After further investigation we propose that Question 53 be deleted due to no correct answer.

The original assumption was due to the lowering VI (Instrument Air) pressure the S/G feed reg valves would fail closed resulting in increasing feed pump discharge pressure to the point where the feed pumps would trip on high discharge pressure. While the failure mechanism for the FRV's is correct on a loss of air (closed), the resulting Reactor trip occurs due to low S/G level not feed pump discharge pressure. This was validated on our simulator with a loss of VI ramped in over a period of 5 to 10 minutes which resulted in the reactor tripping on low S/G level when VI press had degraded to approx. 40 psig. While FWP discharge pressure did increase, the action of the FWP control system to maintain the program D/P at set point resulted in a decrease in FWP speed as the FRV's closed. This action delays the increase in FWP discharge pressure. to the point where lowering S/G level will reach trip set point prior to increasing FWP discharge press. In 4 different VI failure scenarios with varying leak sizes and ramp rates, the reactor tripped due to S/G level prior to FWP discharge press reaching trip set point.

The feed pump speed control system is designed to maintain a D/P set point between Main Steam header pressure and Feedwater Header pressure. This setpoint is ramped from 40 psid at 0 % power to 175 psid at 100% power. (see page 5 of this attachment) When a VI system leak is inserted, the resulting lowering VI header press causes the FRV's to slowly close. As they close, FWP discharge press increases. This causes the SM/CF D/P to rapidly increase and was observed to be as high as 400 psid. The FWP control system responds to slow the FWP speed in an attempt to maintain the programmed D/P. In every case except a catastrophic failure where the VI header rapidly depressurizes this action significantly delays the increase in FWP discharge pressure. A catastrophic failure of VI is not consistent with the conditions established in the stem of this question.

One final data point added to the end of this attachment is copy of PIP M87-00208 which documents an actual loss of VI in 1987 where the reactor tripped due to S/G Lo Lo level after approx. 5 min. This is consistent with what we observed during our simulator validations.

Question 53 asks, "WOOTF will be the FIRST to trip the reactor". Lo Lo S/G level is not listed in any of the proposed answers. Based on the information above and the attached documentation, we feel this question should be deleted due to having no correct answer.

Tier #	1
Group #	1
K/A #	065 AA2.08
Importance Rating	2.9

(Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment)

Proposed Question: Common 53

Both Units are at 100% power when:

- MNS experiences a loss of VI.
- VI Header pressure on both units is 30 psig and continues to decrease.

Which ONE (1) of the following will be FIRST to trip the reactor and its basis?

A. OT Delta T trip due to the MSIVs failing closed.

B. High-High SG level trip because the Feedwater Reg Valves fail open.

C. Turbine Trip due to Feed Pump Trip because the Feedwater Reg Valves fail closed.

D. High Pzr Level trip due to the Charging Flow Control Valve failing open with Letdown isolated.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect. If MSIVs fail closed it could cause a trip on LO-LO SG level or on OTDT due to Tave rising. FRVs failing will cause the trip significantly faster than MSIV failure. Plausible because until a recent mod, the MSIVs did fail close on a loss of VI.

B. Incorrect. Feedwater reg valves fail closed, but plausible because this is the reactor trip mechanism if they did fail open.

C. Correct. Reactor trip will occur due to turbine trip. If fedwater reg valves fail closed, Feed Pumps will trip on high discharge pressure, which directly causes a turbine trip.

D. Incorrect. This would cause a reactor trip, but at a significantly later time than feedwater reg valve failure. Charging flow control valve fails open and letdown isolates on loss of air, so PZR level will be rising

Technical AP-22 Background (Attach if not previously

Reference(s)	Document (p22 Rev 12)	provided)
	AP/1/A/5500/22 Enclosure 12 p104 106, Rev 27	(Including version or revision #)
	CF-CF p19 Rev 32	

Proposed references to be provided to applicants during examination: _____

Learning Objective: CF-CF Obj 4 (As available)

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

Question History: Last NRC Exam _____

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

10 CFR Part 55 Content:	55.41	5
	55.43	_____

Comments:
KA is matched because item evaluates the failure mode of air operated equipment that will cause a reactor trip

RFA Concurs 4/24/08

1.0 INTRODUCTION

1.1 Purpose

Objective # 1

Reference Fig. 7.1. *The purpose of the Feedwater Pump Speed Control System is to provide a variable speed demand signal to the main feedwater pump turbines which is programmed to maintain the Feedwater Regulating (Flow Control) Valves in their optimum throttling position.* The system also operates the Feedwater Pumps in conjunction with the Feedwater Control Valves to maintain steam generator water level with maximum efficiency.

1.2 General Description

The Steam Generator Level Control System computes a desired level that is based on reactor power. To maintain this desired level, a control signal is developed. The signal functions to position the Feedwater Regulating Valves, and thus control feed flow, for each of the steam generators. The Main Feed Pump Speed Control System is designed to complement the operation of the Water Level Control system. It computes a desired pump speed that is based on auctioneered HI NC delta T (reactor power). This signal functions to regulate the steam flow to the main feed pump turbines, controlling their speed. Variation of pump speed in this fashion results in:

- Reduced erosion of feedwater regulating valve flow control surfaces.
- Improved feed regulating valve flow control (throttling) characteristics.

Variable speed main feed pumps are favored because they are more economical in large plants. There is less lost energy when the speed is matched with the desired Differential Pressure (Delta P or DP).

Steam header pressure is compared to feed pump discharge pressure generating an Actual DP signal. The DP signal is compared to a programmed setpoint determined from auctioneered HI NC delta T (reactor power) and generates the speed demand signal. Corrections may be added for flow imbalance between the pumps.

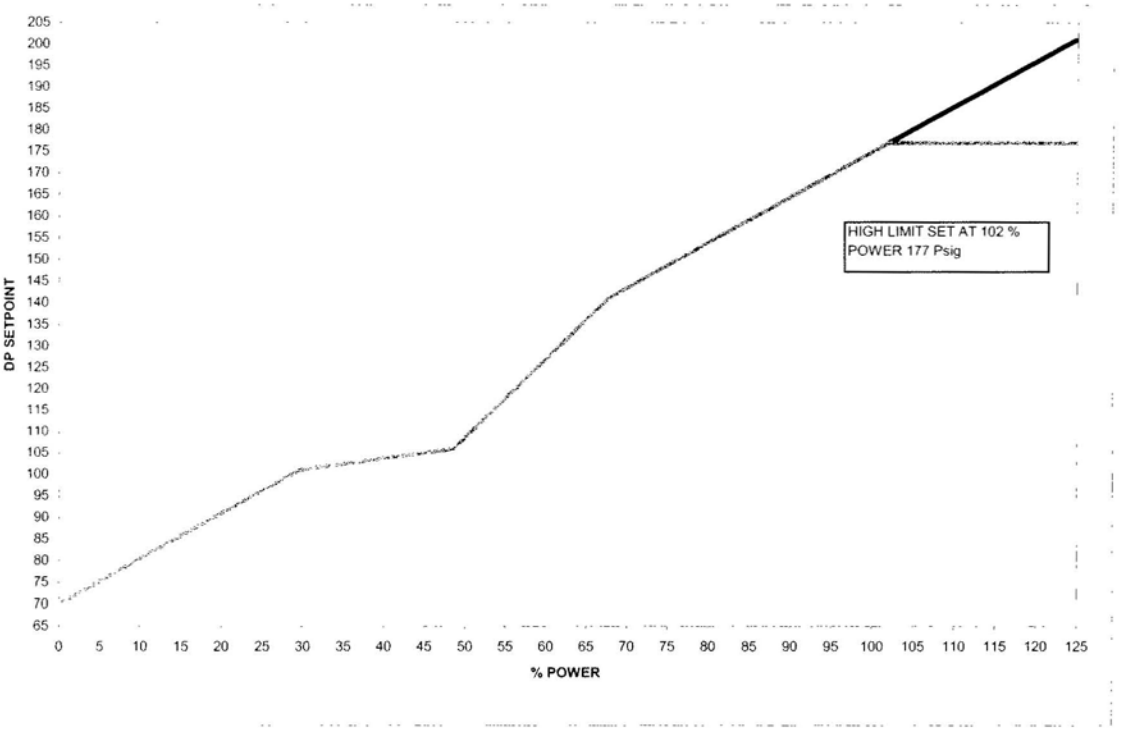
2.0 COMPONENT DESCRIPTION

2.1 General Operation

Objective # 6

The system utilizes three basic control loops to control the feedpump. The feedwater loop, the governor loop, and actuator loop. These loops and their relationship are shown on drawing 4.1. Each of these loops is a proportional plus integral control loop (P+I). Basically these are cascading control loops in that the feedwater control loop output is the input to the governor loop and the output of the governor loop is the input to the actuator loop. This assumes that the controls are in auto. The input for the feedwater loop is the feedwater demand signal which is the programmed delta P for the power level of the reactor as sensed by auctioneered hi NC loop delta T.

7.3 Feedpump ΔP Program w/AMSAC Avoidance (03/17/06)



FEEDPUMP PROGRAM ΔP

70 PSID → 175 PSIG

STEP 13:

PURPOSE:

Prompt the operators to watch S/G levels because the CF control valves fail closed on a loss of VI. If S/G levels can't be controlled, the Operator is directed to trip the reactor.

DISCUSSION:

The CF control valves use 0 – 60# valve operating air. Depending on the nature of the problem with VI and considering line losses, etc., these valves could start failing at 70# or more VI pressure as indicated in the control room. The operating philosophy regarding loss of Main Feedwater at power is to trip the reactor. This will prevent challenging the Lo-Lo S/G automatic reactor trip and will result in better initial conditions at the time of the manual trip. Refer to PIP 2-M-87-0208 where a automatic reactor trip occurred 5 min after loss of offsite power due to loss of VI to the CF valves. If the CF valves were to get to less than 25% open (for 30 sec or more) on 3 out of 4 S/Gs, an AMSAC could also be generated. For most scenarios, it's likely the operator will have manually tripped the reactor prior to this occurring.

REFERENCES:
PIP 2-M-87-0208

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

11. (Continued)

m. Control NC temperature as follows:

__ • Throttle ND flow.

NOTE
- KC to ND Hx flow should be close to flow prior to loss of VI, since it is normally controlled by motor operated valves.
- KC to ND HX flow indications fails low during a loss of VI. Alternate indications are available at the following locations, if needed:
  - 1A: 1KCFT-5670 (aux bldg, 733 +2', west of column MM-54)
  - 1B: 1KCFT-5680 (aux bldg, 733 +4', west side of column JJ-55).

__ • **IF** NC temperature is greater than 200°F, **THEN** maintain KC flow to ND Hx greater than 2000 GPM.

__ • Throttle KC Flow to ND Hx as required.

__ 12. **IF AT ANY TIME VI pressure is less than 70 PSIG, THEN align B Train RN to SNSWP PER Enclosure 7 (Aligning B Train RN to Pond).**

NOTE CF Control Valves will fail closed on low VI pressure, which may result in AMSAC actuation and Lo Lo S/G level.

__ 13. **Check S/G levels - AT PROGRAMMED LEVEL.**

IF S/G levels are going down in an uncontrolled manner, THEN perform the following:

__ a. Trip reactor.

__ b. Continue with this procedure as time allows.

__ c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection).

Objective # 4

The conditions which will result in both main feed pumps tripping:

- **if all three CBPs trip**
- **on a Safety Injection actuation**
- **on S/G Hi-Hi-level (P-14)(83%)**
- **on Hi Hi Dog House level in either inner or outer Dog House (12 inches)**

The conditions which will result in tripping the associated main feed pump:

- **two out of three of its suction pressure switches are low (230 psig)**
- **two out of three of its discharge pressure switches are high (1435 psig)**
- **manual trip**
- **two out of three low bearing oil pressure (7 psig)**
- **two out of three low vacuum (14 inches Hg)**
- **overspeed**
- **fire lockout**
- **thrust bearing wear (0.01 inch axial movement)**

The status of the suction and discharge pressure switches can be determined from the "Condensate 2 of 3 Logic Panel" located in the Turbine Building basement near the BB demineralizer room across for Column 1F24 (refer to Drawing 7.3).

When A(B) CF pump is tripped, solenoid valve 1CFSV0010 (1CFSV0040) is energized which closes A(B) CF Pump piston operated discharge check valve CF1 (CF4). This will prevent flow reversals through the idle pump and prevent damage to the pump and piping due to waterhammer.

The CF pump suction (CM266 and CM272) and discharge (CF2 and CF4) valves are motor operated valves which can be controlled using OPEN/CLOSE pushbuttons from MC-10 (refer to Drawing 7.4). **The A(B) CF pump turbine high and low pressure stop valves can not be raised (opened) until its suction valve CM266 (CM272) and its discharge valve CF2(CF5) are open, as well as the CM Booster Pump recirc valve CM250 and CF System Flush Valve CF124 are closed. In addition CF124 can not be opened unless A CFP suction valve (CM266) and discharge valve (CF 2) are open OR B CFP suction valve (CM272) and discharge valve (CF 5) are open, and both CF pump turbines are tripped.**

Attachment 1a

PIP Serial No:	Action Category:	LER No:	Other Report:
M-87-00208			M87-075-0

Problem Identification

Discovered Time/Date: 10:21 09/16/1987 **Occurred Time/Date:**

Unit(s) Affected:

Unit	Mode	%Power	Unit Status Remarks
2	1		

System(s) Affected:

CF	Feedwater System
EPB	6.9 KV Norm Aux & RCP Swgr Power Sys
ND	Residual Heat Removal System
VI	Instrument Air System

Affected Equipment

Equip ID	Comp ID	Equipment Tag	ECode	Manufacturer

Location of Problem:
Bldg: Column Line: Elev:

Location Remarks:

Method Used to Discover Problem:

Brief Problem Description:
APPROX. 5 MINUTES AFTER LOSS OF OFFSITE POWER TO UNIT 1, UNIT 2 E

Detail Problem Description:
APPROX. 5 MINUTES AFTER LOSS OF OFFSITE POWER TO UNIT 1, UNIT 2 EXPERIENCED A REACTOR TRIP DUE TO S/G-LO LO LEVEL. FEEDWATER R EGULATOR VALVES FAILED CLOSED DUE TO LOSS OF STATION AIR (VI)

Other Units/Components/Systems/Areas Affected(Y,N,U): U

Industry Plants Affected(Y,N,U):

Immediate Corrective Actions:

Immediate Corrective Action Documents / Work Orders:

	Indiv	Team	Group	Date
Problem Identified By:			M-RGC	09/16/1987
Problem Entered By:			M-SRG	09/16/1987

Screening
Action Category: Root Cause performed?

Attachment 1a

Significance Codes:

OEP No:

Other Report Nos: M87-075-0

Event Codes:

88 Convert old pip data to NewPIP relational database

Screening Remarks:

Assignments:

Responsible Groups(s) for Problem Evaluation:	M-SRG	Safety Review
Responsible Group for Present Operability:	M-RGC	Regulatory Compliance
Responsible Group for Report Support Info:	N/A	
Responsible Group for Reportability:	M-RGC	Regulatory Compliance
Responsible Group for Overall PIP Approval:	M-SRG	

Signature Type	Indiv	Team	Group	Date
Screened By:	onfile	RJD4373	M-SRG	09/16/1987

Present Operability

Responsible Group: M-RGC Status: Closed

Sys/Comp Operable? (Y,N,C,E,T): Y

Required Mode:

Comments:

Signature Type	Indiv	Team	Group	Date
Assigned To:			M-RGC	

Reportability

Responsible Group: M-RGC Status: Closed

Problem Reportable(Y,N,E): Y

Reportable Per:

Comments:
LER, N/A

Signature Type	Indiv	Team	Group	Date
Assigned To:			M-RGC	
Due Date:				
Evaluation Assigned To:			M-RGC	09/16/1987

Investigation Report:

Attachment 1a

Responsible Group: Act Date:

Investigator: Group:

Due Date: 10/22/1987
Date Due to VP or Sta. Mgr:
Date Regulatory or Agency Rpt Due:
Date Investigation Report Approved: 10/29/1987

NRC Cause Codes:
A3c3 An Action Was Performed With Insufficient Precision

Report Support Info:

Responsible Group: Status: Closed

Signature Type	Indiv	Team	Group	Date
Assigned To:				

Problem Evaluation

Event	Cause Code	Cause Description	Primary	Causing Groups
88	A3c3	Old Cause Code at McGuire	Yes	V-UNK
88	N/A	Not Applicable	No	M-IAE

Problem Evaluation From: Resp. Group: M-SRG Status: Closed OEDB Checked: No

OEDB Comments:

Remarks Comments:
Due to the electronic up loading of information, any missing PIR information is located in the Master File area.

Signature Type	Indiv	Team	Group	Date
Assigned To:			M-SRG	09/19/1987
Due Date:	10/16/1987			

Corrective Actions

No Corrective Actions for this PIP

Final and Overall PIP Approval

Responsible Group: M-SRG Status: Closed

Signature Type	Indiv	Team	Group	Date
Assigned To:	onfile		M-SRG	
Approved By:	onfile		M-SRG	01/31/1990

Any Supplemental Concurrence Signatures Above Do Not Affect PIP Closure.

Attachment 1a

Microfilm Roll / Frame: /
Closure Document Type **Closure Document No**

Attachments

Generic Applicability

Responsible Group: G-OEA Status: Closed
GO PIP No:

Assessment Remarks:

Signature Type	Indiv	Team	Group	Date
Assigned To:			G-OEA	
Due Date:	01/31/1990			
Approved By:			G-OEA	09/23/1993

Failure Prevention Investigation

No FPI Records for this PIP.

Remarks

No Remarks for this PIP.

Maintenance Rule

No Maintenance Rule Records for this PIP.

End of the Document for PIP No: M-87-208
The status of this PIP is: Closed
The duration of this PIP was: 868 days [Duration = Status Date (01/31/1990) - Discovery Date (09/16/1987)]

(Note: All Corrective Actions were printed in this report.)

Attachment 1b
Common Question 53

The Post-exam Item Analysis indicated that 7 of 12 applicants missed this question.

- **4 chose "A"** – clearly wrong because the MSIVs do not fail closed on loss of VI
- **1 chose "B"** – clearly wrong because the FRVs failed CLOSED, not OPEN on loss of VI
- **2 chose "D"-** plausible because the charging flow valve will fail open and PZR level would eventually reach the reactor trip setpoint. This distracter is wrong because the reactor will already be tripped due to Lo Lo S/G levels prior to the PZR level reaching trip setpoint.

No applicant asked for clarification on this question while taking the exam.

How did this question get through Validation?
Question 53 was revised to its current state around 2-26-08. "C" distracter was validated on the on the simulator where a loss of VI did result in a FWP trip, turbine trip, then reactor trip. However we simulated a catastrophic break on VI, where the pressure dropped rapidly.

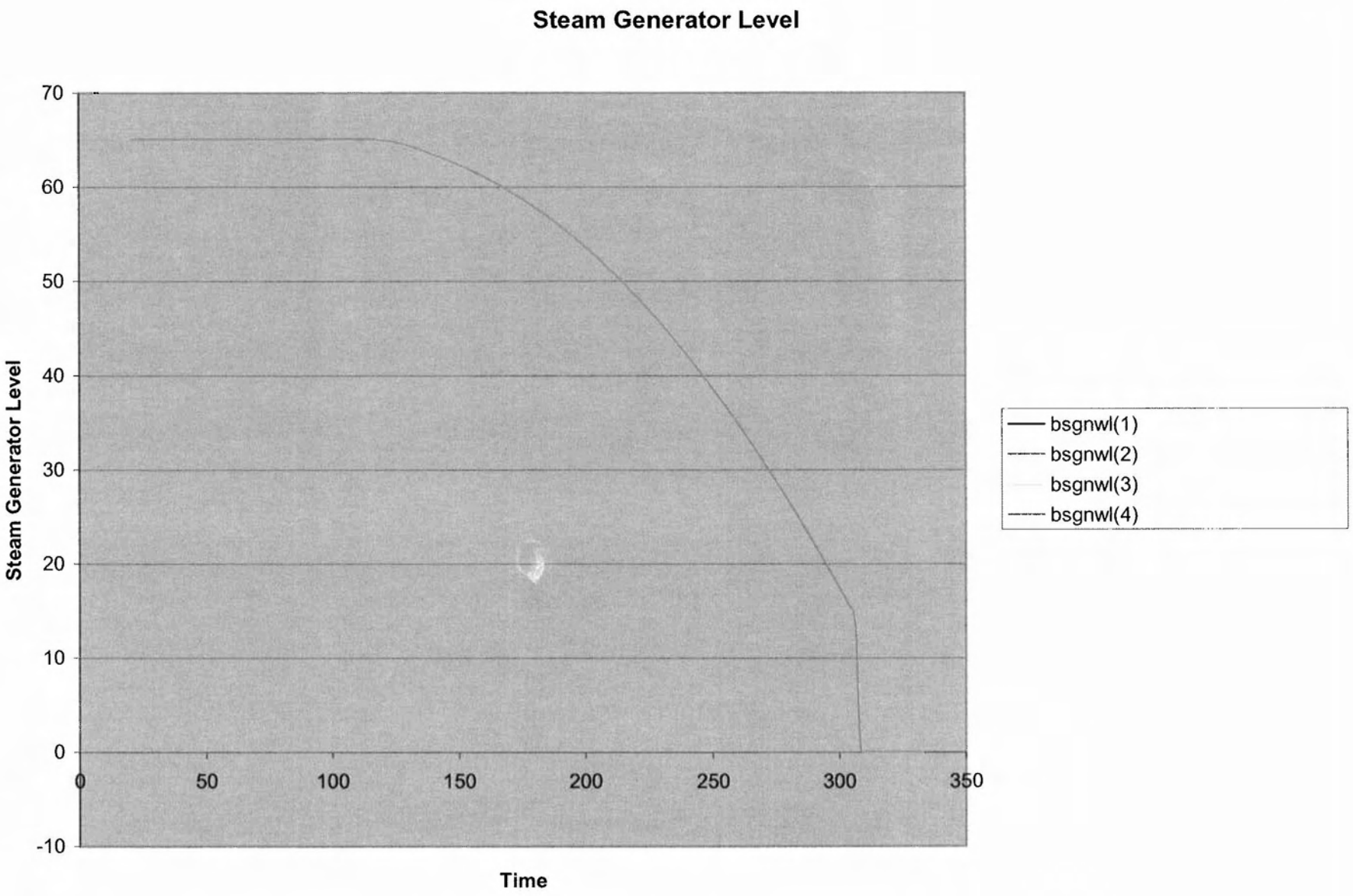
As a result of our post-exam item analysis, we ran the slower VI leak on the simulator to more closely resemble the conditions described in the question stem. That's when we realized that the FWP will not trip as described on page 1.

During validation, six individuals took this version of the question.

- **1 picked A**
- **1 picked B**
- **4 picked D**

The validators admitted memory decay on the failure mode of valves. Therefore, we didn't question if "C" was indeed correct.

We have captured this lesson learned to ensure we validate written question scenarios on the simulator to more closely resemble plant conditions in the question stem.


Steam Generator Level
70
60
50
40
30
20
10
0
-10
0
50
100
150
200
250
300
350
Steam Generator Level
Time
bsgnwl(1)
bsgnwl(2)
bsgnwl(3)
bsgnwl(4)

Tier #	1
Group #	1
K/A #	065 AA2.08
Importance Rating	2.9

Rev. 1 to the package

(Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment)

Proposed Question: Common 53

Both Units are at 100% power when:

- MNS experiences a loss of VI.
- VI Header pressure on both units is 30 psig and continues to decrease.

Which ONE (1) of the following will be FIRST to trip the reactor and its basis?

A. OT Delta T trip due to the MSIVs failing closed.

B. High-High SG level trip because the Feedwater Reg Valves fail open.

C. Turbine Trip due to Feed Pump Trip because the Feedwater Reg Valves fail closed.

D. High Pzr Level trip due to the Charging Flow Control Valve failing open with Letdown isolated.

Proposed Answer: **C**

Explanation (Optional):

A. Incorrect. If MSIVs fail closed it could cause a trip on LO-LO SG level or on OTDT due to Tave rising. FRVs failing will cause the trip significantly faster than MSIV failure. Plausible because until a recent mod, the MSIVs did fail close on a loss of VI.

B. Incorrect. Feedwater reg valves fail closed, but plausible because this is the reactor trip mechanism if they did fail open.

C. Correct. Reactor trip will occur due to turbine trip. If fedwater reg valves fail closed, Feed Pumps will trip on high discharge pressure, which directly causes a turbine trip.

D. Incorrect. This would cause a reactor trip, but at a significantly later time than feedwater reg valve failure. Charging flow control valve fails open and letdown isolates on loss of air, so PZR level will be rising

Technical AP-22 Background (Attach if not previously

Reference(s)	Document (p22 Rev 12)	provided)
	AP/1/A/5500/22 Enclosure 12 p104 106, Rev 27	(Including version or revision #)
	CF-CF p19 Rev 32	

Proposed references to be provided to applicants during examination: _____

Learning Objective:	CF-CF Obj 4		(As available)
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam _____

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

10 CFR Part 55 Content:	55.41	5
	55.43	

Comments:
KA is matched because item evaluates the failure mode of air operated equipment that will cause a reactor trip

RFA Concurs 4/24/08

After further investigation we propose that Question 53 be deleted due to no correct answer.

The original assumption was due to the lowering VI (Instrument Air) pressure the S/G feed reg valves would fail closed resulting in increasing feed pump discharge pressure to the point where the feed pumps would trip on high discharge pressure. While the failure mechanism for the FRV's is correct on a loss of air (closed), the resulting Reactor trip occurs due to low S/G level not feed pump discharge pressure. This was validated on our simulator with a loss of VI ramped in over a period of 5 to 10 minutes which resulted in the reactor tripping on low S/G level when VI press had degraded to approx. 40 psig. While FWP discharge pressure did increase, the action of the FWP control system to maintain the program D/P at set point resulted in a decrease in FWP speed as the FRV's closed. This action delays the increase in FWP discharge pressure. to the point where lowering S/G level will reach trip set point prior to increasing FWP discharge press. In 4 different VI failure scenarios with varying leak sizes and ramp rates, the reactor tripped due to S/G level prior to FWP discharge press reaching trip set point.

The feed pump speed control system is designed to maintain a D/P set point between Main Steam header pressure and Feedwater Header pressure. This setpoint is ramped from 40 psid at 0 % power to 175 psid at 100% power. (see page 5 of this attachment) When a VI system leak is inserted, the resulting lowering VI header press causes the FRV's to slowly close. As they close, FWP discharge press increases. This causes the SM/CF D/P to rapidly increase and was observed to be as high as 400 psid. The FWP control system responds to slow the FWP speed in an attempt to maintain the programmed D/P. In every case except a catastrophic failure where the VI header rapidly depressurizes this action significantly delays the increase in FWP discharge pressure. A catastrophic failure of VI is not consistent with the conditions established in the stem of this question.

One final data point added to the end of this attachment is copy of PIP M87-00208 which documents an actual loss of VI in 1987 where the reactor tripped due to S/G Lo Lo level after approx. 5 min. This is consistent with what we observed during our simulator validations.

Question 53 asks, "WOOTF will be the FIRST to trip the reactor". Lo Lo S/G level is not listed in any of the proposed answers. Based on the information above and the attached documentation, we feel this question should be deleted due to having no correct answer.

Attachment 1b
Common Question 53

The Post-exam Item Analysis indicated that 7 of 12 applicants missed this question.

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- **2 chose "D"-** plausible because the charging flow valve will fail open and PZR level would eventually reach the reactor trip setpoint. This distracter is wrong because the reactor will already be tripped due to Lo Lo S/G levels prior to the PZR level reaching trip setpoint.

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How did this question get through Validation?
Question 53 was revised to its current state around 2-26-08. "C" distracter was validated on the on the simulator where a loss of VI did result in a FWP trip, turbine trip, then reactor trip. However we simulated a catastrophic break on VI, where the pressure dropped rapidly.

As a result of our post-exam item analysis, we ran the slower VI leak on the simulator to more closely resemble the conditions described in the question stem. That's when we realized that the FWP will not trip as described on page 1.

During validation, six individuals took this version of the question.

- **1 picked A**
- **1 picked B**
- **4 picked D**

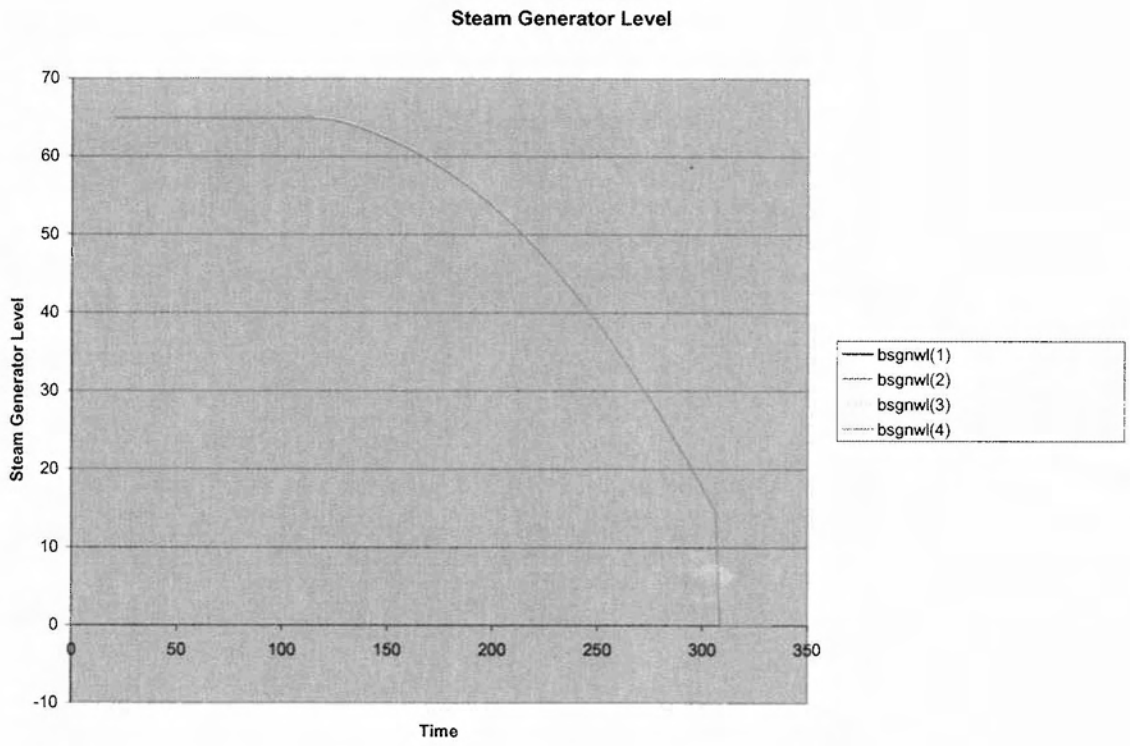
The validators admitted memory decay on the failure mode of valves. Therefore, we didn't question if "C" was indeed correct.

We have captured this lesson learned to ensure we validate written question scenarios on the simulator to more closely resemble plant conditions in the question stem.

Documentation of Post Exam Simulator Data for Common Question # 53

The simulator was set up with a standard 100% MOC snap. A malfunction was inserted consisting of ramped 75% break of the Service Building VI (Instrument Air) header. Over the first 30 sec, VI pressure falls fairly rapidly to a value of approximately 65 PSIG. At this point the FRV demand begins to increase indicating that the actual valve position is closing due to the lower VI header pressure. The result of the FRV's going closed shows that the FW header/Main Steam header D/P begins to increase due to the resulting increase in FWP discharge pressure. This results in a control action by the FWPT Speed control system lower FWPT speed in an attempt to control the D/P at setpoint. These values are shown on the attached graph. At approx. 65 psig the rate of VI pressure decrease slows which would be consistent with the automatic response of the standby VI compressors starting/loading attempting to restore VI pressure. Over the next 3 minutes, VI pressure continues to lower resulting in further closure of the FRV's thereby further reducing feed flow to the S/G's. FWP discharge pressure continues to increase as the FRV's close but the FWPT control system is continuing to decrease the FWPT speed which dampens and slows this effect the to the point where the reduction of feed flow to the S/G's results in a low level trip of the Reactor at 17% level approximately 4.5 minutes into the failure with VI pressure indicating about 30PSIG. Just prior to the trip, FWP discharge pressure peaks at approx. 1400 PSIG which is well below the trip setpoint of 1435 PSIG.

Below I have pasted Excel Spreadsheet trends converted from the Simulator Data Recorder depicting the behavior of S/G level, FWP Discharge pressure, and VI Header pressure during the malfunction described above. The X Axis is showing the Time in seconds. I have also attached an Excel file to this communication with the raw data of the selected parameters and additional trends created from this data.


Steam Generator Level
Steam Generator Level
70
60
50
40
30
20
10
0
-10
0
50
100
150
200
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Time
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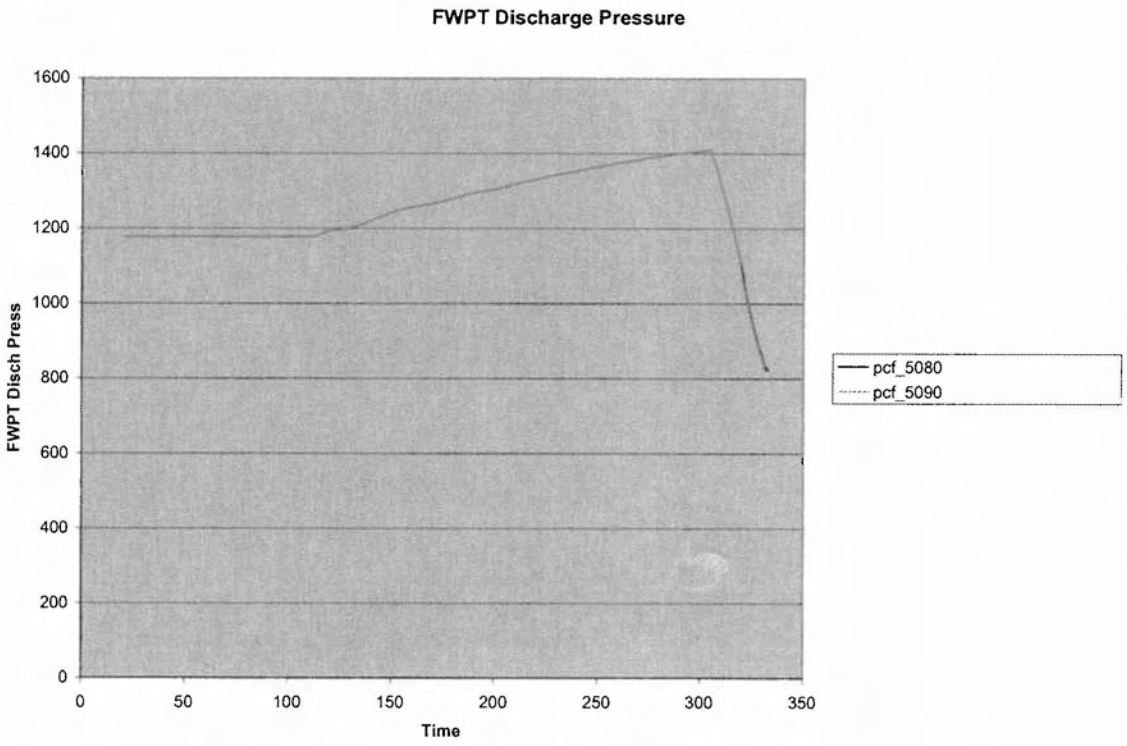
bsgnwl(3)
bsgnwl(4)


FWPT Discharge Pressure
FWPT Disch Press
1600
1400
1200
1000
800
600
400
200
0
0
50
100
150
200
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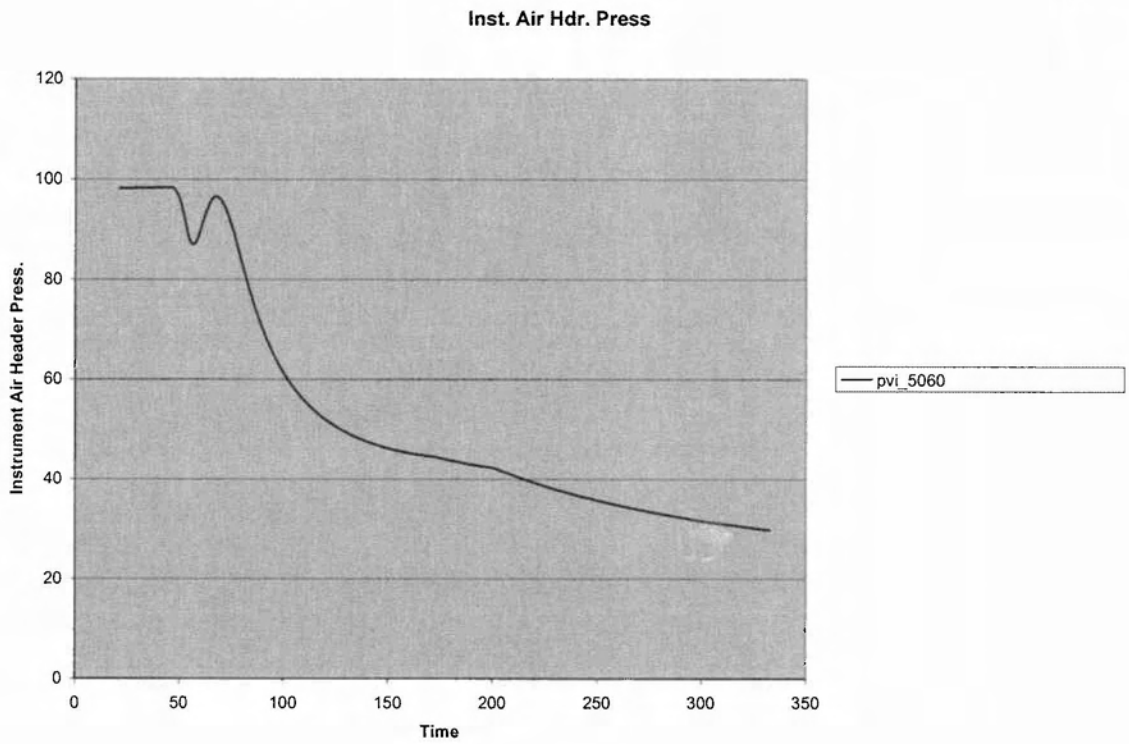
350
Time
pcf_5080
pcf_5090


Inst. Air Hdr. Press
Instrument Air Header Press.
120
100
80
60
40
20
0
0
50
100
150
200
250
300
350
Time
pvi_5060

Attachment 2

SRO Question 77

During the Post Exam Item Analysis we have determined that Question 77 has two correct answers with Distracter "C" being a subset of the Correct Answer "D". Due to this determination we propose that both "C" and 'D" be accepted as correct.

In this question the candidate is asked to determine a leak rate from the KC (Component Cooling) System. The operator is given KC surge tank levels for T=0 and T- +5 min along with a copy of the KC Surge tank level curve from the Data Book. Correctly performed, the leak rate is determined to be approximately 100 GPM. Also in the stem of the question the operator is advised that "Operators have been dispatched to initiate YM makeup to the KC surge tank", but are not told that this makeup has actually been established. The initial level in the surge tank is 6.5 ft. The candidate is then asked to determine "the required action and procedure use required in AP/21, Loss of KC or KC System Leakage".

Per paragraph 7 of Appendix E of NUREG-1021 page 2 of 6 which is part of the briefing given prior to the exam, the candidates are given the following instructions,

"When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise. Finally, answer all questions based on actual plant operation, procedures or the answer choices specifically stated otherwise".

With this direction in mind the candidate can correctly assume that YM makeup is not initially aligned but within a reasonable period of time it will be. (Per the Time Critical Actions for our plant this should occur within 10 min of dispatch) According to the background document for AP-21 pg 10 of 23, "Per engineering, YM makeup should be sufficient to keep up with the FSAR design basis leak of 50 GPM". After determining that a 100 GPM leak exists, even with YM make up aligned the candidate would reasonably come to the conclusion that KC surge tank level would continue to decrease. Procedurally we would align the Nuclear Service Water assured makeup if YM was not sufficient to keep up with the leak but the candidates are told to not assume any operator actions not stated, therefore it is reasonable for the candidate to assume that this has not and will not happen unless otherwise stated.

Procedural guidance per AP-21 directs that once it has been determined that a leak exists operators be dispatched early to initiate YM make up per Step 11. The procedure then checks surge tank level greater than 3 ft which for the scenario given it should be for the first pass thru this step. The operator is then directed to check the "sum of both trains' KC surge tank level drops LESS THAN OR EQUAL TO 0.10 FT/MIN". If true, which for the leak rate calculated, it would be, the crew is directed to Step 20 which will Isolate

"A" KC train from "B" KC train. This action is consistent with the actions stated in the Correct answer "D" making it a valid correct answer. Once again, assuming no operator action has taken place, the surge tank levels will continue to decrease until they reach a level of 2 ft. At this point the "foldout" page would direct the operator to isolate the affected train PER Enc. 2. This action is stated in Distracter "C" and since the stem does not refer to the "First" or "Next" action taken, this action is actually a subset of "D" and therefore a correct answer.

Based on the discussion above and the attached references, we recommend that both "C" and "D" be accepted as correct answers.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	(026) G2.1.25	
	Importance Rating		4.2

Proposed Question: SRO 77

Initial conditions:

Time = 0 minutes

- Unit 1 is at 100% power.
- "A" Train KC pumps are running.
- Operators have been dispatched to initiate YM makeup to the KC Surge Tank.
- "A" KC Surge Tank level is 6.5 ft.
- "B" KC Surge Tank level is 6.5 ft.

Current conditions:

Time = 5 minutes

- "A" KC Surge Tank level is 5.6 feet
- "B" KC Surge Tank level is 6.4 feet.

Which ONE (1) of the following describes (1) the approximate KC system net leak rate, and (2) the required action and procedure use required in AP/21, Loss of KC or KC System Leakage?

(Reference Provided)

A. (1) 50 GPM
(2) Isolate KC Non-Essential Headers in accordance with Enclosure 2.

B. (1) 50 GPM
(2) Isolate "A" KC train from "B" KC train.

C. (1) 100 GPM
(2) Isolate KC Non-Essential Headers in accordance with Enclosure 2.

D. (1) 100 GPM
(2) Isolate "A" KC train from "B" KC train.

Proposed Answer: D

Explanation (Optional):

D is correct per conditions. Applicant must interpret curve and determine the leak rate indications based on level decreases

A incorrect because leak rate is wrong (1/2 of actual, as interpreted by curve.) Also, action is incorrect, as procedure will direct splitting trains for indication shown

B incorrect because leak rate is incorrect. Plausible because action is correct

C incorrect because procedure use is incorrect. Approximately 0.1 feet/minute, perform step 20 to split trains

Technical Reference(s):	AP/21 Step 20 Rev 9	(Attach if not previously provided)
	AP/21 Basis Document Rev 3	

Proposed references to be provided to applicants during examination:	OP/1/A/6100/22 Enclosure 4.3 Curve 7.31

Learning Objective:	None	(As available)

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

Question History:	Last NRC Exam	Modified from 2007 NRC exam 78

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

KA matched because use of a curve is required and interpretation of that curve is required to determine KC (CCW) leak rate. SRO level because assessment of

conditions based on available indications, and selection of procedures (attachments) is required

RFA Concurs 4/17/08

3. For an initial examination, the nominal time limit for completing the examination is 6 hours for the RO exam; 3 hours for the 25-question, SRO-only exam; 8 hours for the combined RO/SRO exam; and 4 hours for the SRO exam limited to fuel handling. Notify the proctor if you need more time.

   For a requalification examination, the time limit for completing both sections of the examination is 3 hours. If both sections are administered in the simulator during a single 3-hour period, you may return to a section of the examination that you already completed or retain both sections of the examination until the allotted time has expired.

4. You may bring pens, pencils, and calculators into the examination room; however, programable memories must be erased. Use black ink to ensure legible copies; dark pencil should be used only if necessary to facilitate machine grading.

5. Print your name in the blank provided on the examination cover sheet ***and*** the answer sheet. You may be asked to provide the examiner with some form of positive identification.

6. Mark your answers on the answer sheet provided, and do not leave any question blank. Use only the paper provided, and do not write on the back side of the pages. If you are using ink and decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change. If you are recording your answers on a machine-gradable form that offers more than four answer choices (e.g., "a" through "e"), be careful to mark the correct column.

7. If you have any questions concerning the intent or the initial conditions of a question, do *not* hesitate to ask them before answering the question. Note that questions asked during the examination are taken into consideration during the grading process and when reviewing applicant appeals. Ask questions of the NRC examiner or the designated facility instructor *only*. A dictionary is available if you need it.

   When answering a question, do *not* make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise. Finally, answer all questions based on actual plant operation, procedures, and references. If you believe that the answer would be different based on simulator operation or training references, you should answer the question based on the *actual plant*.

8. Restroom trips are permitted, but only one applicant at a time will be allowed to leave. Avoid all contact with anyone outside the examination room to eliminate even the appearance or possibility of cheating.

17. **Initiating makeup to the KC surge tank or isolate KC header leak**

Expectation:
Operator locally initiates makeup within 10 minutes of dispatch using either YM or RN, or gets leak isolated prior to emptying surge tank for design basis leak of 50gpm.

Follow-up planned:
None.

References:
FSAR 9.2.4.3
PIP M98-3618
PIP M99-3778
AP/1/A/5500/21 (Loss of KC or KC System Leakage)

Comments:
The FSAR required time to isolate a KC leak is 30 minutes to prevent emptying KC surge tank for a 50 gpm leak. If makeup can stabilize surge tank level, operation can continue with header in service until individual component can be isolated.

A 10 minute local task to initiate makeup is based on: 1) Time to dispatch an operator in AP/21, and 2) Ensure margin to prevent reaching 2 ft in surge tank (for 50 gpm leak). At 2 ft, AP/21 requires isolating KC aux bldg non-essential header. This action has been validated and times can be met.

18. **Isolate letdown for a letdown header break:**

Expectation:
The FSAR assumes isolation of NV letdown within 30 minutes of a complete severance of a letdown header in the aux bldg. Operations expectation is that leak will be isolated as soon as it is identified.

Follow-up planned:
None.

References:
FSAR 15.6.2.2
AP/1/A/5500/10 (NC System Leakage Within the Capacity of Both NV Pumps)

Comments:
The limiting case in FSAR is a complete severance, at rated power conditions, of the 3 inch letdown line just outside containment, between NV-7B and the letdown heat exchanger.

If KC surge tank level is greater than 3 ft, time is given for the operators to attempt to initiate makeup and check results to see if the surge tank level can be maintained. Per engineering, YM makeup should be sufficient to keep up with the FSAR design basis leak of 50 GPM. Operators should be able to initiate makeup prior to reaching 2 ft in the surge tanks (assuming makeup is initiated when KC lo level alarms at 4.5 ft). If the trains are cross-tied, allowing leaving the cross-ties open doubles the volume (and time) to initiate makeup. Note that for larger leaks, or if level reaches 2 ft, the cross-ties will be closed to protect the other train. If makeup is initiated and level stabilizes, operator actions are greatly simplified.

STEP 16 NOTES:

PURPOSE:

Give operator information for determining leak rate.

DISCUSSION:

0.1ft/min level drop is equivalent to the design basis leak (50 gpm) that YM makeup should be able to keep up with.

STEP 16:

PURPOSE:

Procedure flow path controlling step.

DISCUSSION:

If leak is greater than design basis leak, operators need to find the leak and isolate it.

INTRODUCTION

This procedure gives guidance on the actions required in the event of a loss of KC, or leakage on the KC System.

OVERVIEW OF AP-21

AP-21 is written for scenarios involving a loss of KC (like KC pump trips, etc.) and for scenarios involving KC leakage. For either case, the AP first addresses some immediate concerns, like isolating letdown if no KC pumps, making plant page, and securing any dilution.

For scenarios involving a loss of KC not due to leakage, the AP quickly attempts starting the standby KC train (Step 9). It is time critical to restore KC cooling to NC pumps (approx 10 minutes). If cooling is not restored, NC pumps and a reactor trip will be required.

The AP next addresses scenarios involving KC leakage. If KC leakage is small (within design basis of 50 gpm) when makeup is initiated, it should keep up with the leak and avoid the process of header isolations. If the leak is too big for makeup, then the AP goes through a process of header isolations in a preferred order. In any case, the AP ends up with an essential header in operation, and intact non-ess headers in service.

ENTRY CONDITIONS

This procedure can be entered any time the listed symptoms are encountered.

1. **KC header isolation criteria:**

   - **IF** KC surge tank level goes below 2 ft due to KC system leak, **THEN** immediately isolate affected train **PER** Enclosure 2 (Isolation of KC Non-essential Headers).

2. **NC pump trip criteria:**

   - **IF** NC pump motor bearing temperature reaches 195°F, **THEN** perform the following:

     a. Trip the reactor.
     b. **WHEN** reactor is tripped, **THEN** trip all NC pumps.
     c. **GO TO** EP/1/A/5000/E-0 (Reactor Trip or Safety Injection), while continuing in this procedure as time and conditions allow.

3. **ND pump trip and flow isolation criteria (Applies if ND aligned for RHR):**

   - **IF** KC cooling lost to either ND train's HX, **AND** NC temperature is greater than 150°F, **THEN** perform the following on train of ND that lost KC flow to its ND HX:

     a. Stop associated ND pump.

     b. **IF** 1A ND HX lost KC flow, **THEN** close:

        - 1ND-33 (A ND Hx Bypass)
        - 1ND-32 (A ND Hx To Letdown Hx).

     c. **IF** 1B ND HX lost KC flow, **THEN** close:

        - 1ND-18 (B ND Hx Bypass)
        - 1ND-17 (B ND Hx To Letdown Hx).

     d. **IF** both ND pumps off **THEN REFER TO** AP/1/A/5500/19 (Loss of ND or ND System Leak).

4. **KC pump trip criteria:**

   - **IF** KC surge tank level goes below .5 ft and valid, **THEN**:

     a. Trip affected pumps.

     b. Isolate affected train **PER** Enclosure 2 (Isolation of KC Non-essential Headers).

5. **VCT high temperature:**

   - **IF** "VCT HI TEMP" alarm (1AD-7, D-1) is received, **THEN REFER TO** Enclosure 6 (VCT High Temperature Actions).