16A(B) POSITION INDICATION
a)	lowering	rising	closed
b)	rising	lowering	closed
c)	rising	rising	open
d)	lowering	rising	open

RO/SRO

29/36

Proposed Answer: b) rising lowering closed

Explanation (Optional):

Technical Reference(s): OP-13A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO 1.05.a.1.b (As available)

Question Source: Bank # JAF LOR 20505001RHRC19

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Shutdown Cooling	Group #	1	1
Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: (CFR: 41.7 / 45.4) Recirculation loop temperatures	K/A # 205000	K3.04	K3.04
	Importance Rating	3.7	3.7

Proposed Question: The Plant is in Mode 4 with both Recirculation Pumps secured. Shutdown cooling is in service using RHR System 'A'. Reactor water level is steady at 198". Coolant temperature is 140° F with a slow cool down in progress. A loss of Shutdown cooling occurs.

Which one (1) of the following responses will provide for reliable Reactor Coolant temperature indication?

RO/SRO 30/37

a) Opening Recirculation Pumps suction and discharge valves.

b) Raising reactor water level to ≥ 234.5".

c) Securing the Reactor Water Cleanup System.

d) Placing the Control Rod Hydraulic System in service.

Proposed Answer: b) Raising reactor water level to ≥ 234.5".

Explanation (Optional):

Technical Reference(s): OP-13D, AOP-30, ITS Definitions (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AOP, EO-1.03, 1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Reactor Low Water Level / 2	Group #		1
Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) ***Alternate injection systems: Plant Specific***	K/A # 295031		EA 1.08
Link to 10CFR-55.43(b)(1-6)	Importance Rating		3.9

Proposed Question:

A startup is in progress at 20% CTP when an RPS electrical malfunction results in the following:

- HPCI/RCIC & MSIV Isolation on High Temperature
- Full Reactor SCRAM
- One (1) rod remains at position 40 and one (1) other rod is at position 02. All other rods are Full In.
- RPV water level is 150 inches, slowly trending down.
- RPV pressure is 1000 psig, slowly trending up.

The correct course of action is to:

RO/SRO

S38

a) Enter EOP-3, stabilize RPV pressure, and maintain RPV level with Feed/Condensate.

b) Enter EOP-2, commence a normal cooldown, and maintain RPV level with Feed/Condensate.

c) Enter EOP-3, commence a normal cooldown, and maintain RPV level with SLC/CRD.

d) Enter EOP-2, Emergency Depressurize, and maintain RPV level with SLC/CRD.

Proposed Answer: b) Enter EOP-2, commence a normal cooldown, and maintain RPV level with Feed/Condensate.

Explanation (Optional):

Technical Reference(s): EOP-2, EP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP2LP, EO-1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
HPCI	Group #	1	1
Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10)	K/A # 206000	2.3.1	2.3.1
	Importance Rating	2.6	3.0

Proposed Question: While operating at full power, which one (1) of the following would lower the dose rate for an Operator during a twenty (20) minute walk down of the HPCI Pump and Turbine during a Post Work Test?

a) Lowering the injection rate.

RO/SRO
31/39

b) Operating HPCI from it's Torus suction.

c) Operating RHR System 'A' for Torus cooling.

d) Starting additional Crescent Coolers.

Proposed Answer: a) Lowering the injection rate.

Explanation (Optional):

Technical Reference(s): OP-15, Step C.2.9, AP-07.03 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23, EO-1.13.A, LPAP-28.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 12

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Inadvertent Reactivity Addition / 1	Group #		2
Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION : (CFR: 41.8 to 41.10) Prompt critical Also 10CFR-55.43(b)(6)	K/A # 295014		AK1.01
	Importance Rating		3.8

Proposed Question: Currently the plant is in the startup mode with control rods being withdrawn to bring the Reactor critical. The selected control rod is two (2) notches from the ECP's predicted criticality when a control rod drop occurs. The control rod blade that dropped went from position 4 to 48.

Assuming no further Operator action, which of the following barriers are in place to prevent this type of event AND what is a potential impact?

RO/SRO

S40

a) ST-20A, Rod Worth Minimizer Functional Test, the Reactor will heat up until α T turns power.

b) ST-20A, Rod Worth Minimizer Functional Test, the Reactor will go critical until full SCRAM on IRM HI-HI trip.

c) ST-23B, Control Rod Coupling Integrity Test, the Reactor will heat up until α T turns power.

d) ST-23B, Control Rod Coupling Integrity Test, the Reactor will go critical until full SCRAM on IRM HI-HI trip.

Proposed Answer: d) ST-23B, Control Rod Coupling Integrity Test, the Reactor will go critical until full SCRAM on IRM HI-HI trip.

Explanation (Optional):

Technical Reference(s): ST-23B, FSAR-14.5.4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03F, EO-1.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Loss of CRD Pumps / 1	Group #		2
Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : (CFR: 41.10 / 43.5 / 45.13) CRD system status	K/A # 295022		AA2.02
	Importance Rating		3.4

Proposed Question: While at full power, the following alarms and indications are received:

09-5-1-9, CRD CHARGING WTR PRESS LO, is in alarm

03PI-302, CHG WTR PRESS, indicates 0 psig.

03PDI-303, DRV WTR DIFF PRESS, indicates 0 psid.

03FI-306, CLG WTR FLOW, indicates 0 gpm.

03FI-310, CRD FLOW CNTRL, indicates 1 gpm.

Which of the following is the cause and the appropriate mitigating procedure?

RO/SRO

S41

a) 03CRD-56, CRD Charging Water Supply Header Isolation Valve, has been closed, ARP-09-5-1-9, CRD Charging WTR Press Lo.

b) 03FCV-19A(B), in-service CRD Drivewater Flow Control Valve, has failed closed, AOP-69, Control Rod Drive Trouble.

c) 03 MOV-22, CRD Cooling Water Pressure Control Valve, has been closed, ARP-09-5-1-9, CRD Charging WTR Press Lo.

d) 03P-16A(B), in-service CRD Drive Water Pump has failed, AOP-69, Control Rod Drive Trouble.

Proposed Answer: d) 03P-16A(B), in-service CRD Drive Water Pump has failed, AOP-69, Control Rod Drive Trouble.

Explanation (Optional):

Technical Reference(s): OP-25, AOP-69 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C, EO-1.12.B (As available)

Question Source: Bank # Fermi 2 INPO # 8900 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/6/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Loss of CRD Pumps / 1	Group #		2
Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : (CFR: 41.10 / 43.5 / 45.13)	K/A # 295022		AA2.02
CRD system status			
	Importance Rating		3.4

Proposed Question:

The plant is operating at full power. Several annunciators have been received in the last few minutes, including 3D5, CRD Charging H20 Pressure Low, and 3D10, CRD Accumulator Trouble. The following information is available:

E41-R609, HPCI Pump Suction Pressure Indicator, indicates 0 psig.

C11-PDIS-N002, CRD Drive Water ~~Filter~~ Differential Pressure, indicates 0 psig.

C11-R603, Cooling Water to Reactor Differential Pressure Indicator, indicates 0 psid.

C11-R800, CRD Hydraulic Flow Ind, indicates 1 gpm.

modified

Which one of the following is the reason annunciators 3D5 and 3D10 were received?

RO/SRO

S41

a) C11-F034, Charging Header Isolation Valve, has been closed.

b) The in-service CRD flow control valve has failed closed.

c) C1152-F003, ~~CRD~~ Drive/Cooling Water PCV, has been closed.

d) The operating CRD pump has tripped.

Proposed Answer: d) The operating CRD pump has tripped.

Explanation (Optional):

Technical Reference(s): ____________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ____________

Learning Objective: ____________ (As available)

Question Source: Bank # ~~Fermi 2 INPO # 8900~~

Modified Bank # ____________ (Note changes or attach parent)

New ____________

Question History: Last NRC Exam 4/6/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis ____________

10 CFR Part 55 Content: 55.41 ____________

55.43 ____________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Secondary Containment High Differential Pressure / 5	Group #		2
Knowledge of the process for performing a containment purge. (CFR: 43.4 / 45.10)	K/A # 295035		**2.3.9**
	Importance Rating		3.4

Proposed Question: While at full power, a unisolable leak has developed in the RWCU suction piping in the Reactor Building. Secondary Containment pressure has risen due to the leak into the Reactor Building but is still slightly negative.

Which of the following will minimize the radiation hazard and control the Secondary Containment pressure?

a) Initiate SGT System and manually isolate Reactor Building Ventilation.

RO/SRO

b) Ensure that SGT starts and Reactor Building Ventilation isolated when High ΔP Setpoint is reached.

S42

c) Place all Crescent Area Unit Coolers in service.

d) Operate RWCU in the Blowdown Mode to the Main Condenser.

Proposed Answer: a) Initiate SGT System and manually isolate Reactor Building Ventilation.

Explanation (Optional):

Technical Reference(s): OP-20, OP-51A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B, EO-1.14.E (As available)

Question Source: Bank # JAF LOR 20005214B01C

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 12

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Secondary Containment High Sump/Area Water Level / 5	Group #		2
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Operability of components within the affected area.	K/A # 295036		EA2.01
	Importance Rating		3.2

Proposed Question: Thirty (30) minutes after an earthquake, the following conditions exist:

- RPV water level is 185 inches increasing
- RPV Pressure is 1000 psig increasing
- All control rods are at position 00, except for one (1) rod at position 22
- One (1) foot of water is on the Crescent Floors due to a leaking Torus drain flange.
- The MSIVs are Closed.
- Reactor Scram has been reset.
- Torus level is 10.75 feet and slowly lowering.

Regarding the above conditions, which of the following is True?

a) EOP-3 action is based upon ensuring Reactor remains shutdown without Boron Injection.

RO/SRO

S43

b) EOP-4 action is based upon preserving HPCI Injection capability.

c) EOP-5 action is based upon a loss of the Core Spray Hold Pumps.

d) There is NO EOP entry condition. Plant is controlled by AOP-1.

Proposed Answer: c) EOP-5 action is based upon a loss of the Core Spray Hold Pumps.

Explanation (Optional):

Technical Reference(s): EOP-5, OP-14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11, EO-1.07 (As available)

Question Source: Bank # Monticello 1 INPO # 15350 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/23/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
Secondary Containment High Sump/Area Water Level / 5	Group #		2
Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13) Operability of components within the affected area.	K/A # 295036		EA2.01
	Importance Rating		3.2

Proposed Question: Thirty (30) minutes after an earthquake, the following conditions exist: Reactor water level is normal.

- All control rods are at position 00. crescent floors
- One (1) foot of water is on the ~~Torus floor~~ due to a leaking Torus drain flange.
- The MSIVs are Closed.
- Reactor Scram has been reset.
- ~~RPS Busses A & B amber lights are on~~.
- Torus level is ~~2 feet.~~ 10.75 ft and slowly lowering

In carrying out EOP actions For the above conditions, which ~~of the following is correct?~~ one of the following components may be adversely affected?

RO/SRO
S43

a) ~~Water will be flooding the 250 VDC MCC-313.~~ Core Spray HDD Pumps
b) ~~RHR and Core Spray pumps are unavailable.~~ Reactor Building to Torus Vacuum Breakers
c) ~~Full core display "Blue" Scram lights are ON.~~ Crescent Area Coolers
d) ~~RHR Room ECCS sump pumps are running.~~ BMCC-4

Proposed Answer: ~~d) RHR Room ECCS sump pumps are running.~~

Explanation (Optional): ~~**Make new tie and evaluate distracters question was tied to 295036.EA2.02**~~

a) Core Spray HDD Pumps

Technical Reference(s): EOP-5 OP-14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MIT-301.11 1.07 (As available)

Question Source: Bank # Monticello 1 INPO # 15350

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/23/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
HPCI	Group #	1	1
Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) D.C. power: BWR-2,3,4	K/A # 206000	K1.07	K1.07
	Importance Rating	3.7	3.8

Proposed Question: Which of the following would render HPCI incapable of accomplishing it's design purpose?

RO/SRO 32/44

a) Loss of the 10600 Bus

b) Loss of 125 VDC Bus 'B'

c) Loss of Condensate Storage Tank level

d) Loss of Secondary Containment integrity

Proposed Answer: b) Loss of 125 VDC Bus 'B'

Explanation (Optional):

Technical Reference(s): OP-15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-23, EO-1.10.E (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7,8

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
LPCS	Group #	1	1
Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)	K/A # 209001	2.2.13	2.2.13
	Importance Rating	3.6	3.8

Proposed Question: Which one (1) of the following requirements must be met in order to permit operation of Core Spray Hold Pump 'A' under a Striped Tag?

RO/SRO

33/45

a) Tag Holder for the CS Hold Pump must be designated by position such as "Electrical Supervisor".

b) A procedure or Work Request with Step Text must exist to provide CS Hold Pump operation guidance.

c) Tag Holder for the CS Hold Pump with concurrence from the Field Support Supervisor directs CS Hold Pump operation.

d) If the CS Hold Pump is out of it's protected position for > one (1) shift, Tagout control must shift to the Work Week Manager.

Proposed Answer: b) A procedure or Work Request with Step Text must exist to provide CS Hold Pump operation guidance.

Explanation (Optional):

Technical Reference(s): AP-12.01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LP-AP-44.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
LPCS continued	Group #	1	1
Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following: (CFR: 41.7 / 45.4) ADS logic	K/A # 209001	K3.02	K3.02
	Importance Rating	3.8	3.9

Proposed Question: While at full power, a small break LOCA with HPCI inoperable has occurred.
ADS has initiated.
The only Low Pressure ECCS in service is Core Spray 'B' which subsequently trips.
ADS valves will ______________?

RO/SRO
34/46

a) Remain open.
b) Close immediately.
c) Close after a two (2) minute delay.
d) Remain open until RPV level reaches ≥ 59.5".

Proposed Answer: b) Close immediately.

Explanation (Optional):

Technical Reference(s): OP-68, OP-14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14, EO-1.09.B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
SLC	Group #	1	1
Ability to direct personnel activities inside the control room.(CFR: 45.5 / 45.12 / 45.13)	K/A # 211000	2.1.9	2.1.9
	Importance Rating	2.5	4.0

Proposed Question: A failure to SCRAM occurs from power operation:

- Reactor power is 50%
- Main Turbine/ Generator is on line
- Torus temperature is 80°F and steady
- Feedwater/Condensate System is maintaining RPV level
- The SNO-1 reports that APRM power is oscillating 30%.

Which action should be directed?

RO/SRO 35/47

a) Emergency RPV Depressurization
b) SLC System initiation
c) Main Turbine trip
d) MSIV Closure

Proposed Answer: b) SLC System initiation

Explanation (Optional):

Technical Reference(s): EOP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP3LP, EO-1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RPS	Group #	1	1
Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : (CFR: 41.5 / 45.3) Specific logic arrangements	K/A # 212000	K5.02	K5.02
	Importance Rating	3.3	3.4

Proposed Question: While at 20% power, what possible Reactor Protection System (RPS) response(s) can occur if the Inboard and Outboard MSIV's on any two (2) Main Steam Lines are closed?

RO/SRO

36/48

a) No response **OR** full SCRAM

b) No response **OR** half SCRAM

c) Half SCRAM always

d) Full SCRAM always

Proposed Answer: b) No response **OR** half SCRAM

Explanation (Optional):

Technical Reference(s): ST-1I, OP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29, EO 1.09.f, 1.13.C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
IRM	Group #	1	1
Ability to monitor automatic operations of the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status	K/A # 215003	A3.04	A3.04
	Importance Rating	3.5	3.5

Proposed Question: An IRM HI Flux Control Rod Block is automatically bypassed when__________?

RO/SRO

37/49

a) The Reactor Mode Switch is placed in RUN.
b) The IRM is on Range 1.
c) The IRM's companion APRM is downscale.
d) The SRM's are fully inserted.

Proposed Answer: a) The Reactor Mode Switch is placed in RUN.

Explanation (Optional):

Technical Reference(s): OP-16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO- 1.05.C.2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Source Range Monitor	Group #	1	1
Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status	K/A # 215004	A3.04	A3.04
	Importance Rating	3.6	3.6

Proposed Question: The following plant conditions exist:

Reactor Mode Switch is in STARTUP/HOT STBY.

Intermediate Range Monitors (IRM's) all on Range 3.

Source Range Monitor (SRM) A is reading 0.5 cps

SRM's B and C are reading 8.3×10^4

SRM D mode switch is in STANDBY

A rod block signal has been generated.

Which one of the following has caused the rod block?

RO/SRO 38/50

a) SRM Inoperable

b) SRM Count Circuit

c) SRM Downscale

d) SRM Upscale

Proposed Answer: a) SRM Inoperable

Explanation (Optional):

Technical Reference(s): OP-16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.05.b.1, EO 1.05.c, EO 1.14.c (As available)

Question Source: Bank # Perry 1 INPO# 21837 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/1/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Source Range Monitor	Group #	1	1
Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: (CFR: 41.7 / 45.7) Control rod block status	K/A # 215004	A3.04	A3.04
	Importance Rating	3.6	3.6

Proposed Question: The following plant conditions exist: HOT STBY.

Reactor Mode Switch is in STARTUP/~~STANDBY~~

Intermediate Range Monitors (IRM) ~~A, C, D, E, and G~~ all are on Range 3. ~~all other IRMs are on Range 2.~~

Source Range Monitor (SRM) A is reading 0.5 cps. SRMs B and C are reading 8.3 x 10E4. SRM D mode switch is in STANDBY.

A rod block signal has been generated.

Which one of the following has caused the rod block?

RO/SRO

38/50

a) SRM Inoperable

b) SRM Count circuit ~~Detector Wrong Position~~

c) SRM Downscale

a) SRM Upscale

Proposed Answer: a) SRM Inoperable

Explanation (Optional):

Technical Reference(s): OP-16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.05.a.6.b EO 1.05.a.3.i, EO 1.14.c (As available)

Question Source: Bank # Perry 1 INPO# 21837 (Mod)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/1/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41)

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
APRM / LPRM	Group #	1	1
Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: (CFR: 41.7 / 45.7) Full core display	K/A # 215005	A3.02	A3.02
	Importance Rating	3.5	3.5

Proposed Question:

A reactor startup is being performed following a planned outage. Annunciator, 09-5-2-33, LPRM Downscale, clears.

The SNO can confirm that this is expected and correct by verifying?

RO/SRO

39/51

a) All APRM Downscale alarms are clear.

b) All Full Core Display LPRM downscale lights are out.

c) All IRM Range Switches are above Range 1.

d) Reactor Mode Switch is in RUN.

Proposed Answer: b) All Full Core Display LPRM downscale lights are out.

Explanation (Optional):

Technical Reference(s): OP-16, ARP- 09-5-2-33 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07C, EO-1.12.D,1.05.C.1.B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
SLC	Group #		1
Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4) Core spray line break detection system: Plant-Specific Link to 10CFR-55.43(b)(2)	K/A # 211000		K3.02
	Importance Rating		3.2

Proposed Question: Engineering has just informed the Shift Manager of an industry event where flow induced vibration has breached the integrity of the in-vessel section of the Standby Liquid Control System piping.

In addition to SLC, which one (1) of the following system Technical Specification operabilities could be threatened by such an occurrence?

RO/SRO

S52

a) CRD System

b) LPCI System

c) Core Spray System

d) ADS System

Proposed Answer: c) Core Spray System

Explanation (Optional):

Technical Reference(s): OP-14, OP-11, ARP-09-03-1-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-14, EO-1.07.C, 1.16, 1.05.A.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RCIC	Group #	1	1
Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following: (CFR: 41.7 / 45.4) K3.01 Reactor water level	K/A # 217000	K3.01	K3.01
	Importance Rating	3.7	3.7

Proposed Question: The Station has lost all AC electrical power. RPV level is being controlled using RCIC system operation.

Which statement below describes the effect that the loss of 'A' Station Battery will have on level control?

RO/SRO

40/53

a) HPCI system will have to be used to control RPV level.

b) An Emergency Depressurization will be required to enable Low Pressure Injection.

c) All injection sources will be lost. Emergency Depressurize when RPV level drops to TAF.

d) RCIC will continue to operate but must be controlled locally.

Proposed Answer: a) HPCI system will have to be used to control RPV level.

Explanation (Optional):

Technical Reference(s): AOP-45, AOP-49 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP 13, EO 1.09.A, 1.10.B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RCIC	Group #	1	1
Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: (CFR: 41.7 / 45.7) Lights and alarms	K/A # 217000	A3.06	A3.06
	Importance Rating	3.5	3.4

Proposed Question:

During reactor power operation with RCIC in the Standby Mode a RCIC PMP SUCT PRESS HI alarm (09-4-1-33) is received. While investigating at Panel 09-4 the operator observes the following:

- RCIC Pump Minimum Flow Valve, 13MOV-27, opens.
- The RCIC PMP SUCT PRESS HI alarm clears.
- RCIC Pump Minimum Flow Valve, 13MOV-27, closes.

Based on these indications which one of the following actions is warranted:

RO/SRO

41/54

a) Investigate for abnormally high RCIC pump casing and discharge piping temperatures.

b) Investigate for an erroneous RCIC automatic initiation signal.

c) Investigate the malfunction in valve 13MOV-27 opening logic.

d) Investigate the malfunction in valve 13MOV-27 closing logic.

Proposed Answer: a) Investigate for abnormally high RCIC pump casing and discharge piping temperatures.

Explanation (Optional):

Technical Reference(s): OP-19, ARP 09-4-1-33 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO 1.05.a.10 (As available)

Question Source: Bank # JAF LOR # 21701005B01C

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
APRM / LPRM	Group #		1
~~Ability to maintain primary and secondary plant chemistry within allowable limits.~~	K/A # 215005		~~2.1.34~~ 2.1.17
~~(CFR: 41.10 / 43.5 / 45.12)~~			

RANDOMLY RE-SELECTED

Ability to make accurate / clear and concise verbal reports.

(CFR: 45.12 / 45.13)

Importance Rating: ~~2.9~~ 3.6

Proposed Question: A short time after a recirculation pump trip, the SNO-1 reports that the APRM upscale alarm and rod block are in solid and that the LPRM upscale alarm is coming in and quickly clearing every few seconds. He is unable to identify the problem LPRM.

Your response to him is:

RO/SRO
S55

a) Manually scram the reactor, perform immediate actions of AOP-1, REACTOR SCRAM.
b) When identified, bypass the LPRM per OP-16, NEUTRON MONITORING.
c) Identify and bypass the affected APRM per OP-16, NEUTRON MONITORING.
d) Determine if LPRM upscale alarm is valid by observing 09-14 LPRM upscale lamps.

Proposed Answer: a) Manually scram the reactor, perform AOP-1 Immediate Actions.

Explanation (Optional):

Technical Reference(s): ____________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # ____________

Modified Bank # ____________ (Note changes or attach parent)

New NEW

Question History: Last NRC Exam ____________

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ____________

10 CFR Part 55 Content: 55.41 5, 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
RCIC	Group #		1
Knowledge of electrical power supplies to the following: (CFR: 41.7) Gland seal compressor (vacuum pump) Link to 10CFR-55.43(b)(2)	K/A # 217000		K2.04
	Importance Rating		2.6

Proposed Question: The Plant is operating at 100% power steady state with HPCI tagged out for maintenance, day three (3) of the LCO. At 02:00 am this morning, the feeder breaker to BMCC-1 opened on over-current and will not reset.

Which of the following is appropriate for this situation?

RO/SRO

S56

a) HPCI is the only system affected, original LCO applies.

b) A more restrictive LCO applies as both HPCI and ADS are affected.

c) A more restrictive LCO applies as both HPCI and RCIC are affected.

d) A more restrictive LCO applies due to loss of Primary Containment Isolation.

Proposed Answer: c) A more restrictive LCO applies as both HPCI and RCIC are affected.

Explanation (Optional):

Technical Reference(s): OP-43, OP-19 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: FE-1AH

Learning Objective: SDLP-13, EO-1.04.A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

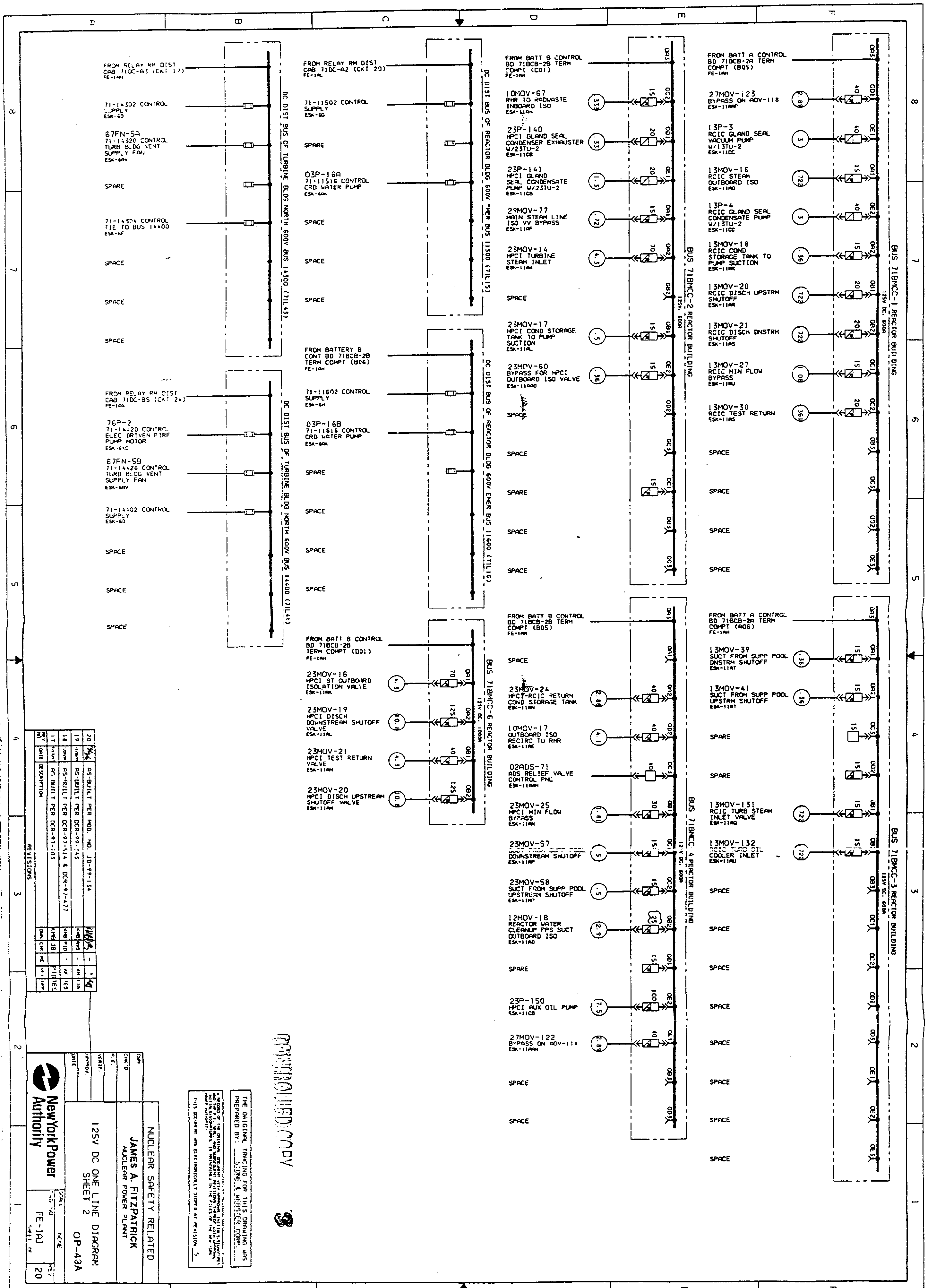
Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43 2

Comments:



FROM RELAY RM DIST CAB 71DC-A5 (CKT 17) FE-1AH
71-14502 CONTROL SUPPLY ESK-6D
67FN-5A 71-14520 CONTROL TURB BLDG VENT SUPPLY FAN ESK-6AV
SPARE
71-14504 CONTROL TIE TO BUS 14400 ESK-6F
SPACE
SPACE
SPACE
DC DIST BUS OF TURBINE BLDG NORTH 600V BUS 14300 (71L43)
FROM RELAY RM DIST CAB 71DC-B5 (CKT 24) FE-1AK
76P-2 71-14420 CONTROL ELEC DRIVEN FIRE PUMP MOTOR ESK-6AC
67FN-5B 71-14426 CONTROL TURB BLDG VENT SUPPLY FAN ESK-6AV
71-14502 CONTROL SUPPLY ESK-6D
SPACE
SPACE
SPACE
DC DIST BUS OF TURBINE BLDG NORTH 600V BUS 14400 (71L44)
FROM RELAY RM DIST CAB 71DC-A2 (CKT 20) FE-1AL
71-11502 CONTROL SUPPLY ESK-6G
SPARE
03P-16A 71-11516 CONTROL CRD WATER PUMP ESK-6AK
SPACE
SPACE
SPACE
DC DIST BUS OF REACTOR BLDG 600V EMER BUS 11500 (71L15)
FROM BATTERY B CONT BD 71BCB-2B TERM COMPT (B06) FE-1AH
71-11602 CONTROL SUPPLY ESK-6H
03P-16B 71-11616 CONTROL CRD WATER PUMP ESK-6AK
SPARE
SPACE
SPACE
SPACE
DC DIST BUS OF REACTOR BLDG 600V EMER BUS 11600 (71L16)
FROM BATT B CONTROL BD 71BCB-2B TERM COMPT (D01) FE-1AH
23MOV-16 HPCI ST OUTBOARD ISOLATION VALVE ESK-11AK
23MOV-19 HPCI DISCH DOWNSTREAM SHUTOFF VALVE ESK-11AL
23MOV-21 HPCI TEST RETURN VALVE ESK-11AM
23MOV-20 HPCI DISCH UPSTREAM SHUTOFF VALVE ESK-11AM
BUS 71BMCC-6 REACTOR BUILDING 125V DC 100A
FROM BATT B CONTROL BD 71BCB-2B TERM COMPT (C01) FE-1AH
10MOV-67 RHR TO RADWASTE INBOARD ISO ESK-11AH
23P-140 HPCI GLAND SEAL CONDENSER EXHAUSTER W/23TU-2 ESK-11CB
23P-141 HPCI GLAND SEAL CONDENSATE PUMP W/23TU-2 ESK-11CB
29MOV-77 MAIN STEAM LINE ISO VV BYPASS ESK-11AF
23MOV-14 HPCI TURBINE STEAM INLET ESK-11AK
SPACE
23MOV-17 HPCI COND STORAGE TANK TO PUMP SUCTION ESK-11AL
23MOV-60 BYPASS FOR HPCI OUTBOARD ISO VALVE ESK-11AAD
SPACE
SPACE
SPARE
SPACE
SPACE
BUS 71BMCC-2 REACTOR BUILDING 125V DC 600A
FROM BATT B CONTROL BD 71BCB-2B TERM COMPT (B05) FE-1AH
SPACE
23MOV-24 HPCI-RCIC RETURN COND STORAGE TANK ESK-11AM
10MOV-17 OUTBOARD ISO RECIRC TO RHR ESK-11AE
02ADS-71 ADS RELIEF VALVE CONTROL PNL ESK-11AAM
23MOV-25 HPCI MIN FLOW BYPASS ESK-11AN
23MOV-57 DOWNSTREAM SHUTOFF ESK-11AP
23MOV-58 SUCT FROM SUPP POOL UPSTREAM SHUTOFF ESK-11AP
12MOV-18 REACTOR WATER CLEANUP PPS SUCT OUTBOARD ISO ESK-11AD
SPARE
23P-150 HPCI AUX OIL PUMP ESK-11CB
27MOV-122 BYPASS ON AOV-114 ESK-11AAN
SPACE
SPACE
BUS 71BMCC-4 REACTOR BUILDING 125V DC 600A
FROM BATT A CONTROL BD 71BCB-2A TERM COMPT (B05) FE-1AH
27MOV-123 BYPASS ON AOV-118 ESK-11AAP
13P-3 RCIC GLAND SEAL VACUUM PUMP W/13TU-2 ESK-11CC
13MOV-16 RCIC STEAM OUTBOARD ISO ESK-11AQ
13P-4 RCIC GLAND SEAL CONDENSATE PUMP W/13TU-2 ESK-11CC
13MOV-18 RCIC COND STORAGE TANK TO PUMP SUCTION ESK-11AR
13MOV-20 RCIC DISCH UPSTRM SHUTOFF ESK-11AR
13MOV-21 RCIC DISCH DNSTRM SHUTOFF ESK-11AS
13MOV-27 RCIC MIN FLOW BYPASS ESK-11AU
13MOV-30 RCIC TEST RETURN ESK-11AS
SPACE
SPACE
SPACE
SPACE
BUS 71BMCC-1 REACTOR BUILDING 125V DC 600A
FROM BATT A CONTROL BD 71BCB-2A TERM COMPT (A06) FE-1AH
13MOV-39 SUCT FROM SUPP POOL DNSTRM SHUTOFF ESK-11AT
13MOV-41 SUCT FROM SUPP POOL UPSTRM SHUTOFF ESK-11AT
SPARE
SPARE
13MOV-131 RCIC TURB STEAM INLET VALVE ESK-11AQ
13MOV-132 COOLER INLET ESK-11AU
SPACE
SPACE
SPACE
SPACE
SPACE
SPACE
SPACE
SPACE
BUS 71BMCC-3 REACTOR BUILDING 125V DC 600A
REVISIONS
20 AS-BUILT PER MOD. NO. JD-97-134
19 AS-BUILT PER DCR-99-145
18 AS-BUILT PER DCR-97-514 & DCR-97-477
17 AS-BUILT PER DCR-97-205
NO. DATE DESCRIPTION
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NUCLEAR SAFETY RELATED
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
125V DC ONE LINE DIAGRAM SHEET 2
OP-43A
New York Power Authority
FE-1AJ
REV 20

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
ADS	Group #	1	1
Knowledge of the physical connections and/or cause effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Safety/relief valves	K/A # 218000	K1.06	K1.06
	Importance Rating	3.9	3.9

Proposed Question: The Plant is at 70% power; the Control Room receives annunciator, 09-4-2-37, "SRV Electric Lift Initiated or Bypassed".

All SRV green lights are on.

Which of the following describes how this impacts the operation of the ADS Valve(s)?

RO/SRO 42/57

a) Will operate normally on hydraulic overpressure.

b) Only operate on High RPV pressure setpoint.

c) Only operate manually from Panel 02ADS-071.

d) Only operate manually from 09-4 Panel.

Proposed Answer: a) Will operate normally on hydraulic overpressure.

Explanation (Optional):

Technical Reference(s): ARP-09-4-2-37, OP-68; GE DWG 791E453 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29, EO-1.05.A.4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New: NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge: X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41: 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
PCIS/Nuclear Steam Supply Shutoff	Group #	1	1
Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Process radiation monitoring system failures	K/A # 223002	A2.04	A2.04
	Importance Rating	2.9	3.2

Proposed Question: While operating at normal full power with the 'A' Standby Gas Treatment System out of service, the 'B' Reactor Building Ventilation Exhaust Radiation Monitor fails downscale. Which of the following describes expected Operator action(s)?

RO/SRO 43/58

a) Using OP-51A, verify Reactor Building Ventilation isolation.
b) Using OP-20, verify automatic start of the 'B' Standby Gas Treatment System.
c) Using AOP-15, reset the isolation and restart Reactor Building Ventilation.
d) Using OP-20, manually start up the 'B' Standby Gas Treatment System.

Proposed Answer: d) Using OP-20, manually start up the 'B' Standby Gas Treatment System.

Explanation (Optional):

Technical Reference(s): JAF LER-98-001, OP-20, ITS-3.3.6.2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B, EO-1.05.c, 1.10.a,1.14.b (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
PCIS/Nuclear Steam Supply Shutoff	Group #		1
Knowledge of electrical power supplies to the following: (CFR: 41.7) Logic power supplies Link to 10CFR-55.43(b)(5)	K/A # 223002		K2.01
	Importance Rating		2.7

Proposed Question: In which of the following complete system loss events, would you expect to find at least one MSIV in each Main Steam Line closed?

a) AOP-59, Loss of RPS Bus A Power
OR
AOP-45, Loss of DC Power System A

b) AOP-18, Loss of 10500 Bus
AND
AOP-45, Loss of DC Power System A

RO/SRO

c) AOP-21, Loss of UPS
AND
AOP-46, Loss of DC Power System B

S59

d) AOP-19, Loss of 10600 Bus
OR
AOP-46, Loss of DC Power System B

Proposed Answer: b) AOP-18, Loss of 10500 Bus
AND
AOP-45, Loss of DC Power System A

Explanation (Optional):

Technical Reference(s): AOP-18, AOP-19, AOP-21
AOP-45, AOP-46 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29, EO 1.04.a, EO 1.05.a.1.c (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
PCIS/Nuclear Steam Supply Shutoff	Group #	1	1
Knowledge of conditions and limitations in the facility license.(CFR: 43.1 / 45.13)	K/A # 223002	2.1.10	2.1.10
	Importance Rating	2.7	3.9

Proposed Question:

During routine full power operation, the SNO identifies that the breaker for 20MOV-94, Drywell Equip. Sump Isolation Valve, has opened and the valve position is open.

Which statement below describes the required plant actions?

RO/SRO

44/60

a) Valve may remain open as long as the system is not operating.

b) An NPO must be dispatched immediately to manually close the valve.

c) A valve in that line must be de-activated closed within four (4) hours.

d) Valve may remain open under administrative control indefinitely.

Proposed Answer: c) A valve in that line must be de-activated closed within four (4) hours.

Explanation (Optional):

Technical Reference(s): ST-1C, ITS- 3.6.1.3 (Attach if not previously provided)

TRM Appendix A

Proposed references to be provided to applicants during examination:

Learning Objective: SDLP-16C, EO 1.13.a (As available)

Question Source: Bank # JAF LOR # 22302001B01S

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 9

55.43 1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Main and Reheat Steam	Group #		2
Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Malfunction of reactor turbine pressure regulating system Link to 10CFR-55.43(b)(5)	K/A # 239001		A2.01
	Importance Rating		3.9

Proposed Question: The reactor is operating at 25% power with recirculation flow at minimum.

If a turbine trip occurs and the bypass valves fail to open, which of the following would be the appropriate procedure(s) to respond to the event?

a) AOP-1, Reactor Scram, AND AOP-6, Malfunction of EHC Pressure Regulator.

RO/SRO

b) AOP-2, Main Turbine Trip Without Scram, AND AOP-6, Malfunction of EHC Pressure Regulator.

S61

c) EOP-2, RPV Control, AND AOP-2, Main Turbine Trip Without Scram.

d) AOP-1, Reactor Scram, AND EOP-2, RPV Control.

Proposed Answer: d) AOP-1, Reactor Scram, AND EOP-2, RPV Control.

Explanation (Optional): High pressure entry into EOP-2 and AOP-1

Technical Reference(s): AOP-1, EOP-2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05, EO 1.07.a.6, 7, & 10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5, 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
SRVs	Group #	1	1
Knowledge of the physical connections and/or cause effect relationships between RELIEF/SAFETY VALVES and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Nuclear boiler	K/A # 239002	K1.01	K1.01
	Importance Rating	3.8	3.9

Proposed Question: After steady state conditions are achieved, which of the below is confirmation of an inadvertent SRV full opening while in normal full power operation?

RO/SRO 45/62

a) Reactor Power at 100% and Main Generator Output at 875 MWe.

b) RPV Water level at 207 inches and Level Set at 202 inches.

c) Feed flow at 11 x 10^6 lbm/hr and Steam flow at 10 x 10^6 lbm/hr.

d) Torus water temperature trending down slowly with Torus cooling in service.

Proposed Answer: c) Feed flow at 11 x 10^6 lbm/hr and Steam flow at 10 x 10^6 lbm/hr.

Explanation (Optional):

Technical Reference(s): AOP-36 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO-1.02,2.27 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Reactor Water Level Control	Group #	1	1
Knowledge of the physical connections and/or cause effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Condensate system	K/A # 259002	K1.13	K1.13
	Importance Rating	3.2	3.2

Proposed Question:

The plant is operating at 100% power with a normal Feed and Condensate alignment. There are no systems or components inoperable. The A Condensate Pump trips due to an electrical fault.

Which one of the following is the expected result of this trip?

a) The operating pumps assume the additional load and the RFPs are not affected. A normal power reduction is required.

b) The A Condensate Booster Pump trips on interlock, but the RFPs are not affected. A normal power reduction is required.

RO/SRO

46/63

c) The A Condensate Booster Pump trips on interlock causing RFPs to trip on low suction pressure. A manual SCRAM is required.

d) Condensate Booster Pump suction pressure decreases causing RFPs to trip on low suction pressure. A manual SCRAM is required.

Proposed Answer: a) The operating pumps assume the additional load and the RFPs are not affected. A normal power reduction is required.

Explanation (Optional):

Technical Reference(s): AOP-42, OP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-33, EO 1.05.b.2 & 1.14.c (As available)

Question Source: Bank # JAF LOR# 25601012B02C

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
SGTS	Group #	1	1
Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: (CFR: 41.7 /45.6) Secondary containment radiation/ contamination levels	K/A # 261000	K3.05	K3.05
	Importance Rating	3.2	3.5

Proposed Question: A Station Blackout has occurred resulting in a full Reactor SCRAM with all rods in. HPCI is operating to maintain RPV water level and pressure control.

As a result of HPCI operation:

RO/SRO

47/64

a) The 'A' Station Battery is expected to rapidly deplete.

b) The Crescent Area contamination levels are expected to rise.

c) The HPCI Turbine MUST be manually tripped on RPV high water level.

d) RPV water level is expected to slowly drop until injection overcomes decay heat losses.

Proposed Answer: b) The Crescent Area contamination levels are expected to rise.

Explanation (Optional):

Technical Reference(s): AOP-49, AOP-45, AOP-46 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-01B, EO-1.09.A, F, LPAOP-49, EO-1.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
AC Electrical Distribution	Group #	1	1
Knowledge of electrical power supplies to the following: (CFR: 41.7) Off-site sources of power	K/A # 262001	K2.01	K2.01
	Importance Rating	3.3	3.6

Proposed Question: The Plant is in day 8 of a refuel outage. A full core off load has just been completed. Niagara Mohawk called to report that the Lake Road 13.2 KV line is being taken out of service immediately.

Which of the below is a priority Control Room action?

RO/SRO

48/65

a) Dispatch an NPO to transfer DHR power to the Diesel Generator.

b) Transfer in house electrical distribution from Normal to Reserve Station Service.

c) Dispatch an NPO to align 115 KV control power to the alternate source.

d) Implement alternate temperature monitoring of Interim Spent Fuel Storage.

Proposed Answer: a) Dispatch an NPO to transfer DHR power to the Diesel Generator.

Explanation (Optional):

Technical Reference(s): AOP-71 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP, EO-1.03, SDLP-71S, EO-1.09, SDLP-32, EO-1.04,1.10.A (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
UPS (AC/DC)	Group #	1	1
Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)	K/A # 262002	2.2.13	2.2.13
	Importance Rating	3.6	3.8

Proposed Question: Operators are tagging out the UPS M/G set for bearing replacement. Worker protection is assured by:

RO/SRO 49/66

a) A Maintenance PTR.
b) A Special Condition PTR.
c) A Striped PTR.
d) A Hold PTR.

Proposed Answer: b) Independent Verification.

Explanation (Optional):

Technical Reference(s): OP-46B, AP-12.01 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAP-EO 44.03 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
Offgas	Group #		2
Ability to predict and/or monitor changes in parameters associated with operating the OFFGAS SYSTEM controls including: (CFR: 41.5 / 45.5) Filter differential pressure Link to 10CFR-55.43	K/A # 271000		A1.06
	Importance Rating		2.5

Proposed Question:

The plant is operating at rated power when the following indications simultaneously occur:

- 09-6-1-23, OFF GAS LINE FILTER DIFF PRESS HI in and clear.
- 09-6 Off Gas Flow Recorder (38FR-101) drops from 120 to 70 SCFM.
- EPIC OFFGASRAD alarm.
- 09-10 Off Gas Radiation Monitors both reading 150-175 mr/hr.

Which of the following describes the plant event and the appropriate mitigating procedure?

RO/SRO

S67

a) Off Gas line blockage will cause a loss of condenser vacuum. AOP-31, LOSS OF CONDENSER VACUUM.

b) A hydrogen fire has ignited in piping downstream of the SJAE's. AOP-5, COMBUSTION IN SJAE AFTERCONDENSER.

c) Fuel failure has resulted in a large radioactive gas release from the RPV. AOP-3, HIGH ACTIVITY IN REACTOR COOLANT OR OFFGAS

d) An explosion has breached the SJAE discharge piping. AOP-4, EXPLOSION IN AIR EJECTOR DISCHARGE PIPING

Proposed Answer: b) A hydrogen fire has ignited in piping downstream of the SJAE's. AOP-5, COMBUSTION IN SJAE AFTERCONDENSER.

Explanation (Optional):

Technical Reference(s): ARP-09-6-1-23/AOP-4 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAOP-EO-1.01 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5,10

55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
DC Electrical Distribution	Group #	1	1
Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Grounds	K/A # 263000	A2.01	A2.01
	Importance Rating	2.8	3.2

Proposed Question: The plant is operating at 100% with operators attempting to locate a ground on the "A" station battery. The next breaker to be opened is the supply for 10700 BKR Control Power (71DCA3 Crkt 24).

How will the opening of this circuit affect the 4KV breakers on the 10700 bus **AND** which procedures will provide guidance?

RO/SRO

50/68

a) The breakers will immediately trip, AOP-20, Loss of 10700 Bus, and AOP-22, DC Power System 'A' Ground Isolation.

b) The breakers can be tripped mechanically only, OP-46A, 4160 V & 600 V Normal AC Power Distribution, and AOP-22, DC Power System 'A' Ground Isolation.

c) Electrical protective trips will operate normally, OP-43A, 125 VDC Power System, and OP-46A, 4160 V & 600 V Normal AC Power Distribution.

d) Breaker position indication lights (red and green) will continue to indicate breaker positions, OP-43A, 125 VDC Power System, and AOP-23, DC Power System 'B' Ground Isolation.

Proposed Answer: b) The breakers can be tripped mechanically only, OP-46A, 4160 V & 600 V Normal AC Power Distribution, and AOP-22, DC Power System 'A' Ground Isolation.

Explanation (Optional):

Technical Reference(s): AOP-22 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-71B, EO-1.09.C.18 (As available)

Question Source: Bank # JAF LOR # 20004211B01C

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
EDGs	Group #	1	1
Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) : (CFR: 41.7 / 45.7) Fuel oil pumps	K/A # 264000	K6.02	K6.02
	Importance Rating	3.6	3.6

Proposed Question: A LOCA and LOOP exists. Off-site power is not expected to be returned to service for two days. All EDG equipment is operable with the exception that Fuel Oil Transfer Pumps 93P1-A1 & 2 have just tripped and cannot be started.

Based upon these events, select the expected plant response assuming NO Operator action.

RO/SRO

51/69

a) EDG's "A", "B", "C", & "D" will continue to run until off- site power is restored.

b) EDG "A" will trip immediately, EDG's "B","C", & "D" will continue to run until off-site power is restored.

c) EDG "A" will continue to run for up to three (3) hours then trip, EDG's "B","C", & "D" will continue to run until off- site power is restored.

d) EDG "A" will continue to run at reduced capacity until off- site power is restored, EDG's "B","C", & "D" will continue to run until off- site power is restored.

Proposed Answer: c) EDG "A" will continue to run for up to three (3) hours then trip, EDG's "B","C", & "D" will continue to run until off- site power is restored.

Explanation (Optional):

Technical Reference(s): OP-22 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-93, EO-1.05.A.4, 1.10.G (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the process for making changes in procedures as described in the safety analysis report. (CFR: 43.3 / 45.13)	Group #		
	K/A #		2.2.6
	Importance Rating		3.3

Proposed Question: Who must approve a temporary change to a Technical Specification related procedure.

RO/SRO

S70

a) The procedure RPO and an Operations QTR

b) A plant management QTR and a SRO license QTR

c) The General Manager-Plant Operations and a plant management QTR.

d) A plant management QTR and any operations licensed QTR.

Proposed Answer: b) A plant management QTR and a SRO license QTR

Explanation (Optional):

Technical Reference(s): AP-2.04 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAP, EO-4.05 (As available)

Question Source: Bank #

Modified Bank # NEW (Note changes or attach parent)

New

Question History: Last NRC Exam 6/5/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43 3

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Instrument Air	Group #	1	1
Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Main Steam Isolation Valve air	K/A # 300000	K1.05	K1.05
	Importance Rating	3.1	3.2

Proposed Question: The plant is operating at 75% reactor power.

The SNO-1 depresses the TEST pushbutton for 29AOV-86B, 'B' OUTBOARD MSIV.

Which one of the following describes the response of 29AOV-86B?

RO/SRO

52/71

a) Instrument Nitrogen bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

b) Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

c) Instrument Nitrogen bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

d) Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

Proposed Answer: b) Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

Explanation (Optional):

Technical Reference(s): ST-1I, OP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Perry 1 INPO # 21861 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/1/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Instrument Air	Group #	1	1
Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Main Steam Isolation Valve air	K/A # 300000	K1.05	K1.05
	Importance Rating	3.1	3.2

Proposed Question: The plant is operating at 75% reactor power.

29AOV-86B 'B' MSIV

~~MSL B INBD MSIV B21-F022B control switch is in the TEST position.~~ The Control Room Operator depresses the ~~MSL B INBD MSIV~~ TEST pushbutton ~~1B21H-S3B.~~

Which one of the following describes the response of ~~MSL B INBD MSIV B21-F022B?~~ 29AOV-86B

RO/SRO

52/71

a) ~~Safety Related~~ Instrument ~~Air~~ Nitrogen bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

b) Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

c) ~~Safety Related~~ Instrument ~~Air~~ Nitrogen bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

d) Instrument Air bleeds off the bottom portion of the MSIV air cylinder and the top portion of the MSIV air cylinder is pressurized to stroke the MSIV closed in 3-5 seconds.

Proposed Answer: b) Instrument Air bleeds off the bottom portion of the MSIV air cylinder causing the MSIV to slowly close.

Explanation (Optional):

Technical Reference(s): ST-11; OP-1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-29 1.05a.1.a (As available)

Question Source: Bank # Perry 1 INPO # 21861 (Mod)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/1/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 K 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the effects of alterations on core configuration. (CFR: 43.6)	Group #		
	K/A #		2.2.32
	Importance Rating		3.3

Proposed Question: The purpose of core spiral fuel un-loading is which one of the following?

RO/SRO

S72

a) it minimizes the possibility of flow induced vibration of nuclear instrumentation

b) it precludes the formation of moderator filled cavities surrounded on all sides by fuel

c) it prevents SRM count rates from spiking when fuel is being off-loaded

d) it enables the completion of a full core off-load in less time

Proposed Answer: b) it precludes the formation of moderator filled cavities surrounded on all sides by fuel

Explanation (Optional):

Technical Reference(s): ITS - Bases, RAP- 7.1.24, RAP-7.1.04B Section 5.10.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-07B, EO 1.13.E, 1.17.G (As available)

Question Source: Bank # JAF LOR # 1332

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Component Cooling Water	Group #	1	1
Knowledge of electrical power supplies to the following: (CFR: 41.7) CCW pumps	K/A # 400000	K2.01	K2.01
	Importance Rating	2.9	3.0

Proposed Question: An electrical transient has occurred and Switchgear L-14 is de-energized.

Which of the following equipment would be lost due to the degraded electrical source?

RO/SRO

53/73

a) 12P-1A, RWCU Pump "A"

b) 15P-2B, RBCLC Pump "B"

c) 11P-2B, SLC Pump "B"

d) 46P-1B, Service Water Pump "B"

Proposed Answer: b) 15P-2B, RBCLC Pump "B"

Explanation (Optional):

Technical Reference(s): OP-46 (Attach if not previously provided)

OP-40

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-15, EO-1.03B (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)	Group #		
	K/A #		2.3.4
	Importance Rating		3.1

Proposed Question: Authorization to receive radiological exposures in excess of 10CFR20 limits is the responsibility of the ______________ .

RO/SRO

S74

a) Radiation Protection Manager

b) Emergency Director

c) TSC Manager

d) General Manager- Plant Operations

Proposed Answer: b) Emergency Director

Explanation (Optional):

Technical Reference(s): EAP-15, AP-07.05 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.3, EO-1.20.B (As available)

Question Source: Bank # LaSalle 1INPO # 19298 (Modified to JAF)

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ______

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ______

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10)	Group #		
	K/A #		2.3.4
	Importance Rating		3.1

Proposed Question: Authorization to receive radiological exposures in excess of 10CFR20 limits is the responsibility of the ______________.

RO/SRO
S74

a) Recovery Manager
b) ~~Station Director~~ Emergency Director
c) ~~Com Ed Medical Director~~ TSC MANAGER
d) TSC Radiation ~~Protection Director~~ Support ~~GSO~~ Coordinator

Proposed Answer: b) ~~Station Director~~

Explanation (Optional): Emergency Director

Technical Reference(s): EAP - 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: No NC

Learning Objective: CP 12.5.3 1.20.b (As available)

Question Source: Bank # LaSalle 1INPO # 19298 (mod 1. JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
CRD Hydraulic	Group #	2	2
Knowledge of electrical power supplies to the following: (CFR: 41.7) Backup SCRAM valve solenoids	K/A # 201001	K2.03	K2.03
	Importance Rating	3.5	3.6

Proposed Question: WHICH ONE of the following supplies power to the Backup Scram Valves?

RO/SRO

54/75

a) 120 VAC UPS

b) 125 VDC

c) 24 VDC

d) 120 VAC RPS

Proposed Answer: b) 125 VDC

Explanation (Optional):

Technical Reference(s): OP-18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05, EO-1.04.A (As available)

Question Source: Bank # Oyster Creek 1 INPO # 13001 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/29/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
CRD Hydraulic	Group #	2	2
Knowledge of electrical power supplies to the following: (CFR: 41.7) Backup SCRAM valve solenoids	K/A # 201001	K2.03	K2.03
	Importance Rating	3.5	3.6

Proposed Question: WHICH ONE of the following supplies power to the Backup Scram Valves?

RO/SRO

54/75

a) 120 VAC Vital instrument

b) 125 VDC

c) 24 VDC

d) 120 VAC RPS

Proposed Answer: ~~d) 120 VAC R~~PS

Explanation (Optional): b) 125V DC

Technical Reference(s): OP-18 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-05 1.04.a (As available)

Question Source: Bank # Oyster Creek 1 INPO # 13001

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/29/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the requirements for reviewing and approving release permits. (CFR: 43.4 / 45.10)	Group #		
	K/A #		2.3.6
	Importance Rating		3.1

Proposed Question: Who must authorize the Liquid Radwaste Effluent Discharge Permit prior to discharge?

RO/SRO

S76

a) Shift Manager

b) Control Room Supervisor

c) Chemistry Supervisor

d) Radiation Protection Manager

Proposed Answer: a) Shift Manager

Explanation (Optional):

Technical Reference(s): SP-1.05 Attachment # 2 (Attach if not previously provided)

OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-20, EO-1.13.B (As available)

Question Source: Bank # LaSalle 1 INPO # 19297 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the requirements for reviewing and approving release permits. (CFR: 43.4 / 45.10)	Group #		
	K/A #		2.3.6
	Importance Rating		3.1

Proposed Question: What is the HIGHEST level of station management that must review and approve the ODCM prior to purging the containment?

RO/SRO

S 76

a) Health Physics Supervisor
b) Shift Manager
c) Unit Supervisor
d) Health Physics Manager

See attached

Proposed Answer: c) Unit Supervisor

Explanation (Optional):

Technical Reference(s): SF 1.05 att. 2 (Attach if not previously provided)

OP-49

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # LaSalle 1 INPO # 19297 (not to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 X 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Control Rod and Drive Mechanism	Group #	2	2
Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Stuck rod	K/A # 201003	A2.01	A2.01
	Importance Rating	3.4	3.6

Proposed Question:

RO/SRO

55/77

A normal reactor startup was in progress at 7% reactor power. Control Rod 26-35 did not move when given a withdraw signal from it's current notch position 12. Drive water differential pressure has been adjusted to 450 psid. All previous attempts to move this rod have been unsuccessful.

The operator should . . .

a) Individually SCRAM Control Rod 26-35, then disarm it electrically and hydraulically.

b) Attempt to move Control Rod 26-35 by performing "Double Clutching."

c) Declare Control Rod 26-35 INOPERABLE, then disarm it electrically and hydraulically.

d) Raise drive water differential pressure an additional 50 psig and attempt to withdraw Control Rod 26-35.

Proposed Answer: d) Raise drive water differential pressure an additional 50 psig and attempt to withdraw Control Rod 26-35.

Explanation (Optional):

Technical Reference(s): AOP-24 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03C, EO-1.15.C (As available)

Question Source: Bank # Quad Cities 1 INPO # 19545 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Control Rod and Drive Mechanism	Group #	2	2
Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) Stuck rod	K/A # 201003	A2.01	A2.01
	Importance Rating	3.4	3.6

26-35

Proposed Question: A normal reactor startup was in progress at 5% reactor power with normal operating pressure and temperature. Rod H-5 did not move when given a withdraw signal from notch position 12. Drive pressure ~~is 450 psig.~~ has been adjusted to 450 psid The operator attempted a second time to move H-5, but the rod failed to move. The operator ~~should~~

See attached

RO/SRO

55/77

a) scram H-5 individually; disarm H-5 electrically and hydraulically.

b) attempt to move the rod by performing "Double Clutching."

c) declare H-5 INOPERABLE; have H-5 disarmed electrically and hydraulically.

d) increase drive pressure 50 psig and re-attempt to withdraw H-5.

Proposed Answer: d) increase drive pressure 50 psig and re-attempt to withdraw H-5.

Explanation (Optional):

Technical Reference(s): AOP-24 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: AOP-24

Learning Objective: SDLP-03C 1.15.a (As available)

Question Source: Bank # Quad Cities 1 INPO # 19545 (Mod to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 x 5 ? ? ?

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RWM	Group #	2	2
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) *Selected rod position indication:*P-Spec(Not-BWR6)	K/A # 201006	A4.06	A4.06
	Importance Rating	3.2	3.2

Proposed Question: A manual SCRAM was inserted based on lowering RPV water level. The condition has been corrected and RPV level has been returned to the Green Band. During the SCRAM, two (2) control rods failed to fully insert. The SNO has attempted to insert control rods using the CRD System per EP-3, Backup Control Rod Insertion.

Which of the following conditions could prevent manual control rod insertion?

a) SDIV High Level Over-ride Switch in 'Normal'.

RO/SRO
56/78

b) Rod Worth Minimizer Bypass Switch in 'Normal'.

c) Alternate Rod Insertion (ARI) NOT reset.

d) Reactor Protection System SCRAM NOT reset.

Proposed Answer: b) Rod Worth Minimizer Bypass Switch in 'Normal'.

Explanation (Optional):

Technical Reference(s): EP-3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP3LP, EO 1.07 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6, 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RWCU	Group #	2	2
Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) Over temperature protection for system components	K/A # 204000	K4.03	K4.03
	Importance Rating	2.9	2.9

Proposed Question: The plant is performing a reactor startup and heatup, currently at 200 psig.

- Reactor water level control is via Reactor Water Cleanup (RWCU) rejecting to the main condenser hotwell
- Main condenser vacuum has been established with the vacuum pump

The operator is cautioned to carefully monitor system parameters while rejecting.

Which of the following RWCU system trips/isolations provide protection while in this lineup?

a) Cleanup Blowdown Flow Control Valve (12FCV-55) closure on low upstream pressure

RO/SRO

b) RWCU system isolation on non-regenerative heat exchanger high outlet temperature.

57/79

c) Cleanup Blowdown Flow Control Valve (12FCV-55) closure on non-regenerative heat exchanger high outlet temperature.

d) RWCU system Containment Isolation Valve closure on low upstream pressure

Proposed Answer: b) RWCU system isolation on non-regenerative heat exchanger high outlet temperature.

Explanation (Optional):

Technical Reference(s): OP-28, ARP 09-4-2-35 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-12, EO 1.05.c.1 (As available)

Question Source: Bank # Peach Bottom 2 INPO # 18536 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/19/1997

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RWCU	Group #	2	2
Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) Over temperature protection for system components	K/A # 204000	K4.03	K4.03
	Importance Rating	2.9	2.9

Proposed Question: ~~Given the following conditions~~:

The plant ~~--Unit 3~~ is performing a reactor startup and heatup, currently at 200 psig.

--Reactor water level control is via Reactor Water Cleanup (RWCU) rejecting to the main condenser hotwell

-- Main condenser vacuum has been established with the vacuum pump

-- The operator is cautioned to carefully monitor system parameters while rejecting

Which of the following RWCU system trips/isolations ~~will~~ provide protection while in this lineup?

a) Cleanup ~~Drain Header~~ Blowdown Flow Control Valve (~~CV-55~~ 12FCV-55) closure on low upstream pressure

b) RWCU system isolation on non-regenerative heat exchanger high outlet temperature.

RO/SRO

57/79

c) Cleanup ~~Drain Header~~ Blowdown Flow Control Valve (~~CV-55~~ 12FCV-55) closure on non-regenerative heat exchanger high outlet temperature.

d) RWCU system isolation on low upstream pressure

Proposed Answer: b) RWCU system isolation on non-regenerative heat exchanger high outlet temperature.

Explanation (Optional):

Technical Reference(s): OP-28, ARP 09-4-2-35 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-12, EO 1.05.c.1 (As available)

Question Source: Bank # Peach Bottom 2 INPO # 18536 (Mod + 3 pt)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/19/1997

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X 7

55.43

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RPIS	Group #	2	2
Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including: (CFR: 41.7 / 45.7) Alarm and indicating lights	K/A # 214000	A3.02	A3.02
	Importance Rating	3.2	3.1

Proposed Question: During a plant startup, with reactor power at 12%, control rod 18-11 was selected and the following indications occur:

Annunciator 09-5-2-2, ROD WITHDRAWAL BLOCK

Annunciator 09-5-2-1, RWM ROD BLOCK

A loss of ALL rod position indications on the Four Rod Display occurred

A loss of ALL red Full-Out and green Full-In indications of Full Core Display

Which of the following may be the cause for these indications?

RO/SRO

58/80

a) Loss of 120 VAC Panel 71RBACB5
b) Loss of Panel 71AC10
c) Loss of Reactor Protection System (RPS) Distribution Panel A
d) Loss of Uninterruptible Power Supply (UPS)

Proposed Answer: d) Loss of Uninterruptible Power Supply (UPS)

Explanation (Optional):

Technical Reference(s): AOP-21 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-03G, EO-1.04, LPAOP, EO-1.01 (As available)

Question Source: Bank # Fermi 2 2 INPO # 7322 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 12/11/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RPIS	Group #	2	2
Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including: (CFR: 41.7 / 45.7) Alarm and indicating lights	K/A # 214000	A3.02	A3.02
	Importance Rating	3.2	3.1

Proposed Question: During a plant startup, with reactor power at 12%, control rod 18-11 was selected and the following indications occur:

Annun 09-5-2-2 ~~-3D109, CONTROL ROD DRIFT, began alarming~~ Rod Withdrawal Block

Annun. 09-5-2-1 ~~-3D117~~, RODWORTH MINIMIZER BLOCKING began alarming RWM Rod Block

-A loss of ALL rod position indications on the Four Rod Display occurred

-A loss of ALL red Full-Out and green Full-In indications of Full Core Display.

Which of the following may be the cause for these indications?

RO/SRO

58/80

a) Loss of ~~MPU #3~~ 120V AC panel 71RBAC85

b) ~~Total loss of ALL Rod Position Indication~~ Loss of panel 71AC10

c) Loss of Reactor Protection System (RPS) Distribution Panel A

d) Loss of Uninterruptible Power Supply (UPS) ~~Distribution Panel B ONLY~~

Proposed Answer: ~~b) Total loss of ALL Rod Position Indication~~

Explanation (Optional): d. Loss of UPS

Technical Reference(s): ____________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________ (As available)

Question Source: Bank # Fermi 2 2 INPO # 7322 (mod. to SAF)

Modified Bank # ____________ (Note changes or attach parent)

New ____________

Question History: Last NRC Exam 12/11/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ____________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ɑ 7

55.43 ____________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RBM	Group #	2	2
Ability to predict and/or monitor changes in parameters associated with operating the ROD BLOCK MONITOR SYSTEM controls including: (CFR: 41.5 / 45.5) Trip reference: BWR-3,4,5	K/A # 215002	A1.01	A1.01
	Importance Rating	2.7	2.8

Proposed Question: While withdrawing control rod 26-27 at 40% power, which of the below is the probable cause of a withdraw rod block?

RO/SRO

59/81

a) Out of sequence rod recognized by the Rod Worth Minimizer.

b) Rod Block Monitor green Push To Setup lamp is lit.

c) Control Rod 26-27 has withdrawn more than one (1) notch beyond the other rods in that group.

d) All Detector 'A' Bypass lamps are lit on the Four Rod Display.

Proposed Answer: b) Rod Block Monitor green Push To Setup lamp is lit.

Explanation (Optional):

Technical Reference(s): OP-16 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: RAP-7.3.16, Attachment 3

Learning Objective: SDLP-7C, EO-1.05.B.4.F (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

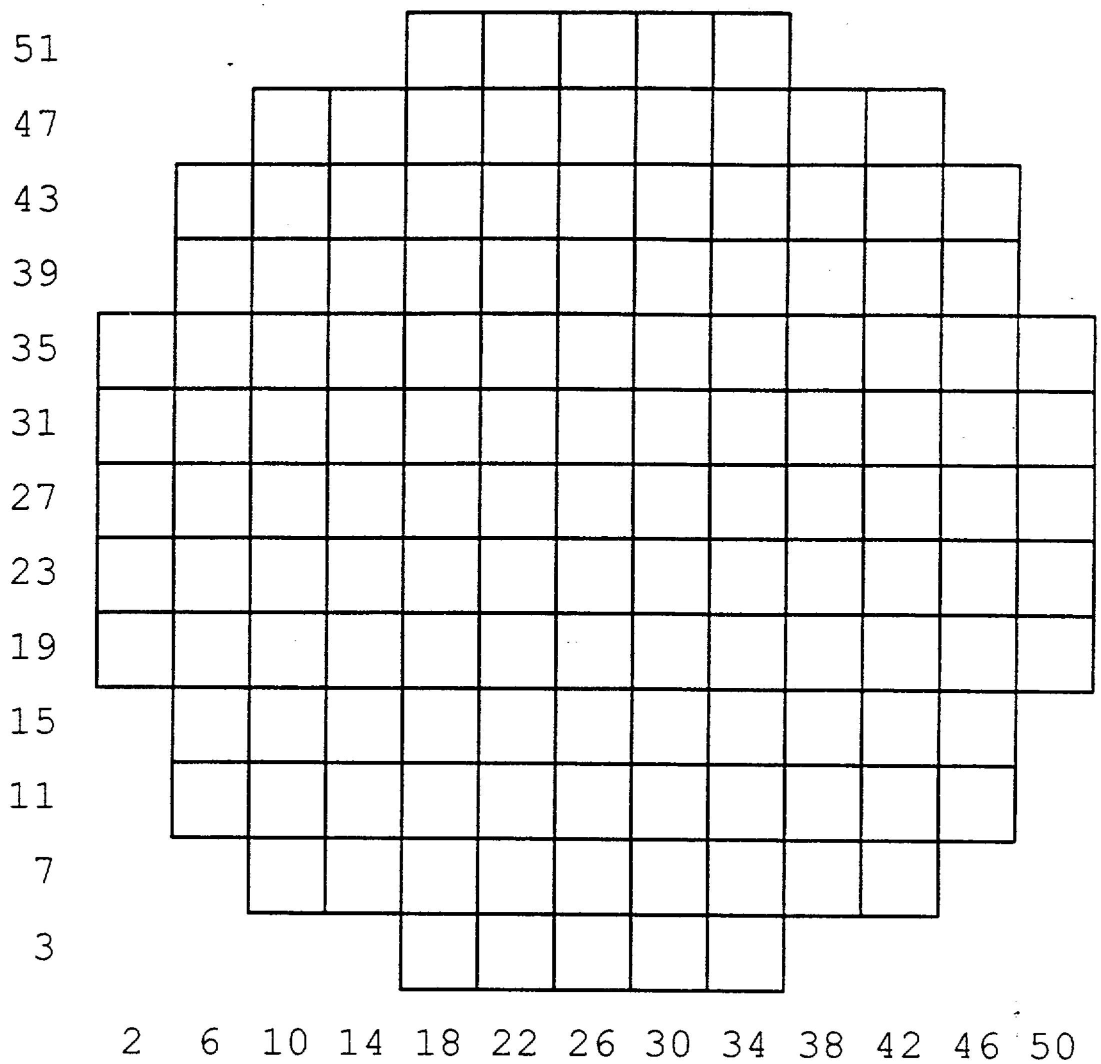
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

APPROVED CONTROL ROD PATTERN



1. Fully withdrawn control rods are indicated by blanks.

2. Immediately notify Reactor Engineer if any control rod is found out of its approved position.

	Date	Date	RE Initial
Applicable From:	________	To: ________	________
Updated From:	________	To: ________	________
Updated From:	________	To: ________	________

- This IS a Quality Record -

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Nuclear Boiler Inst.	Group #	2	2
Knowledge of the effect that a loss or malfunction of the following will have on the *NUCLEAR BOILER INSTRUMENTATION* : (CFR: 41.7 / 45.7) A.C. electrical distribution	K/A # 216000	K6.01	K6.01
	Importance Rating	3.1	3.3

Proposed Question: Given the following plant conditions:

- Drywell temperature - 120F
- Reactor Building temperature - 94F
- Reactor pressure - 880 psig

Immediately following a loss of all AC power, **WHAT** is the MINIMUM reactor water level that can be monitored from the control room?

RO/SRO
60/82

a) +14.5"
b) -150 "
c) –145"
d) +164.5"

Proposed Answer: c) –145"

Explanation (Optional):

Technical Reference(s): EOP- Caution #1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP's

Learning Objective: EOP2LP, EO-1.01 (As available)

Question Source: Bank # LaSalle 1 INPO # 11671 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 10/6/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Nuclear Boiler Inst.	Group #	2	2
Knowledge of the effect that a loss or malfunction of the following will have on the NUCLEAR BOILER INSTRUMENTATION : (CFR: 41.7 / 45.7) A.C. electrical distribution	K/A # 216000	K6.01	K6.01
	Importance Rating	3.1	3.3

Proposed Question: Following a loss of all AC power, what is the MINIMUM reactor water level that can be monitored from the control room?□PLANTCONDITIONS:□Drywell temperature: 120F, Reactor Building temperature: 94F, Reactor pressure: 880 psig

RO/SRO
60/82

a) ~~0~~ inches +145"
b) -150 inches
c) ~~-311~~ inches -145"
d) ~~+25~~ inches 164.5"

Proposed Answer: a) ~~0~~ inches -145"

Explanation (Optional): **QUESTION IS TIED to 216000 K5.01- Make a new Tie**

Technical Reference(s): EOP- caution 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: none

Learning Objective: MIT 301.11B 1.01 (As available)

Question Source: Bank # LaSalle 1 INPO # 11671 (mod + SOJ)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 10/6/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: Torus/Pool Spray Mode	Group #	2	2
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Spray valves	K/A # 230000	A4.02	A4.02
	Importance Rating	3.8	3.6

Proposed Question: During a LOCA, the A and C RHR Pumps are injecting in the LPCI Mode. An operator attempts to place the A loop of RHR in Torus Spray/ Cooling ~~by depressing the open pushbuttons for valves E1150-F028A, Div 1 RHR Torus Iso Vlv; E1150-F027A, Div 1 RHR Torus Spray Iso Vlv; and E11-F024A, Div 1~~ RHR Torus Clg Iso Vlv. as directed by the SOP

Which one of the following will occur?

RO/SRO

62/84

a) The valves may be opened but will immediately close due to a LPCI Initiation ~~interlock~~ signal. present.

b) 10MOV 39A + 34A will open but 10MOV-38A will not open allowing Torus cooling mode of operation ~~E1150-F028A and F027A will open, but E11-F024A will NOT open.~~

c) ~~All three valves can be opened after the LPCI Loop Selection Circuitry has timed out.~~ 10MOV 39A and 38A will open but 10MOV-34A will not open allo Torus spray mode of operation

d) The valves will NOT open due to a LPCI initiation signal ~~still~~ present.

Proposed Answer: d) The valves will NOT open due to a LPCI initiation signal ~~still~~ present.

Explanation (Optional): **NO TIES TO THIS KA Exist- Modify this question to test A.402**

Technical Reference(s): OP-13 attachments (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10 1.05.a.2.b (As available)

Question Source: Bank # Fermi 2 2 INPO # 8890 not to OAF

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/6/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: Torus/Pool Spray Mode	Group #	2	2
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) Spray valves	K/A # 230000	A4.02	A4.02
	Importance Rating	3.8	3.6

Proposed Question: During a LOCA, the A and C RHR Pumps are injecting in the LPCI Mode. An Operator attempts to place the A loop of RHR in Torus Spray as directed by the Control Room Supervisor.

Without further Operator action, design interlocks will result in which of the following when valve operation is initiated?

RO/SRO

62/84

a) The valves may be opened but will immediately close due to a LPCI Initiation signal present.

b) 10MOV-39A and 34A will open, but 10MOV-38A will **NOT** open allowing Torus cooling mode of operation.

c) 10MOV-39A and 38A will open but 10MOV-34A will **NOT** open allowing Torus spray mode of operation.

d) The valves will **NOT** open due to a LPCI initiation signal being present unless the initiation signal is first overridden.

Proposed Answer: d) The valves will **NOT** open due to a LPCI initiation signal being present unless the initiation signal is first overridden.

Explanation (Optional):

Technical Reference(s): OP-13 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-10, EO-1.05.A.2 (As available)

Question Source: Bank # Fermi 2 2 INPO # 8890 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 4/6/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
RHR/LPCI: CTMT Spray Mode	Group #	2	2
Knowledge of the physical connections and/or cause effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) Suppression pool (spray penetration): Plant-Specific	K/A # 226001	K1.12	K1.12
	Importance Rating	3.0	3.0

Proposed Question: Why does the Torus Spray flowpath of EOP-4, Primary Containment Control, prohibit initiation of Torus Spray if Torus Level is > 26 feet?

RO/SRO

61/83

a) Less than 95% of non-condensable gasses exist in the Torus air space.

b) The spray header is covered by Torus water level.

c) The DW to Torus Vent flowpath has been lost.

d) Initiation of Sprays could bring the Torus to sub-atmospheric conditions.

Proposed Answer: b) The spray header is covered by Torus water level.

Explanation (Optional):

Technical Reference(s): BWROG EPG's (Attach if not previously provided)

EOP4LP

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP4LP, EO-1.05 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Fuel Pool Cooling/Cleanup	Group #	2	2
Knowledge of FUEL POOL COOLING AND CLEAN-UP design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) Maintenance of adequate pool temperature	K/A # 233000	K4.03	K4.03
	Importance Rating	2.8	3.1

Proposed Question: A design basis of the Fuel Pool Cooling and Cleanup System is to maintain the Spent Fuel Pool outlet temperature below ___________ for a peak annual refueling heat load of 10 X 10^6 BTU/Hr.

RO/SRO

63/85

a) 155 °F

b) 145 °F

c) 135 °F

d) 125 °F

Proposed Answer: c) 135 °F

Explanation (Optional):

Technical Reference(s): OP-30, AOP-53, FSAR (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-19, EO-1.02 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Reactor/Turbine Pressure Regulator	Group #	2	2
Knowledge of the operational Implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM : (CFR: 41.5 / 45.3) Turbine inlet pressure vs. turbine load	K/A # 241000	K5.05	K5.05
	Importance Rating	2.8	2.9

Proposed Question: Reactor Power is reduced from 100% to 95% by lowering recirculation flow. Turbine Control Valves are repositioned by EHC sensing ________________ as compared to ________________?

RO/SRO

64/86

a) RPV Pressure, Pressure Setpoint.

b) RPV Pressure, Turbine 1st Stage Pressure

c) Turbine Inlet Pressure, Turbine 1st Stage Pressure

d) Turbine Inlet Pressure, Pressure Setpoint.

Proposed Answer: d) Turbine Inlet Pressure, Pressure Setpoint.

Explanation (Optional):

Technical Reference(s): SLP-74C (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-74C, EO-1.05.A.4 (As available)

Question Source: Bank # Dresden 2 INPO # 6524 (Modified to JAF)

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam 3/11/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 5

55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Reactor/Turbine Pressure Regulator	Group #	2	2
Knowledge of the operational implications of the following concepts as they apply to REACTOR/TURBINE PRESSURE REGULATING SYSTEM : (CFR: 41.5 / 45.3) Turbine inlet pressure vs. turbine load	K/A # 241000	K5.05	K5.05
	Importance Rating	2.8	2.9

Proposed Question: You reduce reactor power from 100% to 95% by ~~decreasing~~ lowering recirculation flow. What signals, if any, did the Electro Hydraulic Control use to move the turbine control valve?

RO/SRO

64/86

a) ~~Reactor~~ RPV Pressure, Pressure Setpoint.

b) ~~Reactor~~ RPV Pressure, Max Combined Load Limit.

c) None; the turbine control valve did NOT move.

d) Turbine ~~Throttle~~ INLET Pressure, Pressure Setpoint.

Proposed Answer: d) Turbine ~~Throttle~~ INLET Pressure, Pressure Setpoint.

Explanation (Optional): ~~No KA Ties this question was tied to 241000 A 4.07 – Make the tie to 241000 k5.05~~

Technical Reference(s): ____________________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP 94C 1.05.a.4 (As available)

Question Source: Bank # Dresden 2 INPO # 6524 (mod to OAF)

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam 3/11/1996

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge ________

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X 5

55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
Secondary CTMT	Group #	2	2
Knowledge of the effect that a loss or malfunction of the SECONDARY CONTAINMENT will have on following: (CFR: 41.7 / 45.4) †Off-site radioactive release rates	K/A # 290001	K3.01	K3.01
	Importance Rating	4.0	4.4

Proposed Question: The plant was operating at 100% power when a large steam leak occurred inside the Reactor Building. SGT Train "A" and "B" are operating at rated flows. Secondary Containment pressure is +1.5" WG.

Off-Site radioactivity release rates are expected to be……

RO/SRO

65/87

a) Ground releases via SGT only

b) Ground releases via SGT and Reactor Building Ventilation

c) Elevated releases via SGT and Ground releases via Reactor Building leakage

d) Elevated releases via SGT and Ground releases via Reactor Building Ventilation

Proposed Answer: c) Elevated releases via SGT and Ground releases via Reactor Building leakage

Explanation (Optional):

Technical Reference(s): OP-51A (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-16A EO-1.09b, SDLP-66A, EO-1.05.C (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New **New**

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	Group #		
	K/A #	2.1.32	2.1.32
	Importance Rating	3.4	3.8

Proposed Question: Prior to returning to two loop operation from one loop operation which of the following **LIMITS** must be met and what is the **REASON** for that limit?

RO/SRO

66/88

a) LIMIT - The temperature difference between the bottom head coolant and the recirc loop coolant in the loop to be started is < 145 deg F.

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

b) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

c) LIMIT - The temperature difference between the bottom head coolant and the recirc loop coolant in the loop to be started is < 145 deg F.

REASON - To prevent damage to the fuel cladding that would result from the sudden increase in power due to the injection of cold water.

d) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

REASON - To prevent damage to the fuel cladding that would result from the sudden increase in power due to the injection of cold water.

Proposed Answer: b) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

Explanation (Optional):

Technical Reference(s): ST-26K (Attach if not previously provided)

Proposed references to be provided to applicants during examination: NONE

Learning Objective: SDLP-2H, EO- 1.13g (As available)

Question Source: Bank # Dresden 2 INPO # 21373

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 6/14/2002

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	Group #		
	K/A #	2.1.32	2.1.32
	Importance Rating	3.4	3.8

Proposed Question: Prior to returning to two loop operation from one loop operation which of the following **LIMITS** must be met and what is the **REASON** for that limit?

a) LIMIT - The temperature difference between the bottom head coolant and the recirc loop coolant in the loop to be started is < 145 deg F.

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

b) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

RO/SRO

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

c) LIMIT - The temperature difference between the bottom head coolant and the recirc loop coolant in the loop to be started is < 145 deg F.

66/88

REASON - To prevent damage to the fuel cladding that would result from the sudden increase in power due to the injection of cold water.

d) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

REASON - To prevent damage to the fuel cladding that would result from the sudden increase in power due to the injection of cold water.

Proposed Answer: b) LIMIT - The temperature difference between the recirc loop coolant in the loop to be started and the reactor vessel coolant is < 50 deg F.

REASON - To prevent a violation of the RPV pressure and temperature limitation that minimize the chances of brittle fracture from occurring.

Explanation (Optional):

Technical Reference(s): SD2P-24 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: 1.13.g (As available)

Question Source: Bank # Dresden 2 INPO # 21373

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 6/14/2002

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

10 CFR Part 55 Content:	55.41	X
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of how to conduct and verify valve lineups. (CFR: 41.10 / 45.1 / 45.12)	Group #		
	K/A #	2.1.29	2.1.29
	Importance Rating	3.4	3.3

Proposed Question: WHICH ONE (1) of the following conditions would allow waiving the independent verification requirements to verify a valve's position?

RO/SRO

67/89

a) If the valve was located in the Main Stack Building

b) If the valve was not associated with an ECCS system

c) If excessive radiation exposure would be required to verify the valve's position

d) If the valve had position indication in the control room

Proposed Answer: c) If excessive radiation exposure would be required to verify the valve's position

Explanation (Optional):

Technical Reference(s): AP-12.06 Section 7.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAP-48.03 (As available)

Question Source: Bank # Pilgrim 1 INPO # 18369 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 10/16/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of how to conduct and verify valve lineups. (CFR: 41.10 / 45.1 / 45.12)	Group #		
	K/A #	2.1.29	2.1.29
	Importance Rating	3.4	3.3

Proposed Question: WHICH ONE (1) of the following conditions would allow waiving the independent verification requirements to verify a valve's position?

RO/SRO
67/89

a ~~e)~~ If the valve was located in the Main Stack Building

b ~~f)~~ If the valve was not associated with an ECCS system

c ~~g)~~ If ~~it is estimated that 30 mRem~~ excessive radiation exposure would be required to verify the valve's position

d ~~h)~~ If the valve had position indication in the control room

Proposed Answer: c) If it is estimated that ~~30 mRem~~ would be required to verify the valve's position

Explanation (Optional):

Technical Reference(s): AP 12.06 Sec. 7.6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Pilgrim 1 INPO # 18369 (Mod to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 10/16/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1)	Group #		
	K/A #	2.2.1	2.2.1
	Importance Rating	3.7	3.6

Proposed Question: A plant startup is in progress with the Mode Selector Switch in Startup. Control rods are being withdrawn.

- The Rod Worth Minimizer (RWM) has just failed with 25% of the control rods withdrawn.

What action is required?

RO/SRO

68/90

a) Bypass the RWM, verify all further control rod movements are in compliance using a qualified person, and continue the reactor startup.

b) Suspend withdrawal of the control rods, manually SCRAM the reactor, and verify operability of the RWM before commencing a reactor startup.

c) Suspend withdrawal of the control rods, verify operability of the Rod Block Monitor, and continue the reactor startup.

d) Bypass the RWM, fully insert all control rods, and verify operability of the RWM before commencing a reactor startup.

Proposed Answer: a) Bypass the RWM, verify all further control rod movements are in compliance using a qualified person, and continue the reactor startup

Explanation (Optional):

Technical Reference(s): OP-64 Section E.1 (Attach if not previously provided)

OP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-3D, EO-1.15.A, LPAP, EO-46.04 (As available)

Question Source: Bank # Quad Cities 1 INPO # 20445 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Ability to perform pre-startup procedures for the facility / including operating those controls associated with plant equipment that could affect reactivity. (CFR: 45.1)	Group #		
	K/A #	2.2.1	2.2.1
	Importance Rating	3.7	3.6

Proposed Question:

A Plant Startup is in progress, Mode Selector Switch in

~~The following conditions exist on Unit 1: - Unit 1 is~~ in ~~the~~ STARTUP, ~~mode with~~ control rods being withdrawn in an approach to criticality. - The Rod Worth Minimizer (RWM) has just failed with 25% of the control rods withdrawn. ~~Per QCGP 1-1, NORMAL UNIT STARTUP,~~ What is ~~the~~ action ~~that~~ is required?

RO/SRO

68/90

a) Bypass the RWM, verify control rod movements are in compliance using a qualified person, and continue the reactor startup.

b) Suspend withdrawal of the control rods, place the reactor mode switch in the SHUTDOWN position within 1 hour, and verify operability of the RWM before commencing a reactor startup.

c) Suspend withdrawal of the control rods, verify operability of the Rod Block Monitor, and continue the reactor startup.

d) Bypass the RWM, fully insert all control rods, and verify operability of the RWM before commencing a reactor startup.

Proposed Answer: a) Bypass the RWM, verify control rod movements are in compliance using a qualified person, and continue the reactor startup.

Explanation (Optional):

Technical Reference(s): OP-64 Section E.1 (Attach if not previously provided)

OP-65

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Quad Cities 1 INPO # 20445

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	Group #		
	K/A #	2.2.12	2.2.12
	Importance Rating	3.0	3.4

Proposed Question: The Plant is operating at 90% Reactor power. The Control Room Supervisor has ordered you to perform ST-24J, RCIC Flow Rate and Inservice Test, following maintenance.

During RCIC pump operations:

RO/SRO

69/91

a) A manual SCRAM will be inserted if Torus water temperature exceeds 95 °F.

b) EHC Pressure Set will be adjusted to maintain RPV pressure < 970 psig.

c) Recirculation flow will be reduced to maintain Reactor power <100%.

d) Torus cooling will be in service to prevent Torus water temperature from exceeding 105 °F.

Proposed Answer: d) Torus cooling will be in service to prevent Torus water temperature from exceeding 105 °F.

Explanation (Optional):

Technical Reference(s): ST-24J, ITS-3.6.2.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-13, EO-1.13.d (As available)

Question Source: Bank # LaSalle 1 INPO # 19132 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of the process for determining the internal and external effects on core reactivity. (CFR: 43.6)	Group #		
	K/A #	2.2.34	2.2.34
	Importance Rating	2.8	3.2

Proposed Question: A Reactor startup from Cold Shutdown is in progress. The ECP was calculated based upon the following:

- Reactor Coolant temperature at 140 °F
- Total Core Flow at 10 X 10^6 lbm/hr
- At time of criticality, Reactor has been shutdown for 40 hours
- Feedwater temperature 80 °F

Which of the below will result in criticality later than the predicted ECP?

RO/SRO

70/92

a) Criticality occurs 30 hours after shutdown.

b) Feedwater temperature drops to 75 °F.

c) Total Core Flow is raised to 15 X 10^6 lbm/hr

d) Reactor Coolant temperature drops to 125 °F.

Proposed Answer: a) Criticality occurs 30 hours after shutdown.

Explanation (Optional):

Technical Reference(s): RAP-7.3.13 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPOP-65A, EO-1.10 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 1

55.43 6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of the process for performing a containment purge. (CFR: 43.4 / 45.10)	Group #		
	K/A #	2.3.9	2.3.9
	Importance Rating	2.5	3.4

Proposed Question:

RO/SRO

71/93

The plant is conducting a shutdown, power is currently 30% and lowering. It is desired to de-inert the Primary Containment (both the Drywell and Torus) as soon as possible to permit containment access for maintenance for a forced outage.

Which procedurally allowed flowpath and sequence would permit the most expeditious de-inerting of the Primary Containment?

a) Through the Standby Gas Treatment System with the Drywell and Torus de-inerted simultaneously.

b) Through the Reactor Building Ventilation System with the Drywell de-inerted first and then the Torus de-inerted.

c) Through the Reactor Building Ventilation System with the Drywell and Torus de-inerted simultaneously.

d) Through the Standby Gas Treatment System with the Drywell de-inerted first and then the Torus de-inerted.

Proposed Answer: d) Through the Standby Gas Treatment System with the Drywell de-inerted first and then the Torus de-inerted.

Explanation (Optional):

Technical Reference(s): OP-37 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDLP-1.06C, EO-1.13.C (As available)

Question Source: Bank # Quad Cities 1 INPO # 20444 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of EOP implementation hierarchy and coordination with other support procedures. (CFR: 41.10 / 43.5 / 45.13)	Group #		
	K/A #		2.4.16
	Importance Rating		4.0

Proposed Question:

In an emergency event the reactor scrammed due to high drywell pressure.

The following plant conditions exist:

- Drywell temperature SPDS display DWT "VERTICAL RUN TEMP" indicates 300 deg F.
- RPV pressure is 40 psig and equalized with the drywell.
- RPV water level indications are very erratic and do not correlate well with one another.

Under these circumstances, the operating crew would be required to execute the following Emergency Operating Procedure(s):

RO/SRO

S94

a) EOP-4, Primary Containment Control, ONLY.

b) EOP-2, RPV Control, AND EOP-4, Primary Containment Control, concurrently ONLY.

c) Initially, EOP-2, RPV Control, AND EOP-4, Primary Containment Control, concurrently, Then EOP-2, RPV Control, AND EOP-7, RPV Flooding, concurrently.

d) Initially EOP-2, RPV Control, AND EOP-4, Primary Containment Control, concurrently, THEN EOP-7, RPV Flooding, AND EOP-4, Primary Containment Control, concurrently.

Proposed Answer: d) Initially EOP-2, RPV Control, AND EOP-4, Primary Containment Control, concurrently, THEN EOP-7, RPV Flooding, AND EOP-4, Primary Containment Control, concurrently.

Explanation (Optional):

Technical Reference(s): EOP-2, 4, 7 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: EOP's (As available)

Learning Objective: EOP2LP, EO-1.02, 1.03, EOP4LP, EO-4.02

Question Source: Bank # JAF LOR # 20005204B04C

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)	Group #		
	K/A #	2.2.12	2.2.12
	Importance Rating	3.0	3.4

Proposed Question: The ~~plant~~ Control Room Supervisor supervisor has ordered you to perform a RCIC ~~operability test~~ testing following maintenance using ~~LO-RI-Q3, Reactor Core Isolation Cooling (RCIC) System Pump Operability and Valve Inservice Tests In Conditions 1, 2, and 3.~~ During the RCIC pump run, this surveillance would require the performance of ?

ST-24J

RCIC Flow Rate and Inservice Test

RO/SRO

69/91

a) Chemistry analysis on the ~~Suppression Pool~~ Torus water

b) ~~RCIC Monthly Valve Operability on the RCIC Exhaust Rupture Diaphragm~~

c) ~~Suppression~~ Torus Pool Temperature Monitoring ~~Checks~~.

d) Remote Shutdown Panel ~~Post-Accident~~ Instrumentation Operability Checks.

Proposed Answer: c) ~~Suppression Pool~~ Torus Temperature Monitoring ~~Checks~~.

Explanation (Optional):

Technical Reference(s): ST-24J (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank # 'LaSalle 1 INPO # 19132

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 11/20/2000

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

b) Monitoring Rx power to maintain ≈ 100% due to the rise in Feedwater temperature.

10 CFR Part 55 Content:	55.41	10
	55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10)	Group #		
	K/A #	2.3.1	2.3.1
	Importance Rating	2.6	3.0

Proposed Question: As a result of degrading plant conditions, the Shift Manager has directed you to immediately investigate an equipment problem inside a locked high radiation area. The duty RP technician is assisting workers in the plant stack and is not immediately available.

What action should be taken to expedite your entry?

RO/SRO

72/95

a) Using the key on any NPO Duty key ring.

b) Go to the RP office and sign out a key yourself.

c) Contact and meet the RP tech to obtain a key.

d) Obtain a radiological master key from the Shift Manager.

Proposed Answer: d) Obtain a radiological master key from the Shift Manager.

Explanation (Optional):

Technical Reference(s): AP-07.06 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: LPAP, EO-31.03.H (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of the process for performing a containment purge. (CFR: 43.4 / 45.10)	Group #		
	K/A #	2.3.9	2.3.9
	Importance Rating	2.5	3.4

Proposed Question: The Plant is conducting a shutdown, power is currently 30% and lowering.

~~The following conditions exist on Unit 1:~~

- ~~Unit 1 is in Mode 3.~~ It is desired to de-inert the ~~Unit 1~~ Primary Containment (both the drywell and torus) as soon as possible to permit containment access for maintenance for a forced outage. ~~Based on drywell and torus air samples, Chemistry recommends that the vent path be the Reactor Building Vents. Based on the above,~~ What flowpath and sequence would permit the most expeditious de-inerting of the ~~Unit 1~~ Primary Containment?

RO/SRO

71/93

a) With the Standby Gas Treatment System with the drywell and torus de-inerted simultaneously.

b) Through the Reactor Building Ventilation System with the drywell de-inerted first and then the torus de-inerted.

c) Through the Reactor Building Ventilation System with the drywell and torus de-inerted simultaneously.

d) With the Standby Gas Treatment System with the drywell de-inerted first and then the torus de-inerted.

Proposed Answer: ~~b) Through the Reactor Building Ventilation System with the drywell de-inerted first and then the torus de-inerted.~~

Explanation (Optional): d With Standby Gas Treatment sys. with the drywell de-inerted first then the torus de-inerted

Technical Reference(s): OP-37 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank # Quad Cities 1 INPO # 20444

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 8/13/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR: 43.5 / 45.11)	Group #		
	K/A #		2.4.30
	Importance Rating		3.6

Proposed Question: As a result of an error during I/C Surveillance testing at 100% power, the MSIV's inadvertently closed resulting in the following:

- A Full SCRAM on MSIV closure.
- HPCI initiation and injection.
- HPCI tripped by the Operators.
- Operator control of level using Feed and Condensate.
- Operator control of RPV pressure using SRV's.

Which of the below is the required NRC report?

RO/SRO

S96

a) Immediate Notification due to an Emergency Plan event declaration.

b) One (1) Hour Notification due to a deviation from Technical Specifications authorized by 10CFR-50.54 (X).

c) Four (4) Hour Notification due to ECCS discharge to Reactor Coolant System resulting from a valid signal.

d) Eight (8) Hour Notification due to a valid Containment Isolation signal affecting Containment Isolation Valves.

Proposed Answer: c) Four (4) Hour Notification due to ECCS discharge to Reactor Coolant System resulting from a valid signal.

Explanation (Optional):

Technical Reference(s): ENN-LI-102, AP-03.11 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: AP-03.11

Learning Objective: LPAP-10.06 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New NEW

Question History: Last NRC Exam

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

RO — SRO S96

ENTERGY NUCLEAR OPERATIONS, INC.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ADMINISTRATIVE PROCEDURE

OPERABILITY AND REPORTABILITY DETERMINATIONS
AP-03.11
REVISION 9

APPROVED BY: ______________________ DATE 5/29/02
RESPONSIBLE PROCEDURE OWNER

APPROVED BY: ______________________ DATE 5/29/02
GENERAL MANAGER PLANT OPERATIONS

EFFECTIVE DATE: 6-3-02

FIRST ISSUE ☐ FULL REVISION ☐ LIMITED REVISION ☒

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ADMINISTRATIVE	

PERIODIC REVIEW DUE DATE July, 2006 ~~June, 2007~~ 6 dnh 6-4-02

REVISION SUMMARY SHEET

REV. NO.	CHANGE AND REASON FOR CHANGE
9	Where applicable, indicated steps and paragraphs which address Attachments 2, 7, and 8. ACT-02-62578 (PCR dated 1/15/02)

Changed "Director Engineering" to "Engineering Manager(s)" throughout procedure. ACT-02-62577 (PCR dated 1/15/02)

Included "Entergy Licensing Position #2," on Attachment 8, as Management Expectation 4.2.5.

Added cross-references where applicable.

To Section 7.0, SPECIAL INSTRUCTIONS, added information on using the attachments not already listed which only provide guidance for filling out the attachments, which are forms. This provides a "pointer" for the procedure user to use to go to the needed attachment.

To 8.2.7, added direction to record the PLCO number on Attachment 1, providing clarification to the procedure user.

Made minor changes to 8.2.7.B to better locate the information on Contingent Operator Actions.

To 8.2.8, added clarification to complete the 10CFR50.59 Screen per MCM-4.1, using the guidance in the rest of this step and in Attachment 8.

8.2.8.C - changed "initiation of" to "implementing" for clarification.

8.4 - Added a note at the head of this subsection stating the purpose and use of Attachment 9.

8.4.4.B - Made it easier to locate information for Contingent Operator Actions in Attachment 6.

Made minor typographical, spelling, and formatting changes throughout procedure. Rev. bars not used.

Updated procedure references. DCM-2A & 4A have been replaced by ENN-DC-126.

Deleted reference to Asst Ops Manager review requirements. They were intended as in interim measure and are no longer desired. Maintains alignment with other ENN sites, where the SM is the final approval authority. Deletes responsibility in Section 6, Subsection 8.5, & Attachment 1.

Made the following changes to address reportability in this procedure due to the deletion of AP-03.02 and AP-03.03:

- Where applicable, addressed ENN-LI-102, EAP-1.1, and JAF Corrective Action Process Desk Guide.
- Revised the procedure title, purpose statement and applicability to address immediate reportability.
- Substituted the term "Condition Report (CR)" for "DER" and "Corrective Action (CA)" for "ACT," consistent with the new Paperless Condition Reporting System (PCRS). Rev. bars not used.
- Deleted Performance References AP-03.02 and AP-03.03 due to their deletion and added references to ENN-LI-102, EAP-1.1, JAF Corrective Action Process Desk Guide, AP-03.04, Technical Requirements Manual, NRC IN 89-89, NUREG-1022, and 10CFR26, Fitness for Duty Programs.
- Added new Expectation 4.2.5, Entergy Licensing Position #2, Evaluation and Resolution of Degraded and Nonconforming Conditions.
- Added definitions for Immediate Notification, Reportable Event, and Safety Function.
- Modified 6.4 Operations Manager and 6.6 Shift Manager responsibilities to address reportability.
- Added Director Safety Assurance, CR Screening Committee, and Regulatory Compliance Manager to 6.0 Responsibilities.
- In 7.0 Special Instructions, added requirement to immediately notify SM for certain conditions and guidance for SM to prepare Operability Determinations and Immediate Reportability Determinations.
- In 7.0, addressed the purpose and use of the attachments.
- In Section 8.0, added a subsection for Operability and Reportability Review.

- In Section 8.0, added a new subsection 8.4 for Immediate Reportability Determinations.
- Modified Section 8.0 to address reportability determinations.
- Added a new Attachment 7, Reportability Checklist. Renumbered the remaining attachments. Rev. bars not used.
- Changed Atts. 9 and 10 to accurately address the operability determination process.

TABLE OF CONTENTS

EXP4.2.5

1.0 PURPOSE

1.1 To establish a method for determining the operability of structures, systems, or components (SSCs) that have been identified as being in a degraded or nonconforming condition.

1.2 In addition, to establish a method for determining the potential reportability to outside regulatory agencies of issues identified via the corrective action process.

2.0 APPLICABILITY

To the performance of Operability Determinations and Immediate Reportability Determinations for Problem Identifications (PIDs) and Condition Report (CRs); however, it may be used at the Shift Manager's discretion for performing Operability Determinations and Immediate Reportability Determinations for other concerns.

3.0 REFERENCES

3.1 Performance References

3.1.1 AP-01.01, Plant Operating Review Committee

3.1.2 **[CTS]** AP-01.04, Tech Spec Related Requirements, Lists, and Tables

[ITS] Technical Requirements Manual

3.1.3 AP-02.08, Quality Assurance Record Identification and Control

3.1.4 AP-03.04, Information Reporting Requirements

3.1.5 AP-10.01, Problem Identification and Work Control

3.1.6 EAP-1.1, Offsite Notifications

3.1.7 ENN-LI-102, Corrective Action Process

3.1.8 JAF Corrective Action Process Desk Guide

3.1.9 **[CTS]** ODSO-34, Technical Specification LCO and Maintenance Rule Unavailability Tracking

[ITS] AP-12.08, LCO Tracking and Safety Function Determination Program

3.1.10 ENN-DC-126, Calculations

3.1.11 DCM-14A, Preparation and Control of Computer Generated Calculations (JAF)

3.1.12 NRC Generic Letter 90-05, Guidance for Performing Temporary Non-code Repair of ASME Code Class 1, 2, and 3 Piping

3.1.13 NRC Generic Letter (GL) 91-18, Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability

3.1.14 NRC Inspection Manual, Part 9900: Technical Guidance

3.1.15 NRC IN 89-89, Emergency Notification System (ENS)

3.1.16 NUREG-1022, Event Reporting Guidelines 10CFR50.72 and 10CFR50.73 (Revision 2, October 2000)

3.1.17 JAF Updated Final Safety Analysis Report (UFSAR)

3.1.18 JAF Design Basis Documents

3.1.19 JAF Technical Specifications

3.1.20 ENN-LI-100, Process Applicability Determination

3.2 **Developmental References**

3.2.1 10CFR26, Fitness For Duty Programs

3.2.2 NuAP 4.12, Resolution of Equipment Operability Concerns Related to Degraded or Nonconforming Conditions

3.2.3 NUREG-1433, Standard Technical Specifications, General Electric Plants / BWR 4s

3.2.4 ENN-LI-101, 10CFR50.59 Review Process

3.2.5 MCM-4.2, 10CFR50.59 Evaluations

3.2.6 W4.101, Waterford 3 Management Manual Procedure Operability Confirmation Process

3.2.7 Entergy Licensing Position #2, Evaluation and Resolution of Degraded and Nonconforming Conditions

4.0 REQUIREMENTS

4.1 Regulations, Codes, and Standards

4.1.1 Technical Specification Section 1.0, Definitions

4.1.2 10CFR50, Appendix B, Criterion XVI, Corrective Action

4.2 Expectations

4.2.1 DER-96-0325, ACTS Item 22418, added steps to discuss and invoke the DCM process for Operability Determinations.

4.2.2 DER-97-0680, ACTS Item 26467, revised procedure to require that a Potential LCO be entered when an Operability Determination justifies continued operation, but requires any action(s) to be taken following plant shutdown.

4.2.3 DER-98-02118, ACT-98-35976, revised procedure to require an engineering peer review be obtained prior to using a previously performed calculation as a basis for operability.

4.2.4 DER-00-02054, ACT-00-51185, clarified the need for both engineering supervision and the SM to challenge the scope and assumptions of Engineering Confirmations prior to making operability determination. Identified the need to clarify the concept of "Timeliness Commensurate with Safety Significance".

4.2.5 Entergy Licensing Position #2, Evaluation and Resolution of Degraded and Nonconforming Conditions.

5.0 DEFINITIONS

5.1 Compensatory Actions:

Interim measures prudently taken to improve assurance that specified safety functions are being maintained during the process from initial degraded/nonconforming condition discovery until completion of necessary corrective actions. Compensatory measures shall address time requirements (e.g., duty time of RHR is 180 days post accident) established in the current licensing basis.

5.2 Degraded Condition (See Attachment 8):

A condition of an SSC in which there has been any loss of quality or functional capability. Examples of Degraded Conditions are:

- Performance trend of an IST component that indicates an ALERT or ACTION level will be reached prior to the next scheduled surveillance test
- Leaks external to systems (e.g., steam, water oil)
- Noticeable increases in parameters that are precursors to failure (e.g., vibration, noise, temperature)
- Restricted flow of cooling media or process fluid to heat exchangers
- High resistance electrical contacts due to pitting, oxidation, etc.

5.3 Design Basis:

Information which identifies the specific functions to be performed by an SSC, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

5.4 Design Basis Event:

Those accidents and abnormal operational transients for which safety analysis are described in Chapter 14 of the UFSAR and are part of the licensing basis for the plant.

5.5 Design Function:

UFSAR described design bases functions and other SSC functions described in the UFSAR that support or impact design bases functions. Maintenance Rule functions can be considered design functions.

5.6 Engineering Confirmation (See Attachments 2, 3 and 9):

The engineering evaluation process by which the validity of the Operability Determination performed by the Shift Manager is verified.

5.7 Full Qualification:

Conformance to all aspects of the current licensing basis, including codes and standards, design criteria, and commitments.

5.7.1 Qualification is the documented verification that provides assurance that an SSC or equipment has been designed, procured, tested, installed, etc., to ensure it is capable of performing its specified function under all conditions as assumed in applicable safety analyses (e.g., UFSAR, Fire Protection Programs, etc.).

5.7.2 Qualification constitutes conformance to all aspects of the licensing basis including licensee commitments (e.g., codes and standards referenced in the UFSAR, other UFSAR commitments, and/or corrective action commitments from LERs or NRC Notices of Violation).

5.8 Immediate Notification (See Attachment 7):

Notification of the NRC within 1 hour (in some cases, within 4, 8, or 24 hours) per the Code of Federal Regulations.

5.9 Licensing Basis:

Documents used to grant, amend, or modify the operating license and Technical Specifications and to ensure continued compliance with, and operation within, applicable NRC requirements. Licensing Basis includes, but are not limited to, the UFSAR (including documents referenced therein), **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual, NRC safety evaluation reports, LERs, Generic Letters, Bulletins and similar-type docketed correspondence.

5.10 Nonconforming Condition (See Attachment 6):

An adverse condition affecting a safety-related, quality-related, or trip sensitive system caused by a deficiency in characteristic, documentation, or procedure which renders the quality of an item unacceptable or indeterminate. Examples of nonconforming conditions are:

- Item does not conform to design/license basis.
- Recurring or generic failure.
- Item has a physical defect as a result of design or manufacturing process that prevents or could have prevented the component from performing its intended function.
- Item fails testing performed to prove environmental, seismic, or design conformance.
- Deviation from prescribed processing or inspection.
- Documentation not available to confirm required inspections or tests.
- M&TE out of calibration: A Condition Report is not required when the non-conforming condition is related to the calibration of M&TE and the condition is resolved through a record search, with the determination that plant hardware or system performance is not affected and no further action is required.
- Missed or late preventative maintenance task required to satisfy technical specifications, environmental qualification, or station commitments.
- Conditions where nuclear fuel defects exist or are suspected.

5.11 Operable / Operability:

[CTS]
A system, subsystem, train, component, or device shall be Operable or have Operability when it is capable of performing its specified functions, and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its functions are also capable of performing their related support functions. Otherwise, the system, subsystem, train, component, or device is Inoperable.

[ITS]
A system, subsystem, division, component, or device shall be Operable or have Operability when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

5.12 Operability Determination Status:

The status of an Operability Determination. Following are the types of status:

5.12.1 Active:

The Shift Manager has made an immediate determination of Operable; however, there are conditions or future actions needed to allow the determination to be completed. An Operability Determination is considered Active and a Potential LCO is entered for the following reasons:

- Actions required following plant shutdown.

- Actions require monitoring plant conditions.

  The Operable status is conditional based on certain factors that, if changed, may render the equipment inoperable (i.e., "Operable provided lake water temperature does not exceed 79°F" or "Operable provided leakage does not exceed 5.0 GPM").

- Compensatory measures needed to justify continued Operable status.

  The Operable status is based on compensatory measures that are taken to ensure the equipment can be considered Operable; however, actions are being taken to restore the original design. Measures may include manual operator actions, where allowed, or establishment of administrative controls.

- The Shift Manager has declared equipment Operable with a Reasonable Expectation of Operability (REO), and an Engineering Confirmation is required to develop a final basis for a completed status.

5.12.2 Completed:

The Shift Manager, based on clear, confirmed evidence sufficient to justify the determination, has made an equipment determination of Operable or Inoperable. A completed status means there are no conditional bases or compensatory measures required, and the Engineering Confirmation in support of a previous REO is sufficient to now complete the determination.

5.13 Reasonable Expectation of Operability (REO):

Technical judgment, coupled with the safety significance of a degraded or nonconforming condition, which gives a reasonable indication that an SSC is capable of performing its specified functions. This constitutes a prompt determination of Operability, but indicates that further evaluation is required in support of Operability.

5.14 Reportable Event (See Attachment 7):

Event that requires a notification to the Nuclear Regulatory Commission or other regulatory bodies or a Licensee Event Report (LER) to the NRC per license requirements and 10CFR parts 20, 21, 26, 50, 50.72, 50.73, 70, 72, and 73.

5.15 Safety Function:

Those functions required to prevent the unacceptable consequences for design basis events. The unacceptable consequences for design basis events are loss of:

- The integrity of the reactor coolant pressure boundary.
- The capability to shut down the reactor and maintain it in a safe shutdown condition.
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

5.16 Systems, Structures, or Components (SSCs):

Systems, structures, or components, which are:

5.16.1 Safety-related and relied upon to remain functional during and following design basis events to:

A. Ensure the integrity of the reactor coolant pressure boundary.

B. Ensure the capability to shut down the reactor and maintain it in a safe shutdown condition.

C. Ensure the capability to prevent or mitigate the consequences of accidents that could result in potential offsite consequences comparable to the 10CFR100 guidelines.

5.16.2 Relied on in the safety analyses or plant evaluations that are a part of the plants current licensing basis. Such analyses and evaluations include those submitted to support license amendment requests, exemption requests, or relief requests, and those submitted to demonstrate compliance with the NRC's regulations.

5.16.3 Subject to 10CFR50, Appendix A (Criteria 1) or 10CFR50, Appendix B.

5.16.4 Subject to Technical Specifications, either explicitly or through the definition of operability, such as supporting systems.

5.16.5 Described in the Updated Final Safety Analysis Report (UFSAR).

⇓EXP4.2.4

5.17 Timeliness Commensurate with Safety Significance (See Attachment 8):

Operability evaluations must be conducted in a time frame consistent with the safety significance of the identified condition. An evaluation should be completed within the established LCO period, if the SSC has an associated LCO. When there is no specified LCO, the safety significance should be considered based on the risk impact. Insights as to risk impact can be obtained from Planning or the PRA Group in WPO Reactor Engineering.

5.18 UFSAR Specified Functions:

Those functions that the SSC performs that are identified in the UFSAR. This is not limited to safety related functions. System-specific functions are summarized in MCM-6A. The UFSAR sections referenced provide additional information that should be consulted to identify UFSAR specified functions.

6.0 RESPONSIBILITIES

6.1 All Personnel

Identify equipment of questionable conformance / qualification as it becomes known by initiating a CR or PID and communicating this information to the Shift Manager.

6.2 General Manager Plant Operations (GMPO)

6.2.1 Review Initial Engineering Confirmations whenever any compensatory actions are required.

6.2.2 Review Detailed Engineering Confirmations.

⇓EXP4.2.4

6.3 Engineering Managers

6.3.1 Review Initial Engineering Confirmations whenever any compensatory actions are required.

6.3.2 Review Detailed Engineering Confirmations.

6.4 Operations Manager

NOTE: For the purpose of this procedure, Operations Manager is considered to be the Operations Manager, the Assistant Operations Manager, or a designated SRO.

6.4.1 Assist the Shift Manager in determining SSC operability.

6.4.2 Review completed Engineering Confirmations to determine operability of affected SSCs.

6.4.3 Ensure Active status Operability Determinations are presented for PORC review.

6.4.4 Provide operability and reportability review of CRs, as applicable.

6.4.5 Concur with operability and reportability determination of CRs.

6.5 Shift Manager (SM)

6.5.1 Monitor the operational readiness of SSCs important to the safe operation of the facility.

6.5.2 Determine the operability of SSCs based on the information available.

6.5.3 Determine Reasonable Expectation of Operability. Notify Operations and System Engineering Manager of status and request Engineering Confirmations when needed.

6.5.4 Track active status Operability Determinations as Potential LCOs.

6.5.5 Ensure appropriate compensatory actions are taken when SSCs important to safety are degraded or Inoperable.

6.5.6 Review CRs in the SM PCRS Inbox to ensure operability and immediate reportability requirements are met.

6.5.7 Ensure notifications are made.

6.6 Director Safety Assurance

Make notifications to NRC when on site during day shift.

6.7 System Engineering Manager

6.7.1 Assign responsibility for preparation of Engineering Confirmations to the appropriate Engineering Supervisor on site or in WPO.

6.7.2 Ensure the Engineering Supervisor has sufficient resources to perform the Engineering Confirmation as outlined in this procedure.

6.8 Engineering Supervisor

6.8.1 Provide the System Engineering Manager with results of assigned Engineering Confirmations in a timely manner.

⇓EXP4.2.1

6.8.2 Ensure calculations and analyses to support Operability Determinations are performed per ENN-DC-126.

6.8.3 Keep the System Engineering Manager, Shift Manager, and NRC Resident Inspector informed of any changes to the established schedule or the results of the Engineering Confirmation, as appropriate.

6.8.4 Assign independent reviewers to assess the adequacy of the Engineering Confirmation scope, logic, and supporting technical analysis.

6.9 **Plant Operating Review Committee (PORC)**

6.9.1 Review Active status Operability Determinations regularly.

6.9.2 Review Initial Engineering Confirmations whenever any compensatory actions are required.

6.9.3 Review Detailed Engineering Confirmations.

6.10 **Condition Report (CR) Screening Committee**

Review CRs and PIDs for operability and immediate reportability concerns.

6.11 **Regulatory Compliance Manager**

Provide reportability review of CRs which are potentially reportable.

7.0 SPECIAL INSTRUCTIONS

7.1 A condition having an immediate impact on plant safety, involves a fire or an uncontrolled release of radioactivity, or poses a threat to security shall be immediately reported to the Shift Manager.

7.2 The Shift Manager prepares Operability Determinations and Immediate Reportability Determinations as required and obtains assistance from other departments as necessary.

7.2.1 Operability and Immediate Reportability determinations and reviews are performed in accordance with ENN-LI-102, EAP-1.1, JAF Corrective Action Process Desk Guide, and this procedure.

7.2.2 SMs/SROs may issue CAs to engineering groups for support of Operability Determinations or Evaluations without the concurrence of the responsible individual or group.

7.3 The Shift Manager ensures component and system operability continually by surveillances and formal determinations. Operability Determinations are supplemented by:

- Day-to-day facility operation
- Implementation of ISI/IST programs
- Plant walkdowns and tours
- Control room observations
- QA audits and reviews
- Engineering design reviews

7.4 Questions or concerns relating to SSC qualification are resolved using CRs per ENN-LI-102.

7.5 When an Operability Determination or Immediate Reportability Determination is "completed," it is considered closed and any associated corrective actions required to restore a degraded or non-conforming condition are tracked to completion using the other processes for such actions, including work requests per AP-10.01 or an analysis or evaluation per ENN-LI-102 and the JAF Corrective Action Process Desk Guide.

7.6 Electronic forms are acceptable for use, as long as the forms contain all the necessary information.

7.7 The following attachments provide information and flowcharts that are helpful in understanding the requirements of this procedure. However, the flowcharts do not contain a sufficient level of detail to fully implement the procedure.

- Attachment 1 provides the form and guidelines for operability determination by the SM.
- Attachment 2 provides the Engineering Confirmation Summary Form.
- Attachment 3 provides Initial Engineering Confirmation guidelines.
- Attachment 4 provides Detailed Engineering Confirmation guidelines.
- Attachment 5 provides REO/Reasonable Assurance guidelines.
- Attachment 6 provides REO/Engineering Operability guidelines.
- Attachment 7 provides an Immediate Reportability Checklist.
- Attachment 8 provides the Entergy Licensing Position regarding Evaluation and Resolution of Degraded and Nonconforming Conditions.
- Attachment 9 contains a flowchart overview of the Engineering Confirmation process.
- Attachment 10 contains a flowchart overview of the Operability Determination process.

8.0 PROCEDURE

8.1 General Requirements

8.1.1 The Shift Manager shall ensure SSCs are considered either Operable or Inoperable at all times. An REO determination means an SSC is considered operable.

8.1.2 If the operability of an SSC is questionable, the Shift Manager shall make an Operability Determination and an Immediate Reportability Determination, with immediate and primary attention directed to safety concerns.

8.1.3 Operability Determinations and Immediate Reportability Determinations shall be made promptly, with a timeliness commensurate with the potential safety significance of the issue. Attachment 8 provides guidance for understanding the issue of timeliness.

8.1.4 As requested by the Shift Manager, others may assist in the Operability and Immediate Reportability determinations; however, the final determinations are made by the Shift Manager.

⇓EXP4.2.5

8.1.5 The Shift Manager shall review the following conditions or events which may require an Operability Determination and/or an Immediate Reportability Determination:

A. Degraded equipment condition where performance is called into question.

B. Nonconforming condition where equipment qualification is called into question, such as, non-safety related parts found in a safety related application.

C. Fitness for Duty events may require an Immediate Reportability Determination.

8.2 Operability and Immediate Reportability Reviews

8.2.1 The Shift Manager shall review all CRs that are flagged as potentially affecting operability or as potentially reportable by the CR initiator. (See Attachment 1 for directions for Operability Determinations and Attachment 7 for Immediate Reportability Determinations.)

8.2.2 The CR Screening Committee shall review CRs for potential operability and potential immediate reportability considerations.

8.2.3 Operability Determinations are performed and documented in accordance with this procedure. A summary of the determination should be entered into the Operability tab in PCRS.

8.3 Operability Determinations and Related Actions

8.3.1 Operability Determinations

NOTE: If a deficiency is documented on both a PID and CR, only one Operability Determination is required. Preferably, a CR is used.

A. The Shift Manager shall initiate an Operability Determination using Attachment 1 for CRs and for PIDs identified as corrective maintenance, for the following (Refer to Step 5.16):

- SSCs which are QA Category I or M.
- SSCs which are not QA Category I or M but have operability requirements specified in Technical Specifications or **[CTS]** AP-01.04 **[ITS]** Technical Requirements Manual.
- SSCs described in the UFSAR that support the operability of SSCs which have operability requirements specified in Technical Specifications or **[CTS]** AP-01.04 **[ITS]** Technical Requirements Manual.

B. The Shift Manager shall initiate an Immediate Reportability Determination using Attachment 7 for CRs classified as a potential operability concern.

C. The Shift Manager should consider initiating an Operability Determination if the condition involves:

- Fire Protection Program requirements
- Seismic/Equipment Qualification
- Emergency Operating Procedures
- Accident Analysis
- Refuel Operations
- Surveillance Test Procedures
- Radwaste Effluent Controls
- Equipment that could result in a plant trip

8.3.1 (cont)

D. Track each Operability Determination based on the associated CR or PID number. Active status determinations are numbered and tracked to completion as Potential LCOs per ~~ODSO-34~~ AP-12.08.

NOTE: An SSC with discrepant conditions can be considered functional or operable until available information indicates the SSC is nonfunctional or inoperable.

E. In the absence of reasonable assurance that an SSC is operable, or if evidence suggests that final evaluation will conclude the SSC can <u>not</u> perform its specified design or safety functions, the Shift Manager shall declare the SSC Inoperable.

F. If Engineering assistance is required to complete an Operability Determination, and the Shift Manager has a reasonable expectation of operability, proceed as follows:

1. Complete Attachment 1 as an REO.
   - Use Subsection 8.3.2 and Attachments 5 and 6 for guidance.
   - Track as a Potential LCO per ~~ODSO-34~~ AP-12.08
2. Notify the Operations Manager and System Engineering Manager, or designees, immediately.

G. Initiate any required compensatory or corrective actions. Equipment used in compensatory actions should be controlled per AP-12.01 and AP-12.06, using the guidance in Attachment 8.

H. Document the justification for Operability Determinations on Attachment 1.

8.3.1 (cont)

⇩EXP4.2.2

I. If the Operability Determination justifies continued operation, but requires any of the following, enter a Potential LCO per AP-12.08, mark Attachment 1 in this procedure as Active, and record PLCO number on Attachment 1:

1. Actions to be taken following plant shutdown.

2. Conditional Operability Determinations requiring monitoring of plant conditions. Refer to the section in Attachment 6 which provides guidance on Contingent Operator Actions.

3. Compensatory measures taken to justify continued equipment operable status.

J. When compensatory measures are used to declare SSCs Operable, the Shift Manager shall ensure a 50.59 Screen Control Form is completed per ENN-LI-101, using the following guidance and the information in Attachment 8:

1. The screen may be completed as part of a procedure change or temporary modification.

2. The screen should focus on the effects of the compensatory measures, <u>not</u> the degraded condition.

3. The screen should be completed and reviewed prior to implementing of such measures.

4. If a 10CFR50.59 Evaluation is required, the Shift Manager should re-evaluate the original identified deficiency per Steps 8.3.1.D and 8.3.1.E.

8.3.1 (cont)

K. The Shift Manager shall ensure Operability Determinations are sufficient to address SSC capability to perform its safety functions. Determinations may include:

- Determining safety functions performed by the SSC by reviewing Technical Specifications, **[CTS]** AP-01.04 **[ITS]** Technical Requirements Manual, MCM-6A Reference Document (MCM-6A-REF), UFSAR, and Design Basis Documents.

- Determining circumstances of the non-conformance, including possible failure mechanism.

- Determining requirement established for the equipment and why the requirement may not be met.

- Determining the safest plant configuration, including effects of transitional actions.

NOTE: Successful performance of Technical Specification surveillance requirements alone is usually not sufficient to determine operability when conformance to the appropriate criteria in the current licensed design basis is in question.

L. Describe basis for conclusion on Attachment 1. The Shift Manager should use the following methods to make an operability decision:

- A test, partial test or other functional demonstration
- Analysis
- Past experience with operating events for this SSC
- Engineering judgment

M. Probabilistic Risk Assessment (PRA) or probabilities of the occurrence of accidents or events shall not be used to determine Operability.

N. If, during the operability review, a CR is required, the Shift Manager shall generate a CR per ENN-LI-102 and the JAF Corrective Action Process Desk Guide.

8.3.1 (cont)

O. When a (item) system, subsystem, train, component or device addressed in a Technical Specification LCO becomes inoperable:

1. Verify operability of redundant counterparts.

2. Verify affect of inoperability on supported or supporting items.

P. Enter the results of the initial Operability Determination on a new operability record from the Operability tab in PCRS. If there are no reportability issues, the SM can sign the PCRS record.

8.3.2 **Reasonable Expectation of Operability (REO)**

A. If the impact on the associated SSCs is not apparent, or the determination requires Engineering input, then the SM (or SRO/STA) shall perform an REO evaluation.

1. If calculations, vendor information, etc., can not be obtained to substantiate the Operability Determination, then the SM (or SRO/STA), with assistance from the appropriate engineering department (System Engineering, Design Engineering, Component/Programs Engineering, etc.), will make the operability evaluation using best technical judgment.

2. Engineering departments that provided documented assistance should attach those documents to the assessment.

3. Document the bases for the REO.

4. The time of entry into the Engineering Confirmation process that follows the REO will be indicated.

5. The basis should include a statement regarding the capability of the equipment/system/train being evaluated to perform its UFSAR Specified Functions.

6. The Shift Manager shall either perform or review the operability evaluation bases.

B. Guidance contained in Attachments 5 and 6 should be used during performance of the REO.

8.3.3 **Engineering Confirmation**

NOTE: Attachment 9 provides a flowchart for the Engineering Confirmation process and should be used as a guide for this subsection.

A. The responsible engineer should develop the Initial Engineering Confirmation per Attachment 3.

1. If the Initial Engineering Confirmation will not be completed as scheduled, then the Engineering Supervisor shall obtain approval for an extension from the SM. Notify the System Engineering Manager.

B. If at any time during the development of the Engineering Confirmation, it is determined, through evaluation or engineering judgment, that the SSC is not Operable, then notify the SM immediately.

C. If the SSC required by Technical Specifications is determined to be Inoperable, then the SM shall comply with the Technical Specification requirements.

1. If necessary, the SM should consult with the GMPO, Engineering Manager(s), and Regulatory Compliance Manager to determine if an Emergency Technical Specification Amendment or a Notice of Enforcement Discretion is appropriate.

D. If SSCs important to safety other than that required by Technical Specifications are determined to be Inoperable, then the Initial Engineering Confirmation should address if continued operation is recommended, with appropriate justification and recommendations.

1. The Initial Engineering Confirmation should consider administrative controls or other compensatory actions that can be taken.

2. If compensatory actions are recommended, perform a Process Applicability Determination per ENN-LI-100 and a 50.59 Screen Control Form per ENN-LI-101. Refer to the Section in Attachment 6 which provides guidance on Contingent Operator Action(s).

8.3.3 (cont)

E. The Engineering Supervisor should keep the System Engineering Manager, SM, and NRC Resident Inspector informed of any change or significant development as the Initial Engineering Evaluation is processed.

F. If a Detailed Engineering Confirmation is needed, then it should be identified in the Initial Engineering Confirmation.

1. The responsible engineer will determine the approach, scope, and responsibilities for the Detailed Engineering Confirmation. The schedule for completion of the Detailed Engineering Confirmation should be approved by the GMPO and Engineering Manager(s).

2. The Detailed Engineering Confirmation should be completed per the guidelines in Attachment 4 and attached to the CR.

3. The Engineering Supervisor should notify the SM, System Engineering Manager, Engineering Manager(s), and GMPO when the Detailed Engineering Confirmation is completed.

G. PORC and the GMPO should review the following:

- Initial Engineering Confirmations whenever any compensatory actions are required
- All Detailed Engineering Confirmations

H. The responsible engineer shall ensure the Engineering Confirmation (Initial or Detailed) is provided to the SM for review and acceptance.

1. After the SM accepts the Engineering Confirmation, the date and time of the Engineering Confirmation and exiting of the AP-03.11 process shall be entered into the station logs.

EXP4.2.1

I. Calculations or analyses required to support the completed Operability Determination shall be prepared per ENN-DC-126.

8.3.3 (cont)

⇓EXP4.2.3

J. If a previously performed calculation is used as a basis for operability, obtain an engineering peer review to verify applicability of the calculation.

K. If an Engineering Confirmation is performed to determine system operability of through-wall leaks within the ISI boundary, ensure the flaw evaluation is performed using NRC-GL-90-05 methodology.

L. Engineering Confirmations shall be documented using Attachment 2, following the guidance in Attachments 3 and 4, as appropriate. In addition, the guidance in NRC Inspection Manual (Part 9900) and Generic Letter 91-18, Revision 1, should be used.

1. The preparing engineer signs and dates the completed evaluation.

2. The independent reviewer sign and date the completed evaluation.

3. Independent review is performed by individuals not involved with preparing the evaluation and consists of assessing the adequacy of the Engineering Confirmation scope, logic, and supporting technical analysis.

4. The Engineering Supervisor reviews and signs the evaluation.

   ⇓EXP4.2.4
   The intent of this review is to challenge the scope and assumptions of the evaluation prior to submitting to the SM for review and acceptance.

M. Probabilistic Risk Assessment (PRA) or probabilities of the occurrence of accidents or events shall not be used to determine Operability.

8.3.3 (cont)

⇓EXP4.2.4

N. Completed evaluations shall be forwarded to the Shift Manager for review. The intent of this review is to challenge the scope and assumptions of the evaluation prior to making an operability determination.

1. If the evaluation supports the operable status, then the SM performs the following:

   - Declare equipment Operable on Attachment 1.

   - Mark Attachment 1 as Complete.

2. If the evaluation does <u>not</u> support an operable status, then the SM performs the following:

   - Declare equipment Inoperable on Attachment 1 and initiate required actions for declaration of Inoperable.

   - Mark Attachment 1 as Complete.

3. If the evaluation supports Operable with compensatory measures, then the SM performs the following:

   - Declares equipment Operable on Attachment 1 and initiates required actions for declaration of Operable.

   - Marks Attachment 1 as Active.

8.3.4 **Revisions to Active Operability Determinations**

A. If active Operability Determinations (or Engineering Confirmations) require revision before being completed, the Engineering Supervisor presents the required changes to the SM for acceptance. The SM or Engineering Supervisor (as appropriate) may perform either of the following:

1. Update the original documents and initial changes made.

2. Complete a new Operability Determination (or Engineering Confirmation) and attach the superseded documents.

3. If needed, another operability record can be generated from the Operability tab in PCRS.

8.4 **Immediate Reportability Determinations**

8.4.1 Immediate Reportability Determinations are performed and documented in accordance with Attachment 7 and JAF Corrective Action Process Desk Guide. A summary of the determination should be entered into the Operability tab in PCRS. Refer to NUREG-1022, Event Reporting Guidelines 10CFR50.72 and 10CFR50.73, for additional guidance on reportability determinations.

8.4.2 If an event is determined to require an Immediate Report, Operations shall complete the Event Notification Worksheet, NRC Form 361 (a copy exists in EAP-1.1) and make the proper notifications in accordance with EAP-1.1.

8.4.3 For Independent Spent Fuel Storage Installation (ISFSI) events that are reportable under 10CFR72.75, to the extent that the information is available at the time of notification, the Event Notification Worksheet, NRC Form 361, (a copy exists in EAP-1.1) shall include a description of the quantities and physical forms of the spent fuel or high level waste involved.

8.5 **Management Review of Operability**

↓EXP4.2.5

8.5.1 The Engineering Manager(s) shall review Initial Engineering Confirmations (whenever compensatory actions are required) and Detailed Engineering Confirmations to ensure timeliness is commensurate with safety significance and technical adequacy is consistent with management expectations.

↓EXP4.2.5

8.5.2 The GMPO shall review Initial Engineering Confirmations (whenever compensatory actions are required) and Detailed Engineering Confirmations to ensure timeliness is commensurate with safety significance and technical adequacy is consistent with management expectations.

8.5.3 The Operations Manager shall ensure active status Operability Determinations are presented for PORC review during regularly scheduled PORC meetings per AP-01.01.

8.6 **Records**

8.6.1 Completed Operability Determinations are considered Quality Records.

A. Operability Determinations for CRs are retained.

B. Operability Determinations for PIDs are forwarded to the Operations Department General Clerk for retention as required by AP-02.08.

8.6.2 Active Operability Determinations, including REOs, being tracked as Potential LCOs, are retained with AP-12.08 documentation until completed.

8.6.3 Engineering Confirmations are retained with the completed Operability Determination.

9.0 ATTACHMENTS

1. OPERABILITY DETERMINATION FORM
2. ENGINEERING CONFIRMATION SUMMARY FORM
3. INITIAL ENGINEERING CONFIRMATION GUIDELINES
4. DETAILED ENGINEERING CONFIRMATION GUIDELINES
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10. OPERABILITY DETERMINATION PROCESS FLOWCHART

OPERABILITY DETERMINATION FORM

☐ ACTIVE (PLCO # ____________________)   ☐ COMPLETE

PID# ____________   CR# ____________

Describe potentially degraded or non-conforming equipment/systems:

____

____

____

Describe UFSAR Specified Functions: ______________________________

____

____

Effect of Condition on UFSAR Specified Function		Operable	Inoperable	REO	NA
	Equipment	☐	☐	☐	☐
	Train	☐	☐	☐	☐
	Function	☐	☐	☐	☐

(_) If all are OPERABLE, exit AP-03.11

Basis: ____

____

____

Review: Shift Manager _______

(_) If any are INOPERABLE, enter LCO. LCO No. ____________

Basis: ____

____

____

(Operability of redundant, supported, or supporting items considered.)

Review: Shift Manager _______

Continued on next page

THIS IS A QUALITY RECORD

(_) If any have a Reasonable Expectation of Operability (REO), request Engineering Confirmation

Basis: ____

____

____

____

Engineering Confirmation: Requested: ____________
Date/Time

Interim
Review: Shift Manager: ____________

Engineering Confirmation: Completed: ____________
Date/Time

(_) Operable with no Compensatory Actions, exit AP-03.11

(_) Inoperable, enter LCO per ODSO-34 LCO No. ____________

(_) Operable with the following Compensatory Actions, enter a Potential LCO per ODSO-34 and complete required 10CFR50.59 reviews:

(_) Actions required following plant shutdown.

(_) Actions require monitoring plant conditions.

(_) Compensatory actions taken to justify continued operable status.

LCO No. ____________ 10CFR50.59 reviews complete: ____________
Date/Time

Active Operability Determination PORC review: PORC Meeting # ____________

Review: Shift Manager ____________

THIS IS A QUALITY RECORD

Operability Determination Guidance

1. Review the following documents to determine the UFSAR Specified Functions performed by SSCs identified as potentially degraded or non-conforming:

   - JAF Technical Specifications
   - **[CTS]**AP-01.04 , Technical Specification Related Requirements, Lists, and Tables
     [ITS]Technical Requirements Manual
   - MCM-6A Reference Document
   - JAF Updated Final Safety Analysis Report
   - JAF Design Basis Documents

2. Consider the following questions when performing Operability Determinations:

   - Will the SSC(s) be prevented from performing the design function(s)?
   - Could the problem affect the operability of a Technical Specification required SSC?
   - Could the capability of an SSC to prevent or mitigate consequences of an accident as postulated or described in the UFSAR be reduced?
   - Could the condition result in an SSC not meeting known design requirements contained in design documents?
   - Does the problem involve an INOPERABLE non-Technical Specification SSC that could functionally affect a Technical Specification SSC's ability to perform its design function?
   - Could the problem have adverse safety significance requiring prompt review or correction?
   - Could single failure design criteria have been defeated?

PID#______________ CR#______________ Potential LCO# ______________

- Initial Confirmation (No further confirmation required)
- Initial Confirmation (Detailed Confirmation required)
- Detailed Confirmation

Briefly describe condition:

Discrepancy is against: *(Reference Document Number and Section)*

- Calculation ______________
- Test Results ______________
- Commitment # ______________
- SSCs ______________
- Other (NSE, Procedure, etc) ______________
- Design Basis Document __
- Drawing or Spec ____
- UFSAR or Tech Specs____
- **[CTS]**AP-01.04 **[ITS]**TRM

Are there other affected SSCs? • **NO** • **YES**, Describe:
(Other than identified on the REO)

- Evaluation **SUPPORTS** Operability
- Evaluation **DOES NOT SUPPORT** Operability *(Notify SM)*

Discuss logical and defensible basis for conclusion:

List supporting references: *(Analysis, drawings, NSE, Specs, etc.)* • Attached

	Print/Sign	Date
Preparer		/
Independent Reviewer		/
Engineering Supervisor		/
Engineering Manager (or designee), if applicable		/
GMPO (or designee), if applicable		/

- SEND COMPLETED FORM TO THE SHIFT MANAGER -

THIS IS A QUALITY RECORD

INITIAL ENGINEERING CONFIRMATION GUIDELINES

PART 1 - GENERAL GUIDELINES

The Initial Engineering Confirmation is an evaluation to determine if the equipment in question is capable of performing its specified design functions. A recommendation of Operability should be provided.

1. The evaluation is directed toward gathering information to confirm the operability of the equipment based on analysis, test or partial test, operating experience and technical judgment. The evaluation should conclude that reasonable assurance (refer to Attachment 5) does or does not exist that the equipment will perform its design function until corrective action and/or further investigation can be completed.

2. The magnitude of nonconforming/degraded condition should be noted for consideration. If technical judgment determines that the nonconforming/degraded condition in question has no impact on the design function, the equipment should remain operable.

3. A visual examination of the nonconforming/degraded equipment should be made. Any notable comparisons with similar conforming/qualified equipment should be made.

4. If a clearly physical problem is the basis for the nonconforming/ degraded condition, it should be so indicated. Any immediate corrective actions, such as temporary braces and/or other alternatives or "fixes" that can be quickly used to provide reasonable assurance that the equipment will function until corrective action can be completed should be indicated.

5. The evaluation should give some indication of the safety significance of the nonconforming/degraded equipment. The evaluation should conclude if the equipment is or is not required to perform the design function. This includes any support function to any equipment required by Technical Specifications.

6. The evaluation should be independently reviewed. The independent reviewer should have had minimal involvement in the evaluation preparation. The independent reviewer's signature signifies concurrence with the evaluation and that the scope, logic, and supporting technical analysis are adequate. Obtain and incorporate the Shift Manager's input prior to completing the independent review.

7. If the Operability is based on the use or availability of other equipment (e.g., LER-00-016-01), it must be verified that the equipment is capable of performing the function utilized in the evaluation (i.e., functional testing completed, visual inspection, etc.), if plant conditions allow. This would include any equipment used for contingency and/or having administrative controls placed on them.

INITIAL ENGINEERING CONFIRMATION GUIDELINES

PART 2 - FORMAT

Use the following format for documentation of the Initial Engineering Confirmation.

1. **Summary Statements**

   Succinctly state the nonconforming/degraded condition in clear, concise terminology. Summarize the results of the evaluation, succinctly stating the operability recommendation. Underline the recommendation statement.

2. **References**

   List all procedures, specifications, standards, codes, calculations, drawings, regulatory documents, etc., including revision numbers that were used in the evaluation.

3. **Detailed Problem Statements**

   Clearly identify and discuss each item of nonconforming/degraded condition.

   - Describe the design function performed by the equipment.
   - Describe any background of events leading to the nonconforming/degraded condition, include times, dates, documents, personnel, etc. involved with related circumstances.
   - Describe by what means and when the potential nonconforming/degraded condition was discovered.
   - Describe the failure mechanism.
   - If appropriate, provide a subject background summary of why the equipment/component was designed for the original application/function, i.e., summary of pertinent design basis, including any abnormality/deviation allowances of which the evaluator may be aware.

INITIAL ENGINEERING CONFIRMATION GUIDELINES

4. **Assumptions**

Specifically state all assumptions made in the engineering evaluation.

5. **Engineering Evaluation**

Provide an evaluation for each item in the detailed problem statements.

The evaluation summary should clearly indicate if the component can perform its design function and the basis thereof.

Describe the applicable commitments and design requirements, i.e., 10CFR, IEEE, ANSI, ASME, etc., and why they may or may not be met.

If walkdowns or inspections were conducted, details should be provided here or referenced in the attachment section, including names, dates, criteria and specific results.

Describe the basis for recommending the systems operable (i.e., analysis, test or partial test, operator experience or technical judgment).

If it is determined that the nonconforming/degraded condition is OPERABLE but outside of the existing licensing basis, corrective action must either restore the nonconforming/degraded condition to the existing licensing basis, or the licensing basis must be revised to envelope the evaluated condition.

6. **Impact on Nuclear Safety**

Provide a description of the impact of the nonconforming/degraded condition on nuclear safety, include an evaluation of what other equipment could be affected by a failure of the equipment determined to be inoperable. Specifically:

- Assess the potential impact on accident response.
- Include the effects of any short-term (immediate) actions in the impact assessment.
- For inoperable equipment, include an evaluation of the failure effects.
- Address immediate effects from the existing condition as well as possible effects from related failure.
- Include the likelihood of failure.

INITIAL ENGINEERING CONFIRMATION GUIDELINES

7. **Immediate Actions**

Describe/recommend any immediate actions or alternatives, which can be taken or are needed to quickly provide reasonable assurance that the equipment in question will function until long-term corrective action can be completed (e.g., a temporary seismic support, temporary use of an installed spare, etc.).

If any restrictions or limitations (such as temperature, pressure, etc.) are placed on the OPERABILITY verification, these must be clearly stated and identified for operating the plant. Provide an estimated completion date for these actions if possible.

8. **Long Term Actions**

In some cases it may be possible to identify the appropriate long-term corrective action. If so, describe this and provide the status or schedule if available. As with all 10CFR50 Appendix B conditions adverse to quality, the schedule for corrective actions should be commensurate with importance to safety of the nonconforming/degraded condition. Also, identify if any further detailed engineering evaluation is required. Describe the aspects that need further investigation. If possible, provide an estimated completion date.

If Long Term Corrective Action was previously planned for other reason(s), then revise action (WR, CA, etc.) to reference this CR and/or Engineering Confirmation. Such revision provides linkage to prevent cancellation or deferral without proper review.

9. **Signatures**

Provide the signatures of all preparers, independent reviewers, an Engineering Supervisor, System Engineering Manager, Design Engineering Manager (as appropriate), PORC meeting number (if necessary), and SM (concurrence). The SM shall indicate the date and time of his signature as the official time of the Operability Confirmation.

10. **Attachments**

Provide any attachments necessary to substantiate the evaluation.

NOTE: If time does not allow for the validation process to be completed prior to issuance of the evaluation, then the following statement (or equivalent) should be included in the evaluation:

"Our internal verification process is not yet complete for this response. The verification process will be completed as part of the Detailed Engineering Confirmation."

INITIAL ENGINEERING CONFIRMATION GUIDELINES

GUIDELINES FOR CONFIRMATION OF OPERABILITY

NOTE: The measure of "reasonable" should be commensurate with importance to safety of the nonconforming/degraded condition and the magnitude of the nonconforming/degraded uncertainty problem (refer to Attachment 5).

1. When reasonable technical judgment indicates that the nonconforming/ degraded condition is capable of performing its intended design function when required, the equipment should be declared operable.

    a. If there is reasonable assurance that the equipment is capable of performing its specified design function, and that the confirmation process will support this expectation, but there are some remaining concerns or uncertainties, the equipment can remain operable until further evaluation can resolve the concerns.

    b. If the initial engineering evaluation indicates that it can be shown that the nonconforming/degraded condition in question is irrelevant to the design function of the equipment, the equipment should remain operable.

2. When reasonable technical judgment indicates that the nonconforming/ degraded equipment is not capable of performing its specified design function when required, the equipment should be declared inoperable.

    a. For inoperable equipment in a system not covered by the Technical Specifications, reactor operation may continue if the design function can be accomplished by other designated equipment that is qualified, or if limited administrative controls can be used to ensure the design function is met.

    b. For a system covered by Technical Specifications that is capable of performing its specified function with an inoperable support system that is not covered by Technical Specifications, no additional action outside of restoring the inoperable support system is needed.

DETAILED ENGINEERING CONFIRMATION GUIDELINES

NOTE: In most cases, especially when the equipment is quickly repaired or replaced, a Detailed Engineering Evaluation may not be needed.

1. The Detailed Engineering Confirmation should be a more rigorous analysis beyond the initial "engineering judgment" evaluation and typically includes more rigorous analysis if this can be done within the time allotted.

2. The evaluation may consider available test data to determine whether the test conditions envelope equipment design conditions. The design conditions should ensure that the equipment will perform its design functions when called upon to mitigate the accidents for which it is needed.

3. The evaluation may consider a materials assessment to determine the material susceptibility to aging, peak temperature and radiation.

4. The evaluation may consider similarity analysis to determine that the differences between the nonconforming/degraded equipment and a conforming/qualified one would not impair the equipment design function performance.

5. The evaluation may consider extrapolation of available analyses to assess if the design condition would be met for the nonconforming equipment.

6. The evaluation should give some indication of the safety significance of the nonconforming/degraded equipment and should specify what would be a reasonable time for corrective actions before operational alternatives are taken.

REO/REASONABLE ASSURANCE GUIDELINES

"Reasonable assurance" is a level of confidence that a particular situation or condition exists or does not exist. In making a finding of reasonable assurance, the existing facts are first gathered. Engineering judgment is then applied to those facts resulting in a "weight of evidence" supporting a finding of reasonable assurance. Management involvement is an essential element in the process particularly when uncertainty exists or delays in the process are excessive or unacceptable.

The concept of reasonable assurance can be applied at several stages of the deficiency evaluation process, namely in:

1. The initial categorization of a deficiency
2. The Operability Determination for safety related equipment
3. The development of a Justification for Continued Operation (JCO) if the affected equipment is determined to be inoperable.

Engineering judgment should be applied when it is technically appropriate and defendable to reach conclusions by this method instead of performing more rigorous analyses. The reluctance to use engineering judgment when assigning a level of confidence can contribute to delays in the deficiency evaluation process. Management attention should be applied to expedite the process and focus resources.

Since the facts in existence may change as more information is obtained, the weight of evidence is dynamic. As the weight of evidence changes, the overall conclusion may require update. For this reason, the use of engineering judgment may require follow-up analyses, tests or inspections to confirm the validity of the conclusions reached.

Three different conclusions can be reached when making a finding of reasonable assurance:

1. reasonable assurance that a condition exists,

2. reasonable assurance that a condition does not exist, or

3. uncertainty as to whether a condition does or does not exist.

If uncertainty exists, management involvement is necessary. Management may decide to add to the existing facts (by directing that additional tests or analyses be conducted) or reevaluate the facts using engineering judgment in order to change the weight of evidence. The weight of evidence should be changed enough to reach a finding of reasonable assurance (either positive or negative).

REO/ENGINEERING OPERABILITY GUIDELINES

NOTE: This information is provided as an aid. The SM maintains the responsibility to determine whether an SSC is operable or inoperable.

GENERAL AND MISCELLANEOUS OPERABILITY ISSUES

Operability and Time of Entry into Technical Specification [CTS]AP-01.04 [ITS]Technical Requirements Manual Action Statements

- The start of a Technical Specification **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual Action Statement begins at the time the information was originally received or the event occurred.

- Equipment that has been discovered to be inoperable from some previous time (known or unknown) shall be considered inoperable from the time the information was originally received for the purpose of entry into a Technical Specification Action Statement (i.e., Technical Specification Action Statements are not imposed retroactively.).

Items Clearly Inoperable

- Certain conditions clearly render equipment inoperable. In these instances, the time of declaring equipment inoperable is the time of discovery.

  A. If equipment is unable to perform its function due to obvious failure, damage, or malfunction or due to being removed from service (tagged out), then it is inoperable.

  B. If equipment fails to start upon receipt of a valid safety signal, then it is inoperable.

  C. If equipment fails to meet the quantitative requirements of Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual or of surveillances demonstrating compliance with Technical Specifications, then it is inoperable.

  Examples of this are Technical Specification or surveillance required tank levels, system pressures, valve stroke times, system flow rates, etc.

- Equipment exposed to operating conditions in excess of its design rating

  A. If equipment is exposed to operating conditions in excess of its design rating, then it is inoperable until engineering evaluation determines it to be operable.

REO/ENGINEERING OPERABILITY GUIDELINES

Missed or Deficient Surveillance

[CTS]

- Technical Specification related equipment or systems are declared inoperable upon discovery of a missed or deficient surveillance test. At the time of discovery of the missed or deficient test, the action statement of the appropriate LCO is applicable; however, if actions are required to be performed within 24 hours or less, the 24 hour allowance provided by Technical Specification 4.0.3 may be entered. If reasonable expectation of operability does not exist, immediately declare the equipment or system inoperable. If, during the testing, the surveillance results in inoperability, then the appropriate LCO action statement requirements must be applied.

[ITS]

- If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed from the time of discovery up to 24 hours, or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any surveillance delayed greater than 24 hours and the risk impact shall be managed.

[ITS]
If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

[ITS]
When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

- If the missed or deficient surveillance is not required by Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual or ASME Section XI, then operability is unaffected.

- If this is a failure to retest or perform functional verification of equipment prior to restart or return to service, then the equipment is inoperable.
- If the retest or functional verification is required by Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual or ASME Section XI, then the equipment is inoperable.

Incorrect Inputs Used in a Calculation

If incorrect inputs (e.g. use of the wrong response spectrum, improper cable resistance, incorrect material properties, etc.) were used in a calculation and they are determined to be non-conservative in nature, then an engineering evaluation is required to assess operability.

REO/ENGINEERING OPERABILITY GUIDELINES

Documentation Only Deficiencies

If document deviations are identified that do not constitute a non-conforming condition as described in definition 5.10, then the related SSCs are operable.

For example, if an EQ file is identified with an incorrect radiation margin, but justification is available that envelops the plant design requirements, there is no non-conforming condition since design and licensing requirements are met.

MECHANICAL OPERABILITY ISSUES

Minor process fluid leakage (packing glands, gaskets, and non-welded connections)

If the leakage clearly has no adverse impact on any SSC function, then the equipment is operable unless the leakage constitutes reactor coolant system or Containment boundary leakage. Then an engineering evaluation is required. Examples of possible adverse effects requiring engineering evaluation are: spraying water does not reach electrical equipment, contaminate oil reservoirs or result in significant spread of contamination

Oil leaks

- If the leakage is from non safety-related pumps or motors (i.e. equipment without an assumed long-term operating requirement in the UFSAR) and does not require reservoir replacement more than once per shift, then the equipment is operable.

- If level is found to be outside the optimum range, but remains visible in the sight glass, then the equipment is operable unless the frequency of oil replacement exceeds once per shift or increases significantly.

- If the leakage is from safety-related or Cat M pumps (i.e. equipment assumed to require extended operation per the UFSAR) or requires reservoir replacement more than once per shift, then an Engineering Evaluation is required.

Materials (piping, fittings, bolts, nuts, etc.)

- If the material for piping, fittings, bolts and other components is discovered to be different than the design documents specify, then the equipment is operable pending Engineering Evaluation of chemical compatibility and structural strength requirements.

REO/ENGINEERING OPERABILITY GUIDELINES

- If painting deficiencies (missing, bubbled or chipped) are found on external surfaces outside of Primary Containment that do not pose a Foreign Material Exclusion (FME) concern, then the equipment is operable.

Manual valve position

- If a manual valve that is required by Technical Specifications to be locked in a particular position is found in the correct position, but not locked, then the valve is inoperable until locked or until equivalent compensatory measures are taken.

- If a manual valve that is required by a document other than Technical Specifications (**[CTS]**AP-01.04 **[ITS]**TRM, UFSAR, etc.) to be locked in a particular position is found in the correct position, but not locked, then the valve is operable.

- If a manual valve that is required to be in a particular position is found in an incorrect position, then the valve and its associated system are inoperable until restored to its required position.

Power operated valves

- If a motor operated valve is required by Technical Specifications or other licensing basis document (e.g. Appendix R analysis) to be in a specific position with its associated supply breaker open, and the breaker is not open, then the valve and its associated system are inoperable until the breaker is opened.

- If a Technical Specification required "actuated-closed" power operated valve is backseated, then the valve is inoperable until either:

  A. an engineering evaluation has been performed to demonstrate that the backseating is acceptable, or

  B. the valve has been demonstrated to be capable of closing from its backseated position.

- If a Technical Specification required "actuated-open" power operated valve is torque seated, then the valve is inoperable until either:

  A. an engineering evaluation has been performed to demonstrate that the torque seating is acceptable, or

  B. the valve has been demonstrated to be capable of opening from its torque seated position.

REO/ENGINEERING OPERABILITY GUIDELINES

ASME Section XI qualification

- If ASME Section XI equipment does not meet the overall requirements of the applicable ASME Section XI specifications, then the equipment is inoperable.
- If a valve exceeds the IST stroke time ACTION limit, then the equipment is inoperable.
- If a pump does not meet the IST ACTION criteria, then the equipment is inoperable.

ELECTRICAL/I&C OPERABILITY ISSUES

Setpoint and calibration tolerance

- If an equipment setpoint or calibration is determined to exceed that required by Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual, then the equipment is inoperable.
- If an equipment loop is determined (by test and/or calculation) to be unable to perform its intended function within its required Technical Specification limits, then the loop is inoperable.

Equipment with automatic and manual start/stop capability

- If, for such equipment, the manual start/stop capability is required (by Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual, UFSAR, EOPs, etc.) to fulfill a UFSAR Specified Function and it is lost, then the equipment is inoperable.
- If, for such equipment, the automatic start/stop capability is required (by Technical Specifications **[CTS]**AP-01.04 **[ITS]**Technical Requirements Manual, UFSAR, EOPs, etc.) to fulfill a UFSAR Specified Function and it is lost, then the equipment is inoperable.

REO/ENGINEERING OPERABILITY GUIDELINES

Environmental Qualification (EQ)

- If equipment is installed and maintained in accordance with the JAF Environmental Qualification Program, then, from an EQ standpoint, the equipment is operable (i.e., it is environmentally qualified or has "environmental qualification").

- If, on equipment that is required to be environmentally qualified, a condition exists that obviously would not allow performance of a UFSAR Specified Function under all postulated service conditions, then the equipment is inoperable.

  For example, the EQ Maintenance & Installation Requirement for an instrument transmitter may require it to be sealed against moisture/steam intrusion. If the transmitter does not have a seal installed, it is inoperable because it is obvious that it would not meet the EQ Maintenance & Installation Requirement.

- If, on equipment that is required to be environmentally qualified, a condition exists that may compromise its environmental qualification, but it is not obvious whether its UFSAR Specified Function would be performed under all postulated service conditions, then the condition may require Operability Confirmation.

  For example, the EQ Maintenance & Installation Requirement for an instrument transmitter may require it to be sealed against moisture/steam intrusion. If the transmitter has an unused conduit connection sealed only with a plastic shipping plug, then the transmitter may be operable. This may be either because other testing has been performed for this configuration or the EQ documentation may not have differentiated between LOCA and HELB mitigation, which have different qualification requirements.

  For another example, a procedural EQ Maintenance & Installation Requirement may require replacement of an instrument transmitter's O-rings at five-year intervals. If it is determined that a transmitter has exceeded this five year O-ring replacement interval, it is not obvious that performance of its UFSAR Specified Function is prevented. An evaluation using transmitter/O-ring test data, engineering analysis, etc., is required to confirm its operability.

REO/ENGINEERING OPERABILITY GUIDELINES

Electrical Breakers

- If an electrical breaker trips after having been reset from an earlier tripped condition, then it is inoperable.
- Operability may be re-established once the cause of the breaker trip is corrected and component operation is retested.

Emergency Diesel Generators

- If an Emergency Diesel Generator fails to start or load, then it is inoperable.
- If an EDG trips on a non-safety trip signal that would be bypassed on receipt of an automatic start signal, then the cause of the trip requires an Engineering Evaluation to determine whether the EDG would be capable of sustaining its function after an automatic start.

EPIC Safety Parameter Display System (SPDS)

- If unable to successfully restart either CPU after failure of both for greater than one hour, then the SPDS is inoperable.
- If unable to transmit data to either the Technical Support Center (TSC) or the Emergency Operations Facility (EOF) for a period greater than one hour, then the SPDS is inoperable.

STRUCTURAL/CIVIL OPERABILITY ISSUES

Barriers

- If an inadequate fire barrier exists, then it is inoperable.
- If a Condition exists affecting the structural integrity of a room, building, foundation, or other structural component, then an evaluation may need to be performed to determine its operability.
- If a HELB barrier is breached, then an engineering evaluation is required to confirm operability of affected SSCs.

REO/ENGINEERING OPERABILITY GUIDELINES

Missing or loose nuts/fasteners

If the fastener is part of one of the following joints, then the SSC is operable (provided an FME concern is not created):

- A joint supporting a non-EQ cover plate for an instrument, switch, cable tray cover or other similar application, provided there are no more than two missing or loose fasteners.
- A nut/bolt holding a valve handwheel, provided the stem is directed vertically upwards.
- Cabinet door hinges or latches, provided the door can be opened, closed and latched.
- Valve packing gland assembly, provided there is no leak at the valve stem with the system pressurized, IST stroke times are within limits and the valve is not a Containment Isolation Valve.
- A nut/bolt holding a screen on air cooling ports of equipment, provided there are sufficient bolts in place to prevent movement of the screen.

Pipe & Tubing Supports

- If the component is obviously damaged, then an engineering evaluation is required to confirm support/system operability.
- If the component is incapable of performing its function, then the component is inoperable. An engineering evaluation is required to determine system operability since such a condition may or may not render the system inoperable.
- If the component is performing a function it is not designed for (e.g. pipe is binding due to inadequate gaps or supporting unauthorized/unanalyzed equipment), then an engineering evaluation is required to confirm operability.

REO/ENGINEERING OPERABILITY GUIDELINES

Hydraulic and mechanical snubbers

- If a hydraulic or mechanical snubber is found to be outside the range specified on the drawing or beyond the snubbers setting, including tolerances, then an engineering evaluation is required.
- If a hydraulic snubber has "the remotest amount of fluid available in the reserve reservoir", there is enough fluid to accommodate full hydraulic response for any stroke position and it is operable.

EROSION/CORROSION

If pipe wall thickness is measured to be <87.5% of the nominal wall thickness, then an engineering evaluation is required to determine if the pipe is operable. Procedure CES-7, Procedure for Structural Evaluation of Erosion/Corrosion Thinning in Carbon Steel Piping," provides guidance for this evaluation.

CHEMISTRY SAMPLING OPERABILITY ISSUES

If a chemistry sample is outside the Technical Specification limits AND a second confirmed sample exists, OR the responsible individual concludes that the results are valid based on trend of previous analyses, then it is inoperable.

USE OF ENGINEERING JUDGMENT FOR OPERABILITY ASSESSMENTS

According to the definition of operability, an SSC must be capable of performing its specified function(s), as described in the Technical Specification Bases or the UFSAR.

If a system, structure, or component is covered by the Technical Specifications, then the Technical Specifications must be followed. If an SSC is not covered by Technical Specifications, then the additional guidance of Generic Letter 91-18, can be applied.

REO/ENGINEERING OPERABILITY GUIDELINES

USE OF ENGINEERING JUDGMENT FOR OPERABILITY ASSESSMENTS (cont)

Generic Letter 91-18, and other NRC guidance acknowledge the acceptability of using engineering judgment to justify component or system operability. The two major points that most of the guidance stresses are:

- Engineering judgment should only be used to justify component or system operability when there is reasonable expectation that a detailed analysis or evaluation will prove operability of the component or system.
- A sound basis for the engineering judgment conclusion must be documented.

The scope of the engineering judgment Operability Determination must be sufficient to address the capability of the equipment to perform its UFSAR Specified Functions. The determination should consider the following:

- Determine what equipment is degraded.
- Determine the UFSAR Specified Functions of the affected equipment.
- Determine the extent of the degradation, including the possible failure mechanism.
- Determine if the equipment is capable of performing its UFSAR Specified Function.
- Determine the basis for declaring the system operable.

NOTE: If the component or system can not perform at the level credited in the accident analysis, then it should be considered inoperable unless supported by additional analysis. If the component or system can not perform at the level required by Technical Specifications **[CTS]** AP-01.04 **[ITS]** Technical Requirements Manual, then it should be considered inoperable.

The Operability Determination should discuss the considerations listed above and must document the basis for declaring the system operable.

To continue operation while a formal Operability Determination is being made, there must be a reasonable expectation that the affected safety system is operable and that the confirmation process will support that expectation. If that expectation does not exist or mounting evidence suggests that the final analysis will conclude that the equipment can not perform its UFSAR Specified Functions, then the system should be considered inoperable and appropriate actions must be taken.

REO/ENGINEERING OPERABILITY GUIDELINES

CONTINGENT OPERATOR ACTION(s) GUIDANCE

The following criteria should be considered when operator action(s) are credited in the Operability Assessment:

1. Sufficient number of shift operators are available to perform the required actions.

2. Written procedures, which outline the required actions are clear, complete, unambiguous, available and used.

3. Operators performing the required action are properly trained.

4. Sufficient time is available for the operator to perform the required actions.

5. Locations outside of the control room at which operator actions must be performed shall be qualified to adequately protect the operator from the environmental conditions caused by the design basis event.

6. The dose to an individual operator who is required to take actions shall not exceed 5 Rem TEDE - Limit (Internal Dose & External Dose) whole body, for the duration of the event.

References: Information Notice 97-78 and ANSI 58.8

IMMEDIATE REPORTABILITY CHECKLIST

This checklist should be used for determining if an event is immediately reportable, or reportable within 1 hour, 4 hours, 8 hours, or 24 hours. Refer to NUREG-1022, "Event Reporting Guidelines 10CFR50.72 and 10CFR50.73" (Rev. 2, October 2000) for additional guidance on reportability determinations.

I. IMMEDIATE REPORTABILITY

A. **10 CFR 20.1906(d)** - Immediate notification to the NRC and the final delivery carrier (not later than 1 hour)

Does the event involve the receipt of a shipping package found to be in non-compliance with external contamination or dose rate limitations? YES/NO

B. **10 CFR 20.2201(a)(1)(i)** - Immediate notification not later than 1 hour.

Does the event involve loss or theft of licensed material? YES/NO

C. **10 CFR 20.2202(a)** - Immediate notification (not later than 1 hour)

Does the event involve byproduct, source or special nuclear material possessed by Entergy that may have caused or threatens to cause:

1. An individual to receive a total effective dose equivalent of 25 rems or more, an eye dose equivalent of 75 rems or more, or a shallow - dose equivalent to the skin or extremities of 250 rads. YES/NO

2. The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake (does not apply to location where personnel are not normally stationed during routine operations). YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

I. IMMEDIATE REPORTABILITY (cont)

D. **10 CFR 50.72(a)(1):** Immediate Notification (ENS phone)

(i) Did the event result in the declaration of any of the Emergency Plan classes? (See 10 CFR 72.75(a) for ISFSI events.) YES/NO

E. **10 CFR 72.75(a):** Immediate Notification (ENS phone), followed by a written report within 30 days

Did the event involve ISFSI and result in the declaration of any Emergency Plan classes? YES/NO

ATTACHMENT 7 Page 3 of 12

IMMEDIATE REPORTABILITY CHECKLIST

II. 1 HOUR REPORTABILITY

A. **10CFR50.72(b)(1):** 1 Hour Notification

NOTE: This question may include situations that have occurred in the past 3 years, may continue to exist, or have been recently discovered.

Did the event result in any deviation from Tech Specs authorized pursuant to 10 CFR 50.54(x) (if not reported as declaration of an Emergency Class under 50.72(a)(1) above) YES/NO

If the answer is YES to the above question, notify the NRC Operations Center within 1 hour.

B. **10CFR70.52:** 1 Hour Notification

(a) Does the event involve accidental criticality or any loss, other than normal operating loss, of special nuclear material? (See also 10CFR72.74(a) and 10CFR73 Appendix G (I)(a)(1).) YES/NO

(b) Does the event involve any loss or theft or unlawful diversion of special nuclear material or any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of such material? YES/NO

C. **10CFR72.74(a):** 1 Hour Notification

Did the event involve ISFSI accidental criticality or loss of special nuclear material. (See also 10CFR70.52(a) and 10CFR73 Appendix G (I)(a)(1).) YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

II. 1 HOUR REPORTABILITY (cont)

D. **10CFR73 Appendix G (I)** - 1 hour notification followed by a written report within 30 days.

Does the event require, cause, or result in the following?

(a) Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:

(1) A theft or unlawful diversion of special nuclear material. (See also 10CFR70.52 and 10CFR72.74(a).); YES/NO

OR

(2) Significant physical damage to a power reactor or any facility possessing SSNM or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to a nuclear fuel or spent nuclear fuel a facility or carrier possesses; YES/NO

OR

(3) Interruption of normal operation of a licensed nuclear power reactor through the unauthorized use of or tampering with its machinery, components, or controls including the security system. YES/NO

(b) An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport. YES/NO

(c) Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport for which compensatory measures have not been employed. YES/NO

(d) The actual or attempted introduction of contraband into a protected area, material access area, vital area, or transport. YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

III. 4 HOUR REPORTABILITY

A. **10CFR50.72 (b)(2)** - 4 hour Notification

NOTE: These questions may include situations that have occurred in the past 3 years, may continue to exist, or have been recently discovered.

Does the event require, cause, or result in the following?

(i) The initiation of a shutdown required by Tech Specs. YES/NO

(ii) Reserved

(iii) Reserved

(iv)(A) Any event that results or should have resulted in Emergency Core Cooling System (ECCS) discharge into the Reactor Coolant System as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. YES/NO

(iv)(B) Any event or condition that results in the actuation of the RPS when the reactor is critical, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. YES/NO

(v) Reserved
(vi) Reserved
(vii) Reserved
(viii) Reserved
(ix) Reserved
(x) Reserved

(xi) Any event or situation, related to the health and safety of the general public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials. YES/NO

If the answer is YES to any of the above statements, notify the NRC Operations Center within 4 hours.

IMMEDIATE REPORTABILITY CHECKLIST

III. 4 HOUR REPORTABILITY (cont)

B. **10CFR72.75(b)** - 4 hour Notification followed by a written report within 30 days.

Does the event involve ISFSI and require, cause, or result in the following:

(1) An event that prevents immediate actions necessary to avoid exposures to radiation or radioactive materials that could exceed regulatory limits, or releases of radioactive materials that could exceed regulatory limits (e.g., events such as fires, explosions, and toxic gas releases). YES/NO

(2) A defect in any spent fuel storage structure, system, or component which is important to safety. YES/NO

(3) A significant reduction in the effectiveness of any spent fuel storage confinement system during use. YES/NO

(4) An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under 10CFR72 when the action is immediately needed to protect the public health and safety and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent. YES/NO

(5) An event that requires unplanned medical treatment at an offsite medical facility of an individual with radioactive contamination on the individual's clothing or body which could cause further radioactive contamination. (See also 10CFR50.72(b)(3)(xii) eight hour notification.) YES/NO

(6) An unplanned fire or explosion damaging any spent fuel or high level waste, or any device, container, or equipment containing spent fuel or high level waste when the damage affects the integrity of the material or its container. YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

IV. 8 HOUR REPORTABILITY

A. 10CFR50.72(b)(3) - 8 hour notification

NOTE: These questions may include situations that have occurred in the past 3 years, may continue to exist, or have been recently discovered.

Does the event require, cause or result in the following?

(i) Reserved

(ii)(A) Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; YES/NO

(ii)(B) Any event or condition that results in the nuclear power plant being in an un-analyzed condition that significantly degrades plant safety. YES/NO

(iii) Reserved

(iv)(A) Any event or condition that results in valid actuation of any of the systems listed in paragraph iv)(B) below, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. YES/NO

(iv)(B) **NOTE 1**: The systems to which the requirements of paragraph (iv)(A) above apply are:

(1) Reactor protection (RPS) including: reactor scram and reactor trip. (See also (b)(2)(iv)(B) on page 5.)

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) (Not applicable to JAF)

(4) ECCS including: low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) Reactor core isolation cooling system

(6) (Not applicable to JAF)

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency AC electrical power systems, including emergency diesel generators (EDGs).

IMMEDIATE REPORTABILITY CHECKLIST

IV. 8 HOUR REPORTABILITY (cont)

A. **10CFR50.72(b)(3)** - 8 hour notification (cont)

(v) Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shutdown the reactor and maintain it in a shutdown condition. YES/NO

(B) Remove residual heat. YES/NO

(C) Control the release of radioactive material. YES/NO

(D) Mitigate the consequences of an accident. YES/NO

(vi) **NOTE 2:** Events covered in paragraph (v) above may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to paragraph (v) above if redundant equipment in the same system was operable and available to perform the required safety function.

(vii) Reserved
(viii) Reserved
(ix) Reserved
(x) Reserved
(xi) Reserved

(xii) Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment. (See also 10CFR72.75(b)(5) four hour notification (ISFSI requirement).) YES/NO

(xiii) Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability (e.g., significant portion of Control Room indication, ENS or Off-Site Notification System). YES/NO

If the answer is YES to any of the above statements, notify the NRC Operations Center within 8 hours.

IMMEDIATE REPORTABILITY CHECKLIST

V. 24 HOUR REPORTABILITY

The following checklist should be used for determining if an event is reportable or recordable within 24 hours:

A. **10CFR20.2202(b)** - 24 hour notification

Does the event involve loss of control of licensed material possessed by Entergy that may have caused or threatens to cause:

(1) An individual to receive, in a period of 24 hours (i), a total effective dose equivalent exceeding 5 rems (ii), an eye dose equivalent exceeding 15 rems, or (iii) a shallow-dose equivalent to the skin or extremities exceeding 50 rems. YES/NO

(2) The release of radioactive material, inside or outside of a restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake in excess of one occupational annual limit on intake (does not apply to locations where personnel are not normally stationed during routine operations). YES/NO

If the answer is YES to any of the above statements, notify the NRC Operations Center within 24 hours.

B. **10CFR26.73 (a)** - 24 Hour Report (Fitness for Duty)

Does the event involve a significant fitness for duty event involving:

(1) The sale, use, or possession of illegal drugs within the protected area. YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

V. 24 HOUR REPORTABILITY (cont)

B. **10CFR26.73 (a)** - 24 Hour Report (cont)

(2) Any acts by any person licensed under 10 CFR Part 55 to operate a power reactor or by any supervisory personnel assigned to perform duties:

(i) Involving the sale, use, or possession of a controlled substance YES/NO

(ii) Resulting in confirmed positive tests on such persons YES/NO

(iii) Involving use of alcohol within the protected area YES/NO

(iv) Resulting in a determination of unfitness for scheduled work due to the consumption of alcohol. YES/NO

If the answer is YES to any of the above statements, notify the NRC Operations Center as required.

C. **10CFR72.75(c)** - 24 Hour Notification followed by a written report within 30 days

Did any of the following events related to ISFSI activities involving spent nuclear fuel or high level waste occur:

(1) Any unplanned contamination event that requires access to the contaminated area by workers or the public to be restricted for more than 24 hours by imposing additional radiological controls or by prohibiting entry into the area. YES/NO

IMMEDIATE REPORTABILITY CHECKLIST

V. 24 HOUR REPORTABILITY (cont)

C. **10CFR72.75(c)** - 24 hour notification followed by a written report within 30 days. (cont)

(2) An event in which safety equipment is disabled or fails to function as designed when:

(i) The equipment is required by regulation, license condition, or certificate of compliance to be available and operable to prevent releases that could exceed regulatory limits, to prevent exposures to radiation or radioactive materials that could exceed regulatory limits, or to mitigate the consequences of an accident, and

(ii) No redundant equipment was available and operable to perform the required safety function.

YES/NO

D. **ISFSI Certificate of Compliance No. 1014 Appendix B, Approved Contents, (Ref. 3.2.16) Section 2.2** - 24 Hour Notification followed by a written report within 30 days

Did the event result in violation of any fuel specification or loading condition during storage cask loading? YES/NO

E. **10CFR73 Appendix G (II)** - recorded within 24 hours and submitted in quarterly log.

Does the event require, cause, or result in the following?

(a) Any failure, degradation, or discovered vulnerability in a safeguards system that could have allowed unauthorized or undetected access to a protected area, material access area, controlled access area, vital area, or transport had compensatory measures not been employed. YES/NO

(b) Any threatened, attempted, or committed act not previously defined in 10 CFR 73, Appendix G with the potential for reducing the effectiveness of the safeguards system below that committed to in a licensed physical security or contingency plan or the actual condition of such reduction in effectiveness. YES/NO

If the answer is YES to any of the above statements, ensure Security/Safety Department "records" the incident.

IMMEDIATE REPORTABILITY CHECKLIST

VI. 30 or 60 DAY REPORTABILITY

Many of the events identified above, except for V.B, require a 30 or 60 day written notification. (See AP-03.04 for reportability requirements.)

ENTERGY LICENSING POSITION

EVALUATION AND RESOLUTION OF DEGRADED AND NONCONFORMING CONDITIONS

PURPOSE

This paper presents Entergy's position on evaluating and resolving degraded and nonconforming conditions, as documented on CRs per the corrective action process and PIDs,) per the Work Request process. Included in this paper are discussions pertaining to:

- Appropriate applicability of OPERABILITY Assessments
- Timeliness for resolving degraded and nonconforming conditions
- Appropriate applicability of 10CFR50.59 evaluations to degraded or nonconforming conditions.

This position paper is not intended to circumvent any actions permitted or required by plant Technical Specifications (TS) or any other regulatory requirement. Attachment 1 provides additional supporting information.

BACKGROUND

On November 7, 1991, the NRC published Generic Letter (GL) 91-18 providing to licensees two new sections to its Part 9900 NRC Inspection Manual. These sections were:

1. Resolution of Degraded and Nonconforming Conditions
2. Operable/Operability: Ensuring the Functional Capability of a System or Component

The intent of the guidance provided in these new sections of Part 9900 was to ensure consistency in OPERABILITY assessments and in resolving degraded and nonconforming conditions.

On October 18, 1997, the NRC published Revision 1 to GL 91-18 to inform licensees of a revised section of Part 9900, Resolution of Degraded and Nonconforming Conditions. The changes to this section more explicitly discuss the role of 10CFR50.59 in resolving degraded and nonconforming conditions.

ENTERGY LICENSING POSITION

ENTERGY POSITION

I. APPLICABILITY OF OPERABILITY ASSESSMENTS

Entergy will make every effort to operate its plants in full compliance with applicable design and licensing requirements and commitments with no degraded or nonconforming conditions. However, when a degraded/ nonconforming condition is identified, an OPERABILITY assessment is performed to determine if the affected SSC is capable of performing its intended safety function (i.e., safety-related as defined in 10CFR50.2). Such activities are governed under the corrective action program required by 10CFR50, Appendix B, Criterion XVI. (NOTE: The level of detail provided in the OPERABILITY assessment is commensurate with the safety function of the degraded/nonconforming SSC and the complexity of the issue.)

The terms "OPERABLE" and "OPERABILITY" are defined in each plant's Technical Specifications. In addition, system OPERABILITY is governed by Technical Specifications. The OPERABILITY assessment focuses on the ability of the subject SSC to meet its specified safety functions, thereby meeting the Technical Specification definition of "OPERABLE".

In addition to clearly evaluating the condition's effect on safety function capability, the OPERABILITY assessment should include reviews to determine any compensatory actions needed to maintain OPERABILITY.

II. TIMELINESS OF RESOLVING DEGRADED/NONCONFORMING CONDITIONS

Degraded/nonconforming conditions will be corrected as soon as practicable **commensurate with safety significance of the condition (e.g., the ability of the SSC to perform its safety function).** Corrective actions should be taken to bring the degraded/nonconforming condition into compliance with the design/licensing basis at the first reasonable opportunity. As plant on-line times improve, this opportunity may be during the upcoming refueling outage. Our goal is to restore significant conditions that impact SSC OPERABILITY no later than startup from the next refueling outage unless extenuating circumstances arise which make such actions impracticable (e.g., plant conditions, parts availability, incomplete design).

ENTERGY LICENSING POSITION

In cases where significant degraded/nonconforming conditions cannot be resolved prior to restart from the designated refueling outage, the Operability assessment must be re-evaluated to ensure extended, safe plant operation with the condition is justified with a revised restoration date or milestone specified (usually the next refueling outage). This situation remains under the control of the Appendix B corrective action program; application of 10CFR50.59 is not appropriate. See further discussion in Section III, below. (Plant management may wish to keep the NRC informed of these conditions and of revised plans to correct the condition. This communication can be as simple as discussing the condition with the site resident inspector and the NRR project manager; no formal approval is required.)

GL 91-18 describes the relationship between OPERABILITY and 10CFR50, Appendix B, Criterion XVI. In brief, licensees must develop and implement a corrective action plan in parallel with the OPERABILITY assessment. The purpose of the OPERABILITY assessment is to determine if there is reasonable assurance that the degraded/nonconforming SSC will perform its safety function. Conditions that have been identified as degraded/nonconforming should be evaluated for aggregate impact on specific safety functions. Therefore, the degraded/nonconforming condition may remain in effect while implementing a corrective action plan intended to restore the condition to "Fully Qualified" status. Actions to resolve a degraded/nonconforming condition must be taken promptly, commensurate with the safety significance of the adverse condition.

III. APPLICABILITY OF 10CFR50.59 EVALUATIONS TO DEGRADED/NONCONFORMING CONDITIONS

10CFR50.59 reviews and OPERABILITY assessments are mutually exclusive aspects of addressing a condition adverse to quality. As discussed in Section I above, the OPERABILITY assessment provides reasonable assurance a degraded/nonconforming SSC is capable of performing its safety function during the time period corrective actions are being developed and implemented. The §50.59 process is only applied to certain compensatory actions that may be taken as part of the OPERABILITY assessment and to any proposed changes implemented to resolve the degraded/ nonconforming condition. These items are discussed below.

ENTERGY LICENSING POSITION

- **Compensatory Actions**

As discussed in Section I above, compensatory actions may be required to maintain system/component OPERABILITY, e.g., changes to procedures, temporary physical plant modifications (temporary alterations), implementing new procedures. Such actions require review for 10CFR50.59 applicability. The scope of the applicability review and any resulting evaluation is limited to the specific compensatory actions and should not include the full scope of the degraded/nonconforming condition.

- **Resolving Degraded/Nonconforming Conditions**

Resolving the degraded/nonconforming condition is typically addressed through one of three ways:

1. Restoration to current design

2. A physical change to the plant

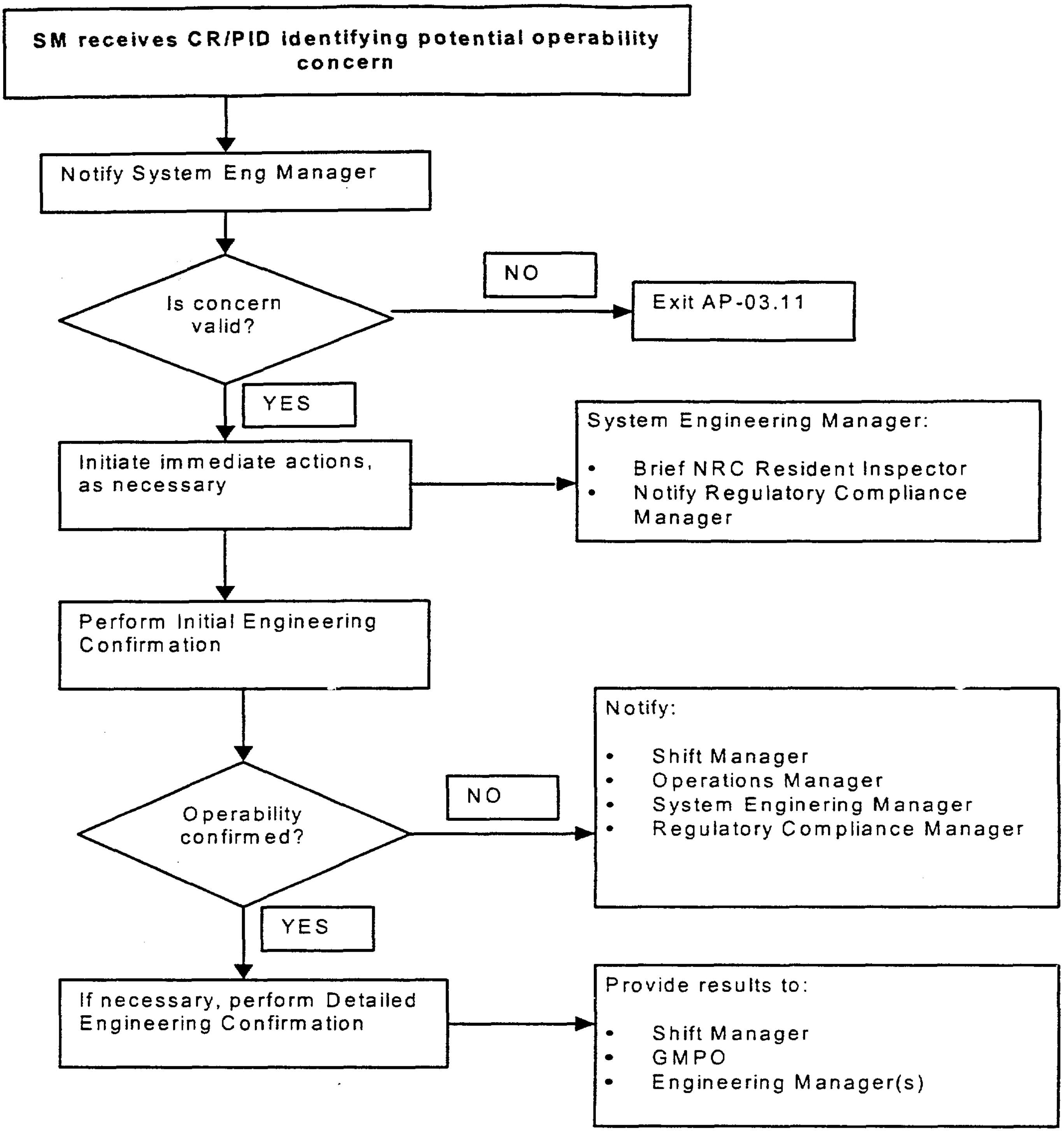
3. A change to design and licensing bases to accept the condition as-is

In cases 2 and 3, the appropriate change process(s) for actions to be taken requires a 10CFR50.59 review. However, as specified in Section I above, the degraded/nonconforming condition itself is governed by the OPERABILITY assessment.

Failure to resolve a degraded/nonconforming condition in a timely manner is an issue with the corrective action program and not with application of 10CFR50.59.

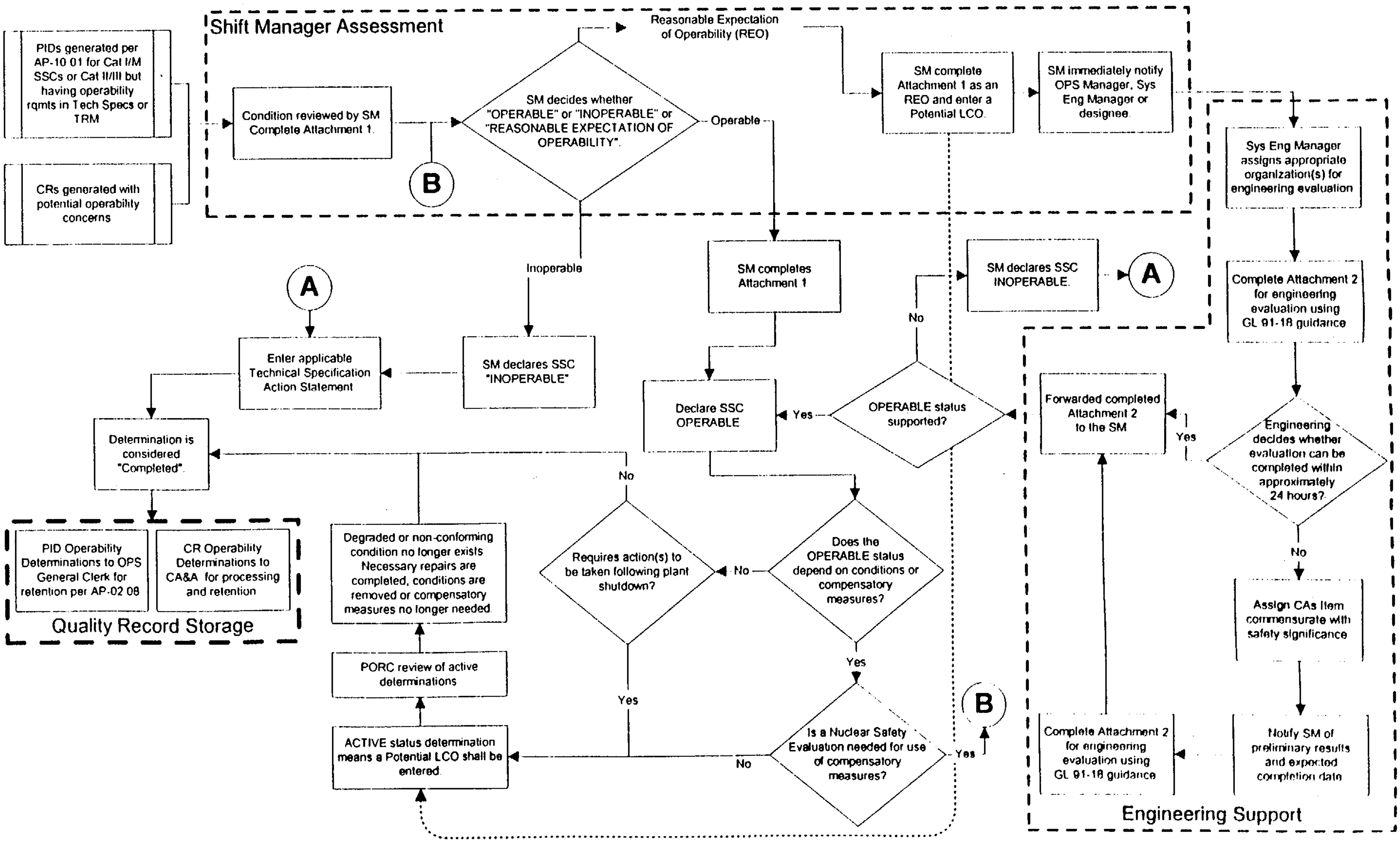
ATTACHMENT 9 Page 1 of 1

ENGINEERING CONFIRMATION PROCESS FLOWCHART



ATTACHMENT 10

OPERABILITY DETERMINATION PROCESS FLOWCHART



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of communications procedures associated with EOP implementation. (CFR: 41.10 / 45.13)	Group #		
	K/A #	2.4.15	2.4.15
	Importance Rating	3.0	3.5

Proposed Question:

The Shift Manager has implemented the Emergency Plan based on high Drywell pressure and assigned an Operator as the NRC Communicator.

WHICH ONE of the following describes when communications with the NRC may be secured?

RO/SRO

73/97

a) Technical Support Center is activated

b) NRC disconnects or authorizes securing line

c) Transient is over and the plant is recovering

d) Once initial classification notice is provided to the NRC

Proposed Answer: b) NRC disconnects or authorizes securing line

Explanation (Optional):

Technical Reference(s): EAP-1.1 attachment 14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.5.1, EO-2.07 (As available)

Question Source: Bank # Limerick 1 INPO # 12345 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/20/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage. (CFR: 43.5 / 45.12)	Group #		
	K/A #	2.4.26	2.4.26
	Importance Rating	2.9	3.3

Proposed Question:

Given the following conditions:

- You are responding to an electrical fire as a member of the plant's fire brigade team.
- You have brought a Class B/C fire extinguisher to the scene.
- Other members have rigged a fire hose with a solid-stream nozzle.

Which one of the following actions should be taken?

a) Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher.

RO/SRO

74/98

b) Use the fire hose first. If it does not put out the fire, use the Class B/C fire extinguisher.

c) Wait for the fire brigade member assigned to bring a Class D fire extinguisher, then use the Class D fire extinguisher.

d) Do not use the Class B/C fire extinguisher. Put the fire out with the fire hose.

Proposed Answer: a) Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher.

Explanation (Optional):

Technical Reference(s): ______ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: SDL-76, EO-1.05.A (As available)

Question Source: Bank # LaSalle 1 INPO # 11156

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 10/9/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of the RO's responsibilities in emergency plan implementation. (CFR: 45.11)	Group #		
	K/A #	2.4.39	2.4.39
	Importance Rating	3.3	3.1

Proposed Question: You are a licensed Reactor Operator on dayshift, working on the FIN Team. You do not have assigned responsibilities in the Emergency Response Organization (ERO). A transient occurs that results in the declaration of an ALERT Emergency and Protected Area Evacuation. To which of the following locations do you report?

RO/SRO
75/99

a) The Operations Support Center (OSC)
b) The Training Building assembly area.
c) The Technical Support Center (TSC).
d) The Offsite Assembly Area (Airport).

Proposed Answer: a) The Operations Support Center (OSC)

Explanation (Optional):

Technical Reference(s): EAP-10 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-12.5.3, EO-1.18 (As available)

Question Source: Bank # Duane Arnold 1 INPO # 8781 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/20/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of communications procedures associated with EOP implementation. (CFR: 41.10 / 45.13)	Group #		
	K/A #	2.4.15	2.4.15
	Importance Rating	3.0	3.5

Proposed Question: D.W. pressure

The Shift Manager has ~~declared an ALERT~~ implemented the Emergency Plan based on high and assigned an ~~RO~~ operator as the NRC Communicator.

WHICH ONE of the following describes when ~~the NRC Communicator can secure~~ communications with the NRC? may be secured

RO/SRO

73/97

a) Technical Support Center is activated

b) NRC disconnects or authorizes securing line

c) Transient is over and the plant is recovering

d) Once initial classification ~~date~~ notice is provided to the NRC

Proposed Answer: b) NRC disconnects or authorizes securing line

Explanation (Optional): **Tied to G2 K1.04 – make a new tie and ensure that reason for ALERT is due to EOP implementation.**

Technical Reference(s): EAP 1.1 attach. 14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: (As available)

Question Source: Bank # Limerick 1 INPO # 12345

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 1/20/1998

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. (CFR: 43.5 / 45.12)	Group #		
	K/A #		2.4.22
	Importance Rating		4.0

Proposed Question: If Torus level cannot be maintained above 10.75 feet, EOP-4, Primary Containment Control, directs the operator to ensure the HPCI turbine is tripped.

Which of the following describes the bases for RCIC and HPCI operation under the same EOP circumstances (ie, Torus water level cannot be maintained above 10.75 feet)?

RO/SRO

S100

a) RCIC operation may continue ONLY if it is the last operable high pressure injection system available to provide adequate core cooling.

b) RCIC must be secured at the same time as HPCI to minimize the containment pressure rise.

c) RCIC operation may continue because the turbine exhaust energy does not exceed the vent capability of the containment.

d) RCIC must be secured prior to HPCI to prevent erratic turbine operation due to exhaust back pressure fluctuation.

Proposed Answer: c) RCIC operation may continue because the turbine exhaust energy does not exceed the vent capability of the containment.

Explanation (Optional):

Technical Reference(s): EPG/ REV 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: EOP4LP, EO-1.05 (As available)

Question Source: Bank # Fermi 2 2 INPO# 19714 (Modified to JAF)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 6/14/2001

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of facility protection requirements including fire brigade and portable fire fighting equipment usage. (CFR: 43.5 / 45.12)	Group #		
	K/A #	2.4.26	2.4.26
	Importance Rating	2.9	3.3

Proposed Question: Given the following conditions:

-You are responding to an electrical fire as a member of the plant's fire brigade team.□-You have brought a Class B/C fire extinguisher to the scene.□-Other members have rigged a fire hose with a solid-stream nozzle.□Which one of the following actions should be taken?

RO/SRO

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a) Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher.

b) Use the fire hose first. If it does not put out the fire, use the Class B/C fire extinguisher.

c) Wait for the fire brigade member assigned to bring a Class D fire extinguisher, then use the Class D fire extinguisher.

d) Do not use the Class B/C fire extinguisher. Put the fire out with the fire hose.

Proposed Answer: a) Do not use the fire hose. Put the fire out with the Class B/C fire extinguisher.

Explanation (Optional): **Question tied to 294001.K1.16- make a new tie**

Technical Reference(s): ____________________ (Attach if not previously provided)

Proposed references to be provided to applicants during examination: ____________

Learning Objective: SOLP-76 1.02 (As available)

Question Source: Bank # LaSalle 1 INPO # 11156

Modified Bank # ________ (Note changes or attach parent)

New ________

Question History: Last NRC Exam 10/9/1995

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ________

10 CFR Part 55 Content: 55.41 X

55.43 ________

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	3
Knowledge of the RO's responsibilities in emergency plan implementation. (CFR: 45.11)	Group #		
	K/A #	2.4.39	2.4.39
	Importance Rating	3.3	3.1

Proposed Question: You are a licensed Reactor Operator on dayshift, working ~~on outage tagouts in the Work Control Center~~ on the FIN Team. You do not have assigned responsibilities in the Emergency Response Organization (ERO). A transient occurs that results in the declaration of an ALERT Emergency and ~~activation of~~ the Evacuation ~~Alarm~~ Protected Area. To which of the following locations do you report?

RO/SRO

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a) The ~~Control Room~~ OSC Operations Support Center

b) The ~~Warehouse~~ Training Bldg assembly area.

c) The Technical Support Center (TSC).

d) The Offsite ~~Relocation and~~ Assembly ~~Location (ORAL)~~ Area (Airport).

Proposed Answer: a) ~~The Control Room~~ OSC

Explanation (Optional):

Technical Reference(s): EP-12.5.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: 1.18 (As available)

Question Source: Bank # Duane Arnold 1 INPO # 8781

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam 9/20/1999

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 X

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. (CFR: 43.5 / 45.12)	Group #		
	K/A #		2.4.22
	Importance Rating		4.0

Proposed Question: If ~~suppression pool~~ Torus level cannot be maintained above ~~68 inches~~ 10.75 ft, Primary Containment Control EOP ~~Flowchart Step TWL-6~~ directs the operator to "SECURE HPCI (DISREGARD ADEQUATE CORE COOLING)."□Which of the following describes the bases for RCIC and HPCI operation under the same EOP circumstances (ie, ~~suppression pool~~ torus water level cannot be maintained above ~~68 inches~~ 10.75 ft)?

a) RCIC operation may continue ONLY if it is the last operable high pressure injection system available to provide adequate core cooling.

RO/SRO

b) RCIC must be secured at the same time as HPCI to minimize the containment pressure rise.

S100

c) RCIC operation may continue because the turbine exhaust energy does not ~~contribute excessively to increasing containment pressure.~~ exceed the vent capability of the Containment

d) RCIC must be secured prior to HPCI to prevent erratic turbine operation due to exhaust back pressure fluctuation.

Proposed Answer: c) RCIC operation may continue because the turbine exhaust energy does not ~~contribute excessively to increasing containment pressure.~~ exceed the vent capability of the Containment.

Explanation (Optional):

Technical Reference(s): EPG / Rev 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination:

Learning Objective: MTT 301.11E 1.05 (As available)

Question Source:	Bank #	Fermi 2 2 INPO# 19714	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	6/14/2001	

(Optional - Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	X

Comments: