FACILITY POST-EXAMINATION COMMENTS

FOR THE PALISADES INITIAL EXAMINATION - MAY 2005

June 7, 2005

NUREG 1021, ES-402

Regional Administrator
U.S. Nuclear Regulatory Commission
2443 Warrenville Road
Suite 210
Lisle, IL 60532-4352

Palisades Nuclear Plant
Docket 50-255
License No. DPR-20

Initial License Examination Comments

In accordance with NUREG-1021, ES-402, Nuclear Management Company LLC (NMC) is submitting comments on the initial license examination administered at the Palisades Nuclear Plant during May 2005.

Enclosure 1 contains the requested information.

for
Daniel J. Malone
Site Vice President, Palisades Nuclear Plant
Nuclear Management Company, LLC

Enclosure (1)

CC Project Manager, Palisades, USNRC (w/o enclosure)
Resident Inspector, Palisades, USNRC (w/o enclosure)
Bruce Palagi, Region III, USNRC

ENCLOSURE 1

PALISADES INITIAL LICENSE EXAMINATION COMMENTS

42 Pages Follow

Question # 13

During a Station Blackout what indication(s) are available to determine when Battery No. 1 (D01) is approaching a fully discharged condition?

A. **ONLY** Voltage indication for Battery No. 1 can be used.

B. **EITHER** Voltage or Amperage indications for Battery No. 1 can be used.

C. **ONLY** Amperage indications for Battery No. 1 can be used.

D. **EITHER** Voltage, Amperage, CR annunciator, or Frequency indications for Battery No. 1 can be used.

NRC Answer Key: B

Facility Comment:

Distractor B: "**Either** Voltage **OR** Amperage..." could be interpreted to imply that either voltage ALONE, or amperage ALONE could be used, but NOT both. While amperage does respond and may be helpful in diagnosing a battery near a fully discharged condition it cannot be used alone. High or low amperage can be indicative of battery loading. Without a relative voltage reading, amperage indication alone is not adequate for diagnosing a battery approaching a fully discharged condition.

EOP-3.0 Station Blackout, requires that if bus voltage drops to 105 volts that the shunt trip push button be pressed for that bus. This ensures the battery can perform its safety function prior to being overdutied. The requirement does not mention bus amperage. Therefore, Distractor A is also acceptable.

Facility Recommendation: Accept both A and B as correct.

Additional Facility References (Attached):

1. EOP-3.0, Station Blackout Recovery, Step 20 excerpt, page 22 of 52
2. EOP-3.0, Station Blackout Recovery, Step 20 Basis, page 49, 50 of 177

INSTRUCTIONS | CONTINGENCY ACTIONS

CAUTION

The following step should only be performed as a last resort since it results in separating the respective DC Bus from the Station Battery. Buses D11A and D21A will still be supplied from the Station Batteries.

20. IF there is an obvious DC Bus problem which can NOT be immediately corrected, THEN **PERFORM** ALL of the following:

    a. IF the condition is indicated on 125V DC Bus D10 OR bus voltage drops to 105 volts, THEN **PUSH** Shunt Trip pushbutton "D-11 Incoming Power Trip" on Panel D11A.

    b. IF the condition is indicated on 125V DC Bus D20 OR bus voltage drops to 105 volts, THEN **PUSH** Shunt Trip pushbutton "D-21 Incoming Power Trip" on Panel D21A.

    c. **GO TO** EOP-9.0, "Functional Recovery Procedure" AND **REFER TO** ONP-25.2, "Alternate Safe Shutdown Procedure."

CEN-152 SBO Step:

None

Technical Basis:

The intent of this step is to respond to an obvious DC Bus problem that can not be immediately corrected. By stripping the battery from D10 or D20 the potential for having the DC buses for operation of the D/G and its associated loads is greatly improved.

Depending on the affected bus, the operator is directed to separate DC Bus D10 or D20, from it's respective Station Battery by pushing the associated Shunt Trip pushbutton on Panel D11A or D21A.

In order to maintain a full compliment of safety grade instrumentation when the Reactor is not in a refueling condition, three of the four Preferred AC Buses must remain energized. In order to maintain three of the four Preferred AC Buses energized, at least three of the four vital DC bus sections must be energized (requiring both DC Buses D10 and D20 to be energized) or two inverters must be powered by one vital DC bus and a third inverter powered by the Bypass Regulator.

Stripping one or both of the DC Buses D10 or D20 will result in less than three of the four Preferred AC Buses being energized. This condition will warrant exiting this procedure and going to EOP-9.0, "Functional Recovery Procedure." ONP-25.2, "Alternate Safe Shutdown Procedure," is referenced to assist the operator with equipment control since DC control power from the deenergized DC Bus is not available.

Associated Notes, Cautions, Warnings:

Caution alerts the operator that these actions should only be performed as a last resort since it results in separating DC Bus D10 or D20 from the Station Battery. Buses D11A and D21A will still be supplied from the Station Batteries and provide power to relays required to open the battery supply breakers to Buses D10 and D20 respectively. De-energizing D10 or D20 does not remove battery power to Bus D11A or D21A.

Deviations from EPG:

A step to respond to a DC Bus problem that can not be immediately corrected is an addition to CEN-152 requirements.

Justification for Deviation:

Indication of an electrical ground or low voltage condition on either DC Bus must be promptly addressed in order to prevent the problem from affecting the battery's capability to perform to its analyzed availability rating. If the shunt trip breaker is open separating the DC Bus from its associated battery, then the safety function status check condition that three of the four Preferred AC Buses be energized will not be met. This condition creates the necessity of exiting to EOP-9.0, "Functional Recovery Procedure."

Question #23

The plant is operating at 100% Rx power when a failure of Cooling Tower Pump P-39A has caused condenser vacuum to degrade. Loss of Condenser Vacuum procedure ONP-14 has been entered. A rapid power reduction (per ONP-26) was ordered by the SRO. Following the power reduction, and reactor trip, condenser pressure stabilized at 15" Hg. During the rapid downpower, what was the fastest allowable rate of power reduction, and assuming condenser pressure remains constant what would PCS temperature be after the reactor trips?

A. 60%/Hr and 532°F

B. 300%/Hr and 532°F

C. 60%/Hr and 535°F

D. 300%/Hr and 535°F

NRC Answer Key: B

Facility Comment:

The question stem asks, "what would PCS temperature be." The briefing provided to the candidates just prior to the exam, in accordance with Appendix E of NUREG 1021, Rev. 9, instructed them to answer all questions based on actual plant operation, procedures, and references, and that if they believe the answer would be different based on simulator operation or training references, they should answer based on the *actual plant.*

By design, the turbine bypass valve (TBV) does control main steam header pressure at 900 psia (531.95 degrees F at saturation). However, pressure losses between the main steam header and the steam generators, along with efficiency losses in the steam generators, result in a stable Tave of slightly less than 535 degrees F.

This question and answer B reflect system design, but not actual plant response. Please see attached copies of both actual plant data and simulator response that show that actual PCS temperature (Tave) stabilizes at approximately 535 degrees F with the turbine bypass valve available.

Facility Recommendation: Change correct answer to D.

Additional Facility References (Attached):

1. PPC trend page from reactor trip report dated 07/21/98 with TBV available.
2. PPC trend page from simulator reactor trip with TBV available.

267 TREND PLOT 1s GROUP 1 PAGE 7 SIM 10/28/2000 12:48:02

FILE ID: _C_19980721_1505 07/21/98 MFP TRIP

1200.00

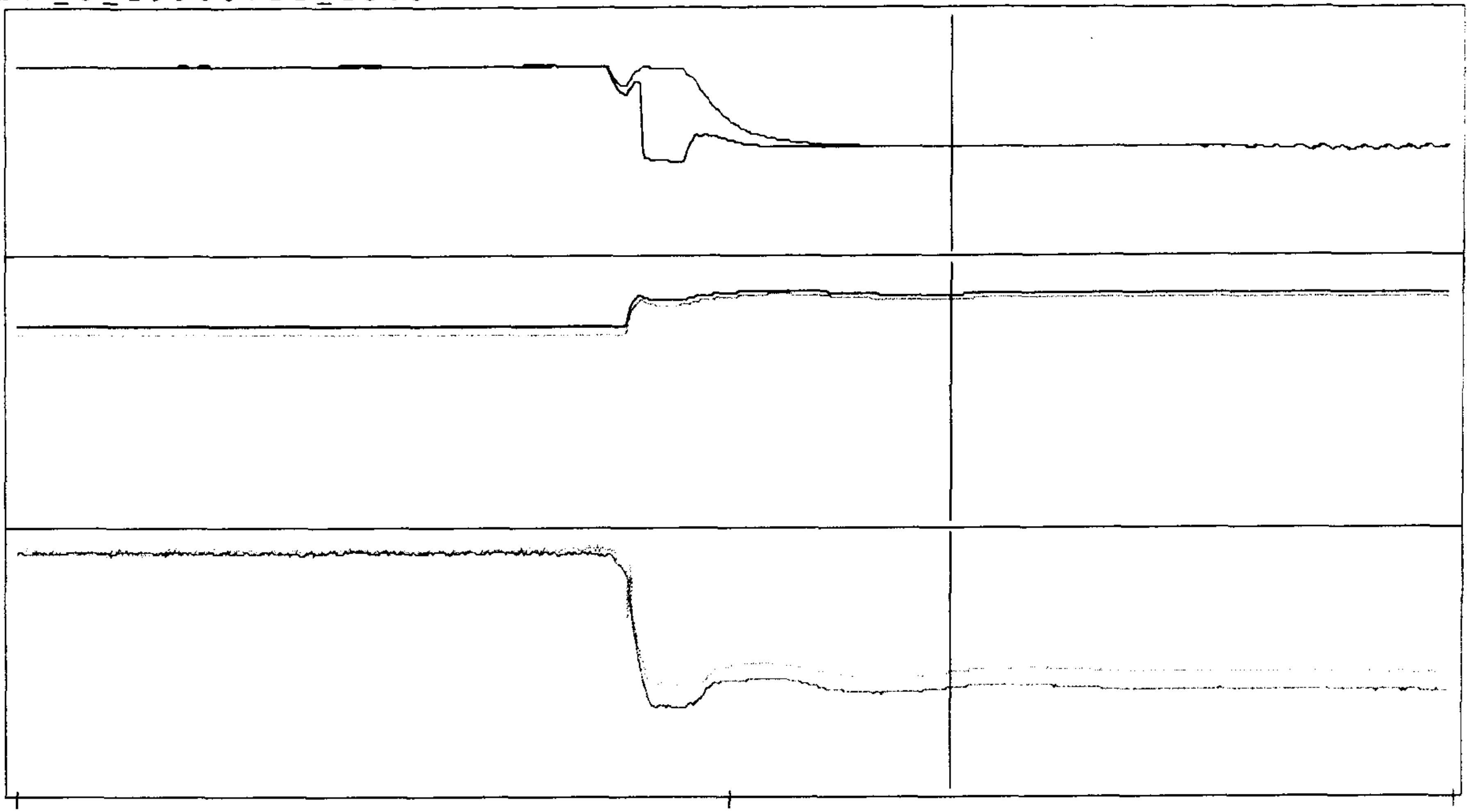
0.00

1000.00

67.00

70.00

0.00



07/21/1998 14:44:20 07/21/1998 14:50:20 07/21/1998 14:56:19

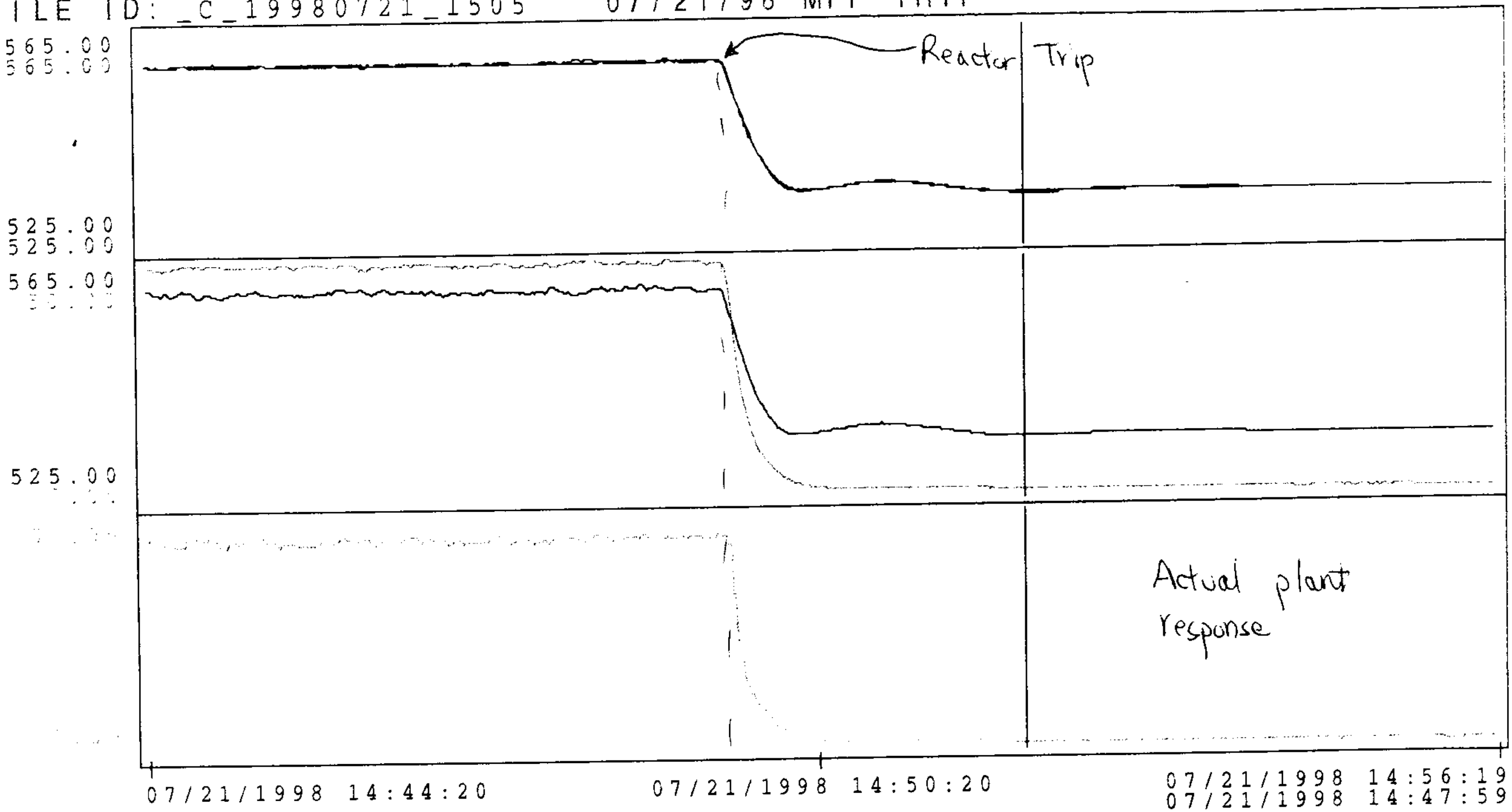
07/21/1998 14:47:59

PT0701	Stm Gen Feed Pump P-1A Disch	psig	531.87
PT0703	Stm Gen Feed Pump P-1B Disch	psig	533.63
PT0751B	Steam Generator E-50A Press	psia	885.86
[illegible]	[illegible]	[illegible]	887.62
[illegible]	[illegible]	[illegible]	31.70
LSGBBE	Stm Gen B Lvl Best Estimate	percent	27.52

F6 1 TREND WINDOW | F7 FAST | F8 ARCHIVE MENU | F9 | F10 | F11 | F12 | F13 TREND ZOOM | F14 SHIFT <<<-- | F15 SHIFT -->>>

267 TREND PLOT 1s | GROUP 1 | SIM
PAGE 1 | 10/28/2000 12:48:02

FILE ID: _C_19980721_1505 07/21/98 MFP TRIP



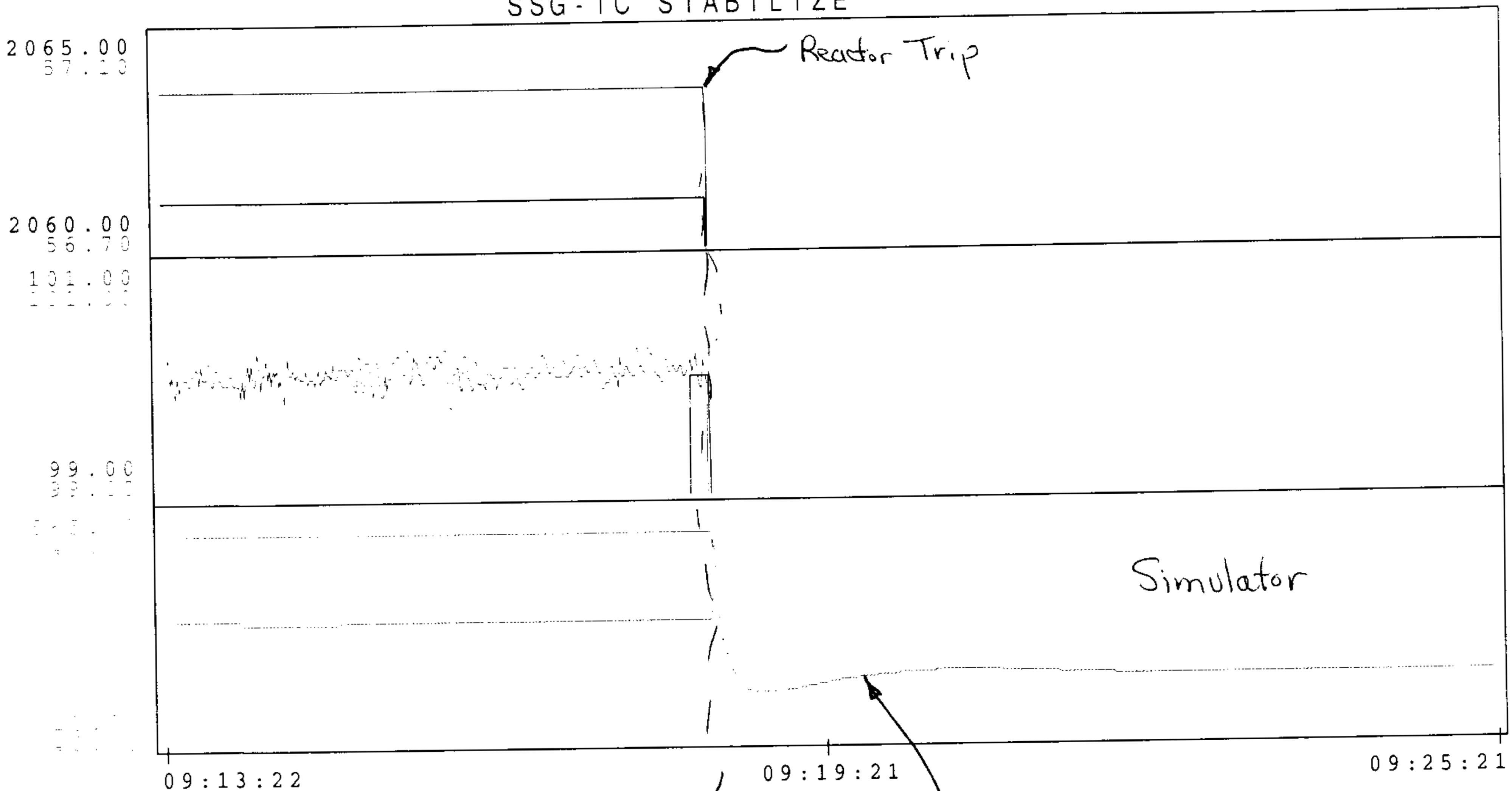
TAVG	Average Temp Hot & Cold Legs	deg F	534.38
TYT_0100	PCS LOOP1 Tav	deg F	534.57
TYT_0200A	PCS LOOP2 Tav	deg F	535.06
[illegible]	PCS LOOP1 Delta T	deg F	1.78
[illegible]	[illegible]	[illegible]	1.58
			558.97

OK

F6 1 TREND WINDOW | F7 FAST | F8 ARCHIVE MENU | F9 | F10 | F11 | F12 | F13 TREND ZOOM | F14 SHIFT <<<-- | F15 SHIFT -->>>

104 TREND PLOT 1s GROUP 51 PAGE 1 SIM 06/03/2005 09:25:21

SSG-IC STABILIZE



PT0105A	Pressurizer Wide Range Press	1954.17	psia
LT_0101A_D	Pzr T-72 Level	44.94	percent
HB_PWR_RAW	Heat Bal Pwr - Unfiltered	-0.68	percent
[illegible]	Power Range Safety Channel A	0.00	percent
[illegible]	Average Temp Hot & Cold Legs	535.41	
[illegible]	Charging Lin. Flow	41.43	[illegible]

OK

F6 1 TREND WINDOW | F7 MEDIUM | F8 SLOW | F9 BASE | F10 HOUR | F11 8 HOUR | F12 DAY | F13 TREND ZOOM | F14 SHIFT <<<-- | F15 SHIFT -->>>

Question #66

A plant shutdown is required for refueling. When can the Operating Crew declare that they have reached Mode 6?

A. When the Reactor Head is removed with SDM > 1%

B. When the Reactor Head is removed with SDM N/A

C. When the first Reactor Vessel Closure Bolt less than fully tensioned with SDM > 1%

D. When the first Reactor Vessel Closure Bolt less than fully tensioned with SDM N/A

NRC Answer Key: D

Facility Comment:

The question does not ask for the definition of Mode 6. The stem presents a decision point and asks, "When can the Operating Crew declare that they have ***reached*** Mode 6?" As soon as the first reactor vessel closure bolt is less than fully tensioned, the conditions of the stem are met. Since both answers C and D contain this condition (less than fully tensioned), and since SDM is N/A for Mode 6, answers C and D are both correct.

Answers A and B are not correct, since the crew would have declared Mode 6 entry long before the conditions of A and B are true.

Facility Recommendation: Accept both C and D as correct.

Additional Facility References (Attached):

1. Technical Specification Table 1.1-1, MODES

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE PRIMARY COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 300
4	Hot Shutdown[b]	< 0.99	NA	$300 > T_{ave} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

Question #73

The following plant conditions exist:

- All Waste Gas Decay Tanks are full except the tank currently in service
- A Containment Purge is in Progress
- D/G 1-2 is currently running for surveillance testing
- Minimum crew manning is onsite due to a Holiday

Waste Gas Decay Tank T-68B needs to be released but Radiation Monitor RE-1113 is NOT OPERABLE. *What conditions must exist for the WGDT to be released?*

A. Radiation Monitor RE-1113 must be returned to OPERABLE status
The Containment Purge must be secured

B. Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
Plant Stack Radiation Monitor is continuously monitored throughout the release

C. *Two independent tank samples are collected*
Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
The Containment Purge must be secured

D. Two independent tank samples are analyzed
Two independent verifications of the release rate calculation are performed
Two qualified Aux. Operators independently verify the WGDT discharge line-up
Plant Stack Radiation Monitor is continuously monitored throughout the release

NRC Answer Key: C

Facility Comment:

The stem of this question only lists some of the conditions needed to be in place to release a gas batch. It does not list ALL required conditions (e.g., main exhaust fan must be in service).

Answer C is correct, since it is reasonable to assume that a "collected" sample would also be "analyzed."

The last requirement in answer D, "Plant Stack Radiation Monitor is continuously monitored throughout the release," was originally intended to be incorrect, with the other three items being correct. However, the attached references show that the plant stack radiation monitor is continuously used as a monitoring instrument. Therefore, D is also correct.

Facility Recommendation: Accept both C and D as correct.

Additional Facility References (Attached):

1. SOP-18A, Radioactive Waste System - Gaseous, p. 12, 13
2. Design Basis Document (DBD) 3.2.1.10, p. 67
3. Health Physics Procedure HP 6.6, Evaluation and Release of Waste Gas Decay Tank, p. 12
4. Health Physics Procedure HP 6.51, *Radioactive Effluent Monitoring* Instrumentation and Equipment Requirements, Attachment 5, p. 2
5. Initial License Training Lesson Plan RMS, Radiation Monitoring System, p. 65 through 69.

NOTE: The reason that MV-WG500 is closed in the following step is that DT-1111 and/or CK-WG404 have historically been problematic. Leakage past this check valve or drain trap can result in fission gas in the Auxiliary Building or the Waste Gas Decay Tank Discharge Header and ultimately make its way to the Stack.

e. WHEN WGST pressure stops decreasing, THEN **CLOSE** the following valves:

1. MV-CRW782, DT-1111 Bypass Valve. (Waste Gas Surge Tank Room)

2. MV-WG500, WGST Drain Valve. (Waste Gas Surge Tank Room)

f. **PLACE** in AUTO C-50A and C-50B OR C-54.

7.5 TO RELEASE GAS FROM WASTE GAS DECAY TANKS

a. **CHECK** Containment purge is NOT in progress. **REFER TO** HP 6.14, Attachment 1, "Containment Purge Data."

b. **OBTAIN** Authorized Batch Release Order from Shift Manager.

c. **CHECK** Radiation Monitor RE-1113 operable per the following applicable procedure:

- Technical Specification Surveillance Procedure DWO-1, "Operator's Daily/Weekly Items Modes 1, 2, 3, and 4"
- Technical Specification Surveillance Procedure DWO-2, "Operator's Daily/Weekly Items Modes 5 and 6"

d. IF RE-1113 is not operable, THEN the WGDT may be released providing all of the following conditions are met:

1. At least two independent samples of the tank contents are analyzed prior to release.

2. At least two independent verifications of the release rate calculations have been performed.

3. At least two qualified Auxiliary Operators independently verify the WGDT discharge line-up correct.

e. **ENSURE** that at least one V-6A or V-6B Main Exhaust Fan is operating and record on HP Form 6.6-3.

f. IF the Main Exhaust Fans trip during WGDT release, THEN immediately **STOP** batch release. This may be done from the Control Room by lowering the trip setpoint on RIA-1113 until CV-1123, WGDT Discharge, trips closed.

NOTE: Pressure settings for operation of HIC-1123 are as follows:

Position	Pressure
Full Closed	0-3 psig
1/8 Open	4-9 psig
Full Open	10-15 psig

1. **PLACE** to CLOSED position HIC-1123, Waste Gas Discharge To Stack.

2. **ENSURE CLOSED** the air supply valves for the following WGDT Discharge Valves:

   MV-CA363, INST AIR SUP CV-1119A, T-68A DISCHARGE
   MV-CA364, T-68B OUTLET CV-1120A A/S
   MV-CA365, DECAY TANK T-68C VLV CV-1121A AIR SUP
   MV-CA367, T-101A OUTLET CV-1160 A/S
   MV-CA368, T-101B OUTLET CV-1161A A/S
   MV-CA369, T-101C OUTLET CV-1162A A/S

3. **ENSURE CLOSED** the following:

   - MV-WG719, T-68A/B/C Outlet Isolation
   - MV-WG718, T-101A/B/C Outlet Isolation
   - MV-WG718A, T-101A/B/C Outlet Isolation

g. **AIR PURGE** RE-1113 to reduce background count rate AND **RECORD** on HP Form 6.6-3.

h. **SOURCE CHECK** RE-1113 prior to batch release AND **RECORD** cpm on HP Form 6.6-3.

i. **RECORD** RIA-1113 background on Form HP 6.6-3.

j. **DETERMINE** Hi Alarm setpoint. (This is the sum of the observed background and the established release limit provided on Form HP 6.6-3.)

3.2.1.10 Main Exhaust Fan System

The air exhausted from the Radwaste Area HVAC System, the Fuel Handling Area HVAC System and the Containment Air Room Purge System all combine in the Main Exhaust Plenum. The plenum is located between the 590' and 607' elevations of the Auxiliary Building. Access to the plenum is via the 607' elevation in the Main Steam Line Penetration Room (2nd Level CCW Room). Two Main Exhaust Fans V-6A (EMB-1215) and V-6B (EMB-1111) draw a suction from the plenum and discharge to the Plant Exhaust Stack via dampers PO-1816 (V-6A) and PO-1818 (V-6B). The fans are currently only run one at a time and are used to keep the main exhaust plenum at a negative pressure compared to atmosphere. This ensures that the other HVAC system exhaust fans (V-8s, V-14s, V-68s, V-70s and Containment) have a place to exhaust their air without reversing flow on one of the other fans. The negative pressure main exhaust plenum also provides the location for the Vent Gas Collection Header to exhaust (the Vent Gas Collection Header, though, is outside the scope of this DBD) with the V-6s providing the negative pressure motive force for gasses to be exhausted via the header. The stack provides a mixing area for gasses to be discharged from the plant (these gases and their release points are outside the scope of this DBD). The gasses that are released include the discharge line from the Waste Gas Decay Tanks, the exhaust from the Steam Jet Air Ejectors and the Flash Tank and Blowdown Tanks relief/vent line. The contents discharged from the stack are monitored by the Radioactive Gaseous Effluent Monitor (RGEM) System (which is also outside the scope of this DBD). RGEM continuously samples the gas in the stack, monitors it for radioactivity and then exhausts the gas back into the stack via the air ejector exhaust line.

Safety Classification

The Main Exhaust Fan System is not safety-related.

5.10.2 Verify that the calculated release rate is greater than 150 CFM (as established in EA-CR-940839-01), and authorize the release by signing Form HP 6.6-3. In the event that calculated release rate for any waste gas decay tank is less than 150 CFM, further holdup (decay) of the tank is required prior to release.

5.11 BATCH RELEASE

5.11.1 Deliver batch authorization to Shift Manager for release. The Shift Manager shall be responsible for completion of sections labeled Auxiliary Operator, Control Room Operator, Shift Manager, and Approval to Release on Form HP 6.6-3.

5.12 RETS EFFLUENT LOGBOOK

5.12.1 The RETS Analyst shall enter the following data in the RETS Effluent Logbook:

a. Release start date and time

b. Release stop date and time

c. $\Sigma Q_i/EC_i$ = Total "μCi/EC" (cc) ÷ release time (sec)

d. Average annual release fraction for the period of release

Fraction = $\Sigma Q_i/EC_i \div 4.7E+11$ cc/sec

6.0 ACCEPTANCE CRITERIA

6.1 Release shall be reviewed for limits being met and for completion of required data. Work Order or Action Request shall be filed if appropriate.

6.2 Following isolation and/or prior to release, WGDT contents shall be sampled and analyzed for compliance with Palisades ODCM Appendix A, Sections III.A, III.B, and III.E and Table B-1 Item A.

6.3 Waste gas holdup time meets requirements of Palisades ODCM Appendix A, Section III.E.

6.4 During release of gaseous wastes to the Plant vent stack, the following conditions shall be met:

6.4.1 At least one main exhaust fan (V-6A or V-6B) should be in operation.

6.4.2 If RIA-1113 is not operational, ensure that the provisions of Palisades ODCM Appendix A, Table A-1 are met.

TABLE 6.51-5

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION AND EQUIPMENT REQUIREMENTS

INSTRUMENT	OPERABLE	APPLICABILITY	ACTION
8. STACK GAS EFFLUENT SYSTEM			
a. Noble Gas Activity Monitor (RIA-2326)	(1)	At All Times	7A
b. Iodine/Particulate/Sampler/Monitor (RIA-2325)	(1)	At All Times	7
c. Sampler Flow Rate Monitor (FE-2346)	(1)	At All Times	6
d. Hi Range Noble Gas (RIA-2327)	(1)	Above 210°F (Modes 1, 2, 3, 4)	8
9. STEAM GENERATOR BLOWDOWN VENT SYSTEM			
a. Noble Gas Activity Monitor (RIA-2320)	(1)	Above 210°F (Modes 1, 2, 3, 4)	5
10. MAIN STEAM SAFETY AND DUMP VALVE DISCHARGE LINE			
a. Gross Gamma Activity Monitor (RIA-2323 and RIA-2324)	(1) per Main Steam Line	Above 325°F (Modes 1, 2, 3, 4)	8
11. ENGINEERED SAFEGUARDS ROOM VENT SYSTEM			
a. Noble Gas Activity Monitor (RIA-1810 and RIA-1811)	(1) per Room	Above 210°F (Modes 1, 2, 3, 4)	12
12. WASTE GAS DECAY TANKS			
a. T-68A, T-68B, T-68C, T-101A, T-101B, and T-101C	(1)	Min 15-60 day decay	10
13. CONTAINMENT HIGH RANGE GAMMA MONITORS			
a. RIA-2321, RIA-2322	(2)	Above 210°F (Modes 1, 2, 3, 4)	11

VA-81 RIA-5712 Auto Actions

4) Automatic Actuations

a) On a HIGH alarm, RIA-5712 automatically trips Auxiliary Building ventilation fan V-69.

b) The exhaust fan selected for standby, V-70A or V-70B, may or may not trip.

c) For fans that trip, the associated damper closes.

o. RIA-2325, 2326, and 2327, Radioactive Gas Effluent Monitoring System (RGEMS)

VA-82 RGEMS Purpose

1) Purpose

a) RIA-2325, RIA-2326, and RIA-2327 are the process radiation detectors in the Radioactive Gas Effluent Monitoring System (RGEMS).

(1) Analog channels RIA-2325 and RIA-2326 are normal range particulate/iodine and noble gas detectors, respectively.

(2) Digital channel RIA-2327 is a high-range noble gas monitor.

b) The RGEMS monitors actual stack effluent for radioactive contamination greater than expected.

(1) It serves as a backup to all effluent streams that are discharged through the stack, such as the ventilation exhaust monitors, the gaseous waste discharge monitor, and the condenser off-gas monitor

VA-83 RGEMS Flow Path

2) Flow Path

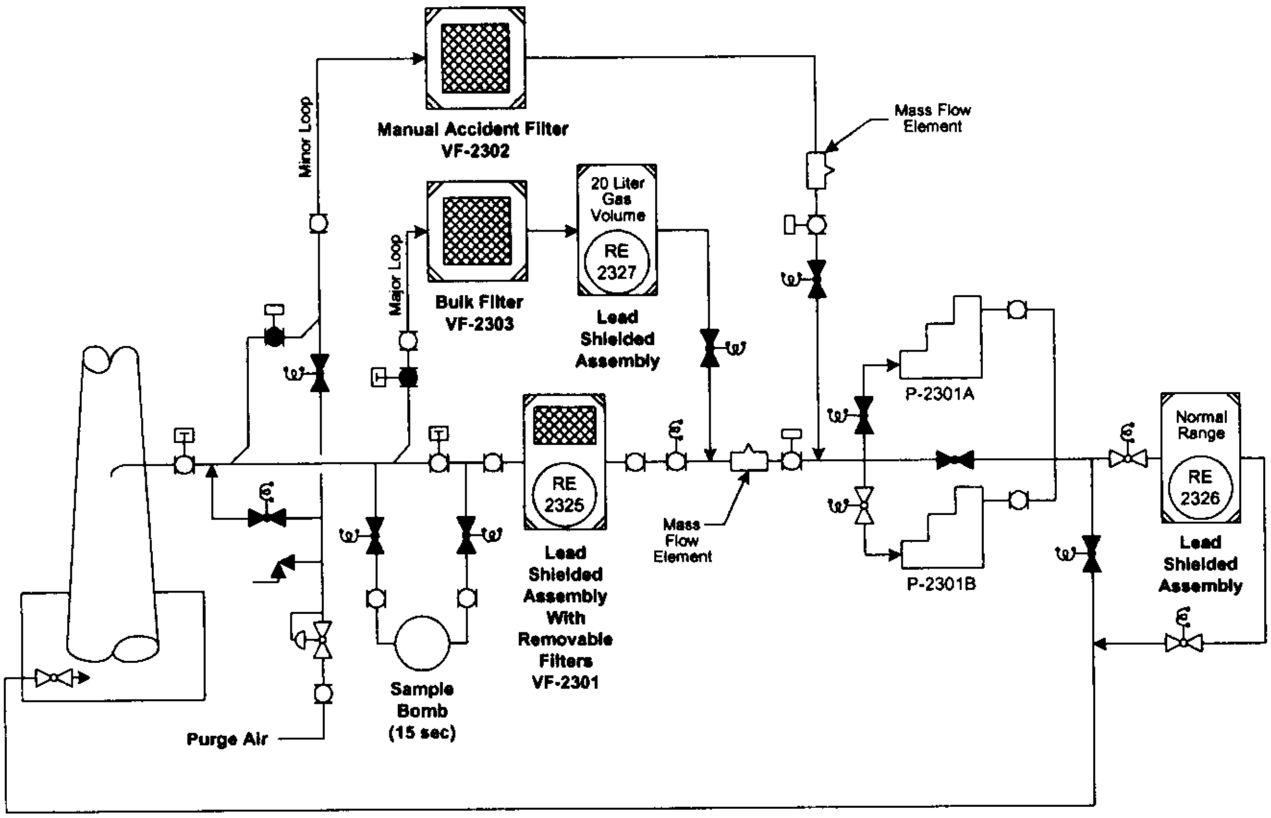


Figure 15. RGEMS Flow Path

a) An isokinetic nozzle draws sample gas from the stack flow volume.

(1) Two 100% capacity diaphragm-operated variable-speed positive displacement pumps control the flow rate into the isokinetic nozzle.

(a) Only one pump is in service at a time.

(2) By controlling pump speed, a flow controller matches the flow speed of the gas sample entering the isokinetic nozzle with the flow speed of gas as it travels up the stack. This ensures that the RGEMS monitors a very representative sample of actual stack flow.

b) Normally, the sample gas flows into a lead-shielded assembly that contains a particulate and iodine filter and radiation detector RE-2325.

(1) The detector monitors the buildup of particulates and iodine from the gas sample that collect on the filter.

c) Filtered sample gas leaves the particulate/iodine assembly and then enters the suction of the positive displacement pumps, P-2301A and P-2301B.

d) From the pump discharge, the gas sample enters another lead-shielded assembly that contains radiation detector RE-2326.

(1) RE-2326 *monitors the gas sample for* radioactive noble gases.

e) After leaving the noble gas assembly, the sample gas is returned to the stack base enclosure, where it mixes with all the air entering the stack for discharge.

VA-84 RGEMS Alert Flow Path

f) On an ALERT condition (from RIA-2326 only), the sample gas is automatically routed to a grab-sample bomb for 15 seconds.

(1) The diversion to the sample bomb is upstream of the RE-2325 iodine/particulate *filter chamber.*

VA-85 RGEMS High Flow Path

g) On a HIGH radiation condition (from RIA-2326 only), the normal monitoring loop is bypassed.

(1) The sample gas is routed through two parallel paths. About 0.2 cfm is routed to Manual Accident Filter VF-2302.

(a) *This flow path is known as the "minor* loop."

(2) The remainder of the gas sample (about 1.8 cfm) is routed through a filter assembly, from which it enters a 20-litre lead-shielded chamber that is monitored by RE-2327.

(a) This flow path is known as the "major loop."

<u>Majority</u> of sample flow goes through <u>major</u> loop.

(3) *Both of these parallel flow paths recombine* at the suction of the sample pumps.

(4) As with the normal flow path, the pump discharge flow is routed to the stack base enclosure.

h) The sample lines in the RGEMS are heat-traced to prevent moisture condensation.

3) Alarms and Setpoints

a) RIA-2325 ALERT alarm occurs at 1.4E5 CPM, and actuates Control Room annunciator EK-0219, "STACK EFF RAD C-169 ALERT" (Reflash).

b) RIA-2326 ALERT alarm occurs at 1.6E4 CPM, and actuates Control Room annunciator EK-0219, "STACK EFF RAD C-169 ALERT" (Reflash).

c) RIA-2325 HIGH alarm occurs at 1.5E5 CPM, and actuates Control Room annunciator EK-0207, "STACK EFF RAD C-169 HIGH" (Reflash).

d) RIA-2326 HIGH alarm occurs at 1.3E6 CPM, and actuates Control Room annunciator EK-0207, "STACK EFF RAD C-169 HIGH" (Reflash).

e) A low signal output from RIA-2325, RIA-2326, or RIA-2327 actuates Control Room annunciator EK-0231, "STACK EFF/HT C-169/C-172 FAIL/TROUBLE" (Reflash)

This alarm is also actuated by the following conditions:

(1) Heat trace failure, sensed by low temperature (< 105 degrees F) at temperature switch TS-2302.

(2) Sampling system failure as sensed by programmable controller JIC-2301A or JIC-2301B.

(3) Heat tracing circuit low current at C-172

(4) Heat tracing failure at C-169

(5) A loss of power to RIA-2325, RIA-2326, or RIA-2327

(a) If the loss of power occurs on RIA-2326, then the channel ALERT and HIGH alarm automatic actions will also occur.

VA-86 RGEMS Auto Actions

4) Automatic Actuations

   a) On an ALERT alarm from RIA-2326 (only), sample flow is routed through the sample bomb for 15 seconds. This is enough time to flush existing air from the sample bomb and store a representative sample of stack effluent for retrieval and manual analysis.

   b) On a HIGH alarm from RIA-2326 (only), the normal sample flow path is bypassed and sample flow is diverted to the accident filters and high-range noble gas monitor (RE-2327).

p. RIA-2323 and RIA-2324, Main Steam Monitors

VA-87 RIA-2323 & RIA-2324 Purpose

1) Purpose

   a) Analog channels RIA-2323 and RIA-2324 monitor the main steam flow leaving Steam Generators B and A, respectively.

   b) The monitors are designed to detect radioactive contamination in the main steam flow that would indicate a primary-to-secondary leak into a steam generator.

VA-88 RIA-2323 & RIA-2324 Flow Path

2) Flow Path

   a) Each detector resides in a lead-shielded collimator adjacent to its respective main steam pipe, near the main steam safety valves.

3) Alarms and Setpoints

   a) RIA-2323 WARN alarm occurs at 215 CPM and activates Control Room annunciator EK-0217, "MAIN STEAM E-50B RIA-2323 ALERT."

   b) RIA-2324 WARN alarm occurs at 215 CPM and activates Control Room annunciator EK-0218, "MAIN STEAM E-50A RIA-2324 ALERT."

   c) RIA-2323 HIGH alarm occurs at 1260 CPM and activates Control Room annunciator EK-0205, "MAIN STEAM E-50B RIA-2323 HIGH."

Question #82

Given the following:

- Power level is stable at 100%.
- Pressurizer level is being controlled by Pressurizer Level Controller LIC-0101A.
- The output of level controller LIC-0101A has just failed at 100% output signal.
- No other failures occur.

Assuming no Operators actions, what will charging flow be after the level controller output failure, and what is the expected plant response?

A. 0 gpm; and the Reactor will trip on Thermal Margin/Low Pressure.

B. 33 gpm; and Pressurizer level cycles in an approximately 11% band.

C. 44 gpm; and Pressurizer level stabilizes at 57%.

D. 133 gpm; and the Reactor will then trip on High Pressurizer Pressure.

NRC Answer Key: B

Facility Comment:

This question has no correct answer. The correct answer was selected originally based on an understanding of the backup pressurizer level control system design, specifically, that it controls in an approximately 11 percent band. However, with the pressurizer level control malfunction standing, the pressurizer level will actually oscillate over a 2 percent range, the range between where the backup program takes control (~-6%) and where it gets a signal to reset (~ - 4%).

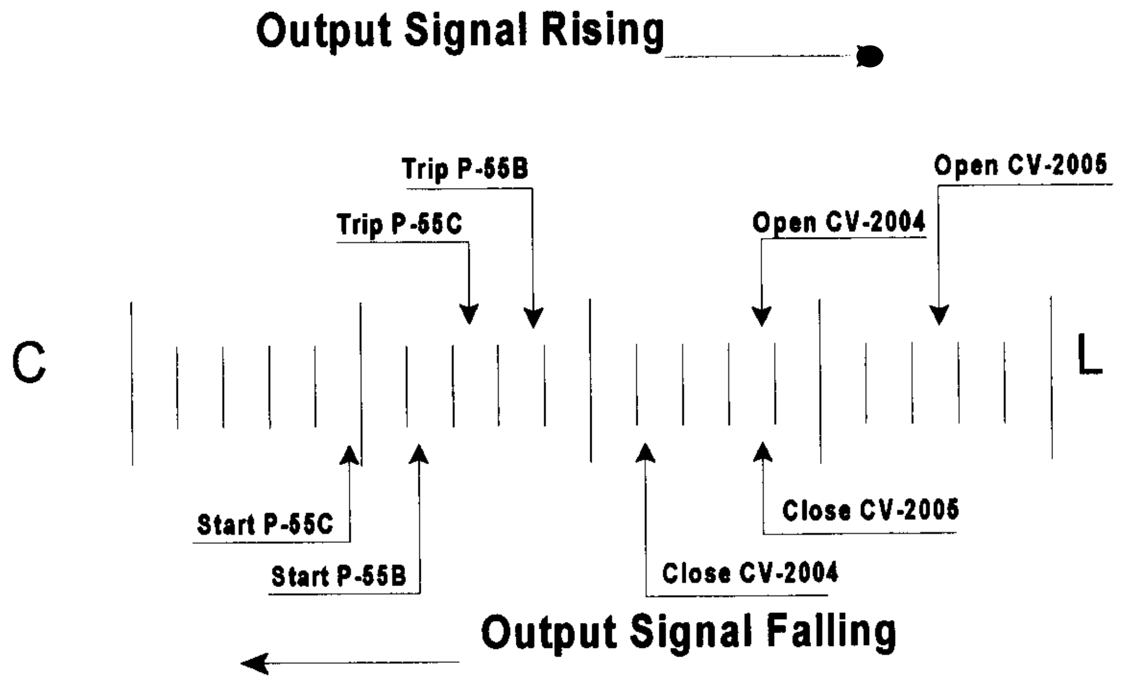
Facility Recommendation: Delete question from exam since no correct answer is provided.

Additional Facility References (Attached):

1. System Operating Procedure SOP-2A, Attachment 2, Pressurizer Level Control Table

PRESSURIZER LEVEL CONTROL TABLE

PRESSURIZER LEVEL CONTROL PROGRAM



Backup **PZR Level Control Program**
Deviation From *Calculated* Setpoint

~ + 6 %	High / Low Alarm On
~ + 5 %	Stop P-55 B&C, Open #2 & 3 Orifice, Backup Heaters On
~ + 4 %	(Reset) Hi / Low Alarm Off
~ + 2 %	(Resets) Backup Heaters Off, Permit P-55B & C Start, Permit #2 & 3 orifice closure

---

2% band

~ - 4 %	(Resets) Hi / Low Alarm Off, Permit P-55B & C Stop, Permit #2 & 3 orifice open
~ - 6 %	Start P-55B & C, Close #2 & 3 Orifice, High / Low Alarm On

Question #95

All plant equipment functioned as designed following a Large Break LOCA. When and why are the Charging Pump suctions aligned to the SIRWT in EOP-4.0, Loss of Coolant Accident Recovery?

A. Approximately 30 to 45 minutes; to reduce the effects of boric acid precipitation in the core

B. Approximately 30 to 45 minutes; to prevent Charging Pump cavitation due to loss of suction

C. Within 1 hour; to ensure adequate SIRWT inventory is injected into the PCS/Containment

D. Within 1 hour; to ensure adequate shutdown margin is established

NRC Answer Key: A

Facility Comment:

The concern for boric acid precipitation in the core is addressed by securing emergency boration. Refer to EOP-4.0 Basis, Step 19, and EOP Supplement 40 Basis.

Re-aligning charging pump suction from the concentrated boric acid storage tanks to either the volume control tank (VCT) or the safety injection refueling water tank (SIRWT) is done for the purpose of flushing the lines associated with boric acid injection. However, it does *assist* in reducing the effects of boric acid precipitation in the core. Therefore, answer A is correct.

During a LBLOCA, once shutdown margin (SDM) requirements are met, emergency boration would be secured. However, prior to this, charging pump suction would be re-aligned to a lower boron source (SIRWT) for the purpose of flushing the injection lines, as noted in the previous paragraph. Refer to EOP Supplement 40, Charging Pump Suction Alignment. At this point, boron is still being injected, only from a lower concentration source. Once the flush is complete, boration would be secured by shutting off charging pumps. This action is the one that addresses the concern for excess boron in the PCS (boron precipitation).

The stem is worded to ask when and why the suction source of the charging pumps would be re-aligned to the SIRWT. It is reasonable that answer D is also correct; i.e., the action given in the stem (swapping suction to SIRWT) is done only after emergency boration is secured, and emergency boration is only secured once adequate SDM is established. This would occur within one hour of the condition stipulated in the stem of the question. Refer to EOP-4.0 Basis, Step 43.

Facility Recommendation: Accept both A and D as correct.

Additional Facility References (Attached):

1. EOP Supplement 40, Charging Pump Suction Alignment
2. EOP Supplement 40, Charging Pump Suction Alignment Basis
3. EOP-4.0, Loss of Coolant Accident Recovery Basis for Step 19
4. EOP-4.0, Loss of Coolant Accident Recovery Basis for Step 43

TITLE: Charging Pump Suction Alignment

1.0 ALIGN CHARGING PUMP SUCTION TO THE VCT

___ 1. Open Charging Pumps Suction VCT Outlet Valve, MO-2087.

___ 2. Stop the Boric Acid Pumps.

- P-56A
- P-56B

___ 3. Close the following valves:

- Boric Acid Pump Feed Valve, MO-2140
- Gravity Feed Valves
    - MO-2169
    - MO-2170

___ 4. Ensure closed Charging Pumps Suction From SIRWT, MO-2160.

___ 5. Operate each Charging Pump for at least five minutes.

___ 6. **WHEN** each pump has been operated at least five minutes, **THEN** operate Charging Pumps as necessary to maintain safety functions.

2.0 ALIGN CHARGING PUMP SUCTION TO THE SIRWT

___ 1. Open Charging Pumps Suction From SIRWT Valve, MO-2160.

___ 2. Stop the Boric Acid Pumps.

- P-56A
- P-56B

___ 3. Close the following valves:

- Boric Acid Pump Feed Valve, MO-2140
- Gravity Feed Valves
    - MO-2169
    - MO-2170

___ 4. Ensure closed Charging Pumps Suction VCT Outlet Valve, MO-2087.

___ 5. Operate each Charging Pump for at least five minutes.

___ 6. **WHEN** each pump has been operated at least five minutes, **THEN** operate Charging Pumps as necessary to maintain safety functions.

Completed By: ____________________

Date/Time: ________/________

Reviewed By: ____________________ (SS)

EOP Supplement 40 Technical Basis:

The intent of this supplement is to provide the steps necessary to align operating Charging Pumps from the Boric Acid supply to either the VCT or the SIRWT.

Emergency boration is performed during the initial stages of plant events to ensure an adequate Reactor shutdown margin exists. This supplement provides the steps for changing the Charging Pump suction to a lower boron concentration source to prevent excess boron in the PCS and possible boron precipitation. Each Charging Pump is operated at least 5 minutes to flush the high concentration of boric acid out of the associated piping.

Deviations from EPG:

This is a plant specific addition to CEN-152 guidance.

Justification for Deviation:

In numerous locations throughout CEN-152, the guidelines state to control Charging and Letdown. If Charging Pump suction was changed due to emergency boration or SIAS then the Charging Pumps will have suction from the Boric Acid Storage Tank. When adequate boration has occurred the Charging Pumps must have the suction source swapped to a source with PCS boron concentration. This supplement controls the Charging and Letdown system by providing the appropriate suction to the Charging Pumps and a location for Letdown flow.

STEP 19

NOTE: IF emergency boration is in progress, THEN cooldown may commence/continue while the required shutdown margin value is calculated.

© 19. **VERIFY** PCS boron concentration greater than or equal to required boron concentration as verified by sample or hand calculation. Refer to EOP Supplement 35. Determination of Emergency Boration Requirements

a. IF Emergency boration is in progress
AND PCS boron concentration is greater than or equal to required boron concentration, THEN **SECURE** emergency boration. Refer to EOP Supplement 40.

19.1. IF PCS boron concentration is less than required boron concentration, THEN **PERFORM** BOTH of the following:

a. **ENSURE** emergency boration is in progress.

b. WHEN required boron concentration is reached, THEN **SECURE** emergency boration. Refer to EOP Supplement 40.

Please see provided EOP Supp. 40, incl. basis.

CEN-152 LOCA Step:

None

Technical Basis:

The intent of this step is to maintain the **[reactor shutdown]** throughout the cooldown.

To accomplish this, the operator verifies boron concentration is greater than or equal to the required boron concentration by sample results or hand calculation using EOP Supplement 35. The operator may have to emergency borate to obtain the required PCS boron concentration.

If a PT Curve maximum cooldown rate is maintained, Pressurizer level may not be able to be maintained constant during the initial stages of the cooldown. Therefore, Pressurizer level may lower. This outsurge from the Pressurizer will tend to dilute the PCS boron concentration. The possible dilution effects should be considered in determining cold shutdown boron concentration.

Training Emphasis

Cooldown is allowed prior to establishing the required boron concentration as long as emergency boration is in progress. Emergency boration will maintain the required shutdown margin when the cooldown rate is within the allowed technical specification cooldown rate. If plant conditions do not dictate immediately cooling down, then the boron concentration should be established prior to cooling down to T_{AVE} less than 525°F. This is a judgement call that must be made by the on-shift SROs based on present plant conditions.

Associated Notes, Cautions, and Warnings:

This note provides information that the cooldown may commence or continue if emergency boration is in progress. Calculations (EA-MAB-97-003 Rev 0) show that if emergency boration is in progress at 30 gpm using a concentrated boric acid tank and the plant is cooled down within technical specification rates, the reactor shutdown margin will remain within the required technical specification value.

Deviations from EPG:

In the event that shutdown boron concentration is not met, cooldown is allowed as long as emergency boration is in progress. Analysis show that boration at 33 gpm indicated (30 gpm actual) will maintain the reactor shutdown by the required shutdown margin during the cooldown if the cooldown rate is within the allowed technical specification cooldown rate.

Justification for Deviation:

Engineering Analysis EA-MAB-97-003 Rev 0, "Maintaining Shutdown Margin While Emergency Borating And Simultaneously Cooling the PCS At The Technical Specification Limit" allows a cooldown with Technical Specification limits to take place as long as Emergency Boration is in progress. This provides additional options to the operators without jeopardizing shutdown margin.

STEP 43

43. WHEN required shutdown boron concentration has been established (approximately 30 to 45 minutes using all charging pumps), THEN **ALIGN** Charging Pump suction to SIRWT. Refer to EOP Supplement 40.

CEN-152 LOCA Step 33:

*33. [**IF** the charging pumps are taking suction from a concentrated borated water source, **THEN** align suction to the RWT or other suitable source within [1 hour] after the start of the Loss of Coolant Accident].

Technical Basis:

The intent of this step is to realign the Charging Pumps from the Concentrated Boric Acid Tanks (CBATs) to the Safety Injection/Refueling Water Tank (SIRWT) to reduce the effects of boric acid precipitation in the core which may occur due to boil off during a large break LOCA.

For large breaks, the Reactor Vessel refills only to the elevation of the break. Borated water is injected into the Reactor Vessel via the Charging and Safety Injection Pumps while steam is boiled away. This may result in boric acid being concentrated in the Reactor Vessel. Switching the suction of the Charging Pump to the SIRWT helps limit the excessive buildup of boric acid in the Reactor Vessel while still allowing for sufficient long-term reactivity control.

Switching of Charging pumps suction to the SIRWT should be done when required cold shutdown boron concentration has been established (approximately 30 to 45 minutes using all charging pumps) **{time to swap charging pumps suction}**.

Associated Notes, Cautions, and Warnings:

None

Deviations from EPG:

In addition to the time constraints, added boric acid quantity constraints.

Justification for Deviation:

Since time restriction are dependent on the number of operating charging pumps, the quantity of boric acid addition was added. This meets the intent of the guidance of CEN-152. The concern is establish adequate shutdown margin and not add so much boric acid to cause a boron precipitation problem. The term required shutdown boron relates to the minimum required by technical specifications. The upper limit of the equivalent contents of one boric acid tank was chosen to provide adequate boron concentration for refueling concentration. The basis for as specified in technical specification for the quantity of boric acid in one concentrated boric tank with a level of 118 inches of 6 1/4 weight percent boric acid has sufficient boron to establish **[adequate shutdown margin]**.

Question #96

Following a refueling outage, during core reloading in what manner is the core reloaded and why?

A. The core reloading is started at the center of the core and loaded towards the periphery to *ensure both source range detectors are monitoring the core*

B. The core reloading is started near an operable source range detector and loaded to the center of the core so that core uncoupling does not occur

C. The core reloading is started at the center of the core and loaded towards the periphery to ensure a potential critical configuration is not shielded from the source range detectors

D. The core reloading is started near an operable source range detector and loaded to the center of the core so that the initial fuel assemblies are supported by the core barrel

NRC Answer Key: B

Facility Comment:

Answer D is also correct. When reloading the core, or any fuel bundle, procedures require that the bundle be supported on at least one side by either another fuel assembly, or by the core shroud. This is done by starting loading on the peripheral of the core, and working inward, as noted in the provided in Reference 1. Reference 2 describes that the core shroud is an integral part of the core barrel.

Facility Recommendation: Accept both B and D as correct.

Additional Facility References (Attached):

1. EM-04-29, Guidelines for Preparing Fuel Movement Plans, Step 6.1.19
2. Lesson Plan RXVI, "Reactor Vessel and Internals," pages 14 - 19

6.0 PROCEDURE

> **USER ALERT**
> REFERENCE USE PROCEDURE
>
> Refer to the procedure periodically to confirm that all procedure segments of an activity will be or are being performed. Where required, sign appropriate sign-off blanks to certify that all segments are complete.

6.1 GENERAL REQUIREMENTS

NOTE: This procedure is intended to provide guidance for fuel movements. Reactor Engineering personnel may override guidance in this procedure if the proper justification exists.

6.1.1 Fuel movement steps shall be developed by a qualified person and independently reviewed by a qualified person.

6.1.2 Attachment 1 contains a fuel movement example sheet. The functional equivalent can be used for developing and performing fuel movements.

6.1.3 Avoid placing freshly discharged assemblies in locations within two cells of the Spent Fuel Pool walls to minimize gamma heating. (PCR023425 implementing OE13221, "Gamma Heating of Spent Fuel Pool Walls and Floor"~~Reference 3.2.12~~)

6.1.4 Fuel assemblies should decay for one year before placing them into Region II, to reduce gamma irradiation of the boraflex absorbers in the racks.

6.1.5 An evaluation shall be performed before placing fuel assemblies into Region II. This evaluation shall include all Technical Specifications requirements for placing fuel into Region II. This evaluation is typically documented in an Engineering Analysis.

6.1.6 Refer to Technical Specifications DSGN 4.3 for enrichment limits for fuel storage in the New Fuel Storage racks. Technical Specifications essentially limit storage to every other cell of the outer rows 'X' and 'Z' for new fuel storage. The center row 'Y' should not be used due to maximum planar average enrichment limits.

6.1.7 Fuel assemblies shall not be moved from the Reactor Core to the Spent Fuel Pool sooner than that permitted by the current Spent Fuel Pool heat load analysis, as in EA-CCW-87-01, Revision 2, ~~Reference 3.2.10,~~ "Spent Fuel Pool Heat Load and Required SFPHX Component Cooling Water Flow During a DBA." Both minimum decay time and maximum assembly addition rate to the pool shall be observed.

6.1.8 The Spent Fuel Handling Machine (SFHM) mast shall be at an orientation of 30, 120, or 210 degrees when accessing the transfer boxes.

6.1.9 The SFHM camera cannot be turned toward the SFP wall when accessing the cells closest to the wall.

6.1.10 Reference the System Operating Procedure SOP-28, "Fuel Handling System," for unusable cells (ie, stuck bundles) and various other requirements such as use of the Region II racks in the SFP. SOP-28 should be reviewed before designing most fuel shuffle plans.

6.1.11 It is preferable to have the SFHM mast at 90 degrees when accessing the New Fuel Elevator (NFE); however, 0° also works well and 180° will work when the proper attention to the drop light is given. A mast orientation of 270° shall not be used when entering the elevator due to interference with the SFHM camera and the wall.

6.1.12 When making fuel movement plan amendments during the shuffle, pay particular attention to the orientation and location of bundles in the SFP that will go back into the core. This can present a potential error for misloading the core.

6.1.13 Ensure plenty of clearance between the refueling machine camera and the core shroud when planning the refueling machine orientation. A one row buffer is necessary, but two rows are preferred.

6.1.14 K-1 core location requirements: ONLY NE and SW fuel assembly orientations should be used.

6.1.15 Ensure the mast will not contact the SFP walls with the aux hoist installed.

6.1.16 All fuel assemblies to be placed in the North Tilt Pit shall have decayed for greater than one year.

6.1.17 While handling control rods in the core, the Refueling Machine Mast orientation must be 45°, 135°, 225°, or 315°. Control rods are handled at the same orientations used for fuel assemblies in all other locations.

6.1.18 If possible, while placing freshly irradiated assemblies in Region I racks, scatter load the assemblies to reduce sharing common walls between the assemblies. This will help reduce the gamma dose rates on the rack walls, decreasing the required wall vent rate and reducing the risk for an assembly becoming stuck in a cell due to a blocked vent hole.

6.1.19 Fuel assemblies in the core must be supported on at least one side by either another fuel assembly or the core shroud.

6.1.20 If practical and efficient, orient the refueling mast (reactor side) so that the camera is not placed over an empty cell. The operators have a better visual reference when the camera is oriented over an occupied core location.

6.2 CORE OFFLOAD

6.2.1 Temporary placement of any fuel bundle to any core periphery position (adjacent to core shroud) is allowed, provided the nearest fuel bundle is a minimum of two core locations away.

6.2.2 The core shall be off loaded in a manner such that core uncoupling does not occur. This can be accomplished by working from the center of the core toward operable excore detectors. It is imperative that a potential critical configuration is not shielded from the excore detectors.

6.3 CORE RELOAD REQUIREMENTS

6.3.1 Core reload fuel bundle placement in the core shall be limited to the following:

a. Final core location

b. Symmetric Core location

c. Temporary placement of a fuel bundle to any core periphery position (adjacent to core shroud), provided nearest fuel bundle is a minimum of two core locations away.

d. Final core location of lesser burnt fuel

6.3.2 The core shall be loaded in a manner such that core uncoupling does not occur. This can be accomplished by working from operable excore detectors toward the center of the core. It is imperative that a potential critical configuration is not shielded from the excore detectors.

VA – 1 and 11

4. Core Support Assembly

   a. The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the core support plate and support columns, the core shrouds, the core support barrel to pressure vessel snubbers and the core support barrel to upper guide structure guide pins.

   b. The core support assembly is supported at its upper flange from a machined ledge in the reactor vessel flange. The lower end is restrained in its lateral movement by six core support barrel to pressure vessel snubbers. Within the core support barrel are axial shroud plates and former plates that are attached to the core support barrel wall and the core support plate and form the enclosure periphery of the assembled core.

   c. The core support plate is positioned within the barrel at the lower end and is supported both by a ledge in the core support barrel and by 52 columns. The core support plate provides support and orientation for the fuel bundles. Also within the core support barrel just below the nozzles are *four guide pins, which align and prevent excessive* motion of the lower end of the upper guide structure relative to the core support barrel during operation.

   d. The core support plate, 1-1/2 inches thick, is a perforated member with flow distribution and pin locating holes for each fuel bundle. The plate is supported by a ledge and by columns. The ledge on the CSB supports the periphery of the plate, and the plate is pinned, bolted and lock welded to the ledge for maintaining accurate location of the plate.

   e. A series of columns are placed between the plate and the beams across the bottom of the core support barrel. The columns provide stiffness and transmit the core load to the bottom of the core support barrel.

VA – 11, 12

5. Core Support Barrel (CSB)

   a. Cylindrical in shape with an inside diameter of 150 inches and a wall thickness of 1 inch.

   b. Suspended by 4-inch thick flange from the reactor vessel core support ledge.

   c. The core support barrel carries the entire weight of the core and other internals.

   d. A 1.5-inch thick core support plate containing flow distribution holes and fuel assembly alignment pinholes is supported by a ledge inside the core support barrel and by 52 core support columns. The core shroud is attached to the core support plate and limits core bypass flow.

VA – 13-15

6. Core Support Barrel Snubbers

   a. The core support barrel snubbers are attached to the reactor vessel internal surface near the bottom of the core support barrel

      1) Six core support barrel snubbers provide a close fit between the core support barrel and the lower vessel wall. (Tongue-and-groove arrangement. Tongue is located on the vessel).

   b. Snubbers prevent flow-induced vibration and seismically-caused movement of the CSB

   c. Cap screws, secured by locking pins, are used to attach two shims to each CSB stabilizing lug.

   d. The CSB Snubbers are sometimes referenced as CSB Stabilizing Lugs.

7. Core Support Lugs (Core Stops)

   a. Nine core support lugs are attached to the vessel lower wall. The Core Support Lugs are designed to catch the core barrel in the event that the upper core supports failed. The Core Support Lugs contain a small yield pad that is intended to deform and absorb the energy of the falling core.

   b. The Core Support Lugs are sometimes referred to as the Core Stops.

VA - 16

8. Core Shroud

   a. The core shroud follows the perimeter of the core and limits the amounts of coolant bypass flow. The shroud consists of rectangular plates 5/8 inch thick, 145 inches long and of varying widths. The bottom edges of these plates are fastened to the core support plate by use of anchor blocks.

   b. The critical gap between the outside of the peripheral fuel bundles and the shroud plates is maintained by seven tiers of centering plates attached to the shroud plates and centered during initial assembly by adjusting bushings located in the core support barrel. The overall core shroud assembly, including the rectangular plates, the centering plates, and the anchor blocks, is a bolted and lock-welded assembly.

   c. In locations where mechanical connections are used, bolts and pins are designed with respect to shear, binding and bearing stresses. The core shroud assembly is designed with some inherent flexibility to minimize internal stresses at fastener locations while maintaining necessary clearances. Because pressure is equalized across inner and outer shroud faces at both the upper and lower ends of the shroud, differential pressure across the shroud during transients will remain relatively low.

   d. All bolts, block and pins used in assembly of the core shroud are lock-welded and pre-designed to be captured in the event of a fracture.

   e. Holes in core support plate allow some bypass flow upward between core shroud and CSB - minimize thermal stress, eliminate stagnant pockets of coolant.

VA – 17, 18

9. Flow Skirt

   a. The Inconel flow skirt is a perforated (2-1/2 inch diameter holes) right circular cylinder, reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum.

   b. The flow skirt is a cylinder broken into 900 2.5-inch holes that break the large annular wall jet into approximately 900 2.5-inch diameter jets directed toward the center of the reactor vessel.

   c. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is hung by welded attachments from the core stop lugs near the bottom of the pressure vessel and is not attached to the core support barrel.

VA – 19-22

10. Upper Guide Structure

Weight is around 27 tons

   a. *This assembly consists of a flanged grid structure,* 45 control rod shrouds, a fuel bundle alignment plate and a ring shim. The upper guide structure aligns and supports the upper end of the fuel bundles, maintains the control rod channel spacing, prevents fuel bundles from being lifted out of position during a severe accident condition and protects the control rods from the effect of coolant cross flow in the upper plenum. It also supports the incore instrumentation guide tubing. The upper guide structure is handled as one unit during installation and removal.

   b. The upper end of the assembly is a flanged grid structure consisting of a grid array of 18-inch-deep long beams in one direction with 9-inch-deep short beams at 90 degrees to the deeper beams. The grid is encircled by an 18-inch-deep cylinder with a 3-inch-deep flange welded to the cylinder.

c. The periphery of the flange contains four accurately machined and located alignment keyways, equally spaced at 90-degree intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of the control rod shrouds

d. The fuel bundle alignment plate is designed to align the upper ends of the fuel bundles and to support and align the lower ends of the control rod shrouds. Precision machined and located pins attached to the fuel bundle alignment plate align the fuel bundles. The fuel bundle alignment plate also has four equally spaced slots on its outer edge which engage with stellite hard-faced pins protruding out from the core support barrel to prevent lateral motion of the upper guide structure assembly during operation. Since the weight of a fuel bundle under all normal operating conditions is greater than the flow lifting force, it is not necessary for the upper guide structure assembly to hold down the core. However, the assembly does capture the core and would limit upward movement in the event of an accident condition.

11. Control Rod Shrouds

a. The control rod shrouds are of cruciform configuration and extend from about 1 inch above the fuel bundles to about 2 inches above the top of the pressure vessel flange. They enclose the control rods in their fully withdrawn position above the core, thereby protecting them from adverse effects of flow forces.

b. The shrouds consist of 4 formed plates, 0.187 inch thick by approximately 138 inches long, which are welded to 4 end bars to form a cruciform-shaped structure. The shrouds are fitted with support pads at the upper end machined for a bolted and lock-welded attachment to the flanged grid structure.

c. The lower ends of the shrouds are also fitted with support pads machined for a bolted and lock-welded attachment to the fuel bundle alignment plate. The cruciform design provides a stiff section, resulting in low stresses and deflections. In the area of maximum cross flow, the shroud is supported between the flanged grid structure and the fuel bundle alignment plate as a beam with fixed ends.

VA – 23, 24

12. Surveillance Capsules

a. *Surveillance capsules are installed* inside the reactor vessel to measure the long term effects of temperature and radiation on reactor vessel materials.

1) RT_{NDT} and PTS concerns (why we utilize surveillance capsules)

a) The exposure of the reactor vessel walls to fast neutron flux during operation weakens the steel because the crystal structure is disturbed (neutron embrittlement). This weakening causes the reference temperature for nil ductility transition (RT_{NDT}) to shift to a higher value. In general terms, the material becomes less ductile (more susceptible to brittle fracture).

b. Neutron Fluence Surveillance capsules are mounted in the reactor vessel at eight (8) locations. Six locations are on the inside wall of the vessel and two locations are on the outside wall of the core support barrel shell.

c. Surveillance capsules contain the following items:

1) Reactor vessel metal samples.

a) Base metal, weld metal and the heat affected zone metal (base metal affected by the heat of the welding process) samples are included in the capsules.