Alabama Power Company
J. M. Farley Nuclear Plant
P. O. Drawer 470
Ashford, Alabama 36312
Telephone 205 899-5156

ENCLOSURE 3

Alabama Power
the southern electric system

FNP-89-0541-TRN
October 27, 1989

The Regional Administrator, Region II
Nuclear Regulatory Commission
101 Marietta St., N.W.
Atlanta, GA 30323

ATTN: Mr. John F. Munro, Chief
Operator Licensing Branch

Enclosed are Alabama Power Company's comments concerning the written examinations for reactor operator and senior reactor operator given at Farley Nuclear Plant on October 23, 1989.

All of the questions addressed appear on both the SRO exam and the RO exam; both numbers have been placed at the top of the question.

The pre-exam review conducted in Atlanta appears to have been very beneficial in producing a high quality test.

The courteous and professional manner which your staff displayed in preparing and administering this exam is appreciated.

For further clarification or discussion of these comments, please contact Mr. Lee Williams at (205) 899-5156, extension 6106.

D. N. Morey, III
General Manager - Nuclear Plant

DNM,III/LSW/RJV:mjk

Enclosures

9002260155 891027
PDR ADOCK 05000348
V PDC

QUESTION RO 2.03/SRO 5.05 (1.00)

The refueling cavity seal ring has failed, causing level in the refueling cavity to decrease rapidly. A fuel assembly is in the process of being withdrawn from the containment upender. Which of the following is the appropriate action to be taken?

a. Verify both containment sump pumps are operating and that the liquid waste processing system is capable of receiving discharge from the sump pumps.

b. Run one RHR pump in cooldown alignment and align the other RHR pump to the RWST.

c. Immediately stop all RHR pumps.

d. Place the fuel assembly being moved into the core.

ANSWER 2.03/5.05 (1.00)

(b)

Reference:

FNP Question bank no. 052521HT1001R
000036G010 ...(KA'S)

COMMENT

This question was originally written as an open reference question. It is based on FNP-1-AOP-30.0 (copy attached). Choice b. is based on subsequent action steps 5.7 and 5.7.1. An operator with procedure in hand would use those steps of the subsequent actions to combat the casualty in progress, making choice b. a correct answer. Subsequent actions are not required to be memorized, so an operator without the procedure may not recognize this as a procedure requirement.

Choice d. is an allowable action per immediate action step 4.1. It is not the most logical action to take with the fuel being removed from the upender, but it is the only response listed that is an immediate operator action. Unlike subsequent actions, immediate actions are required to be memorized.

Based on the above information, there are two correct answers, choices b. and d.

RECOMMENDATION

Based on two correct answers being available, the question should be deleted from the exam.

For future use, the question should be modified such that only one correct response is available.

FARLEY NUCLEAR PLANT
UNIT 1
ABNORMAL OPERATING PROCEDURE AOP-30.0

REFUELING ACCIDENT

1.0 Purpose

This procedure provides the Symptoms, Automatic Actions, Immediate Operator Actions and Subsequent Operator Actions for a refueling accident.

2.0 Symptoms

2.1 Damaged fuel assembly and high radiation on radiation monitor [R-2, R-5, R-11, R-12, R-24A(B), R-25A(B)].

2.2 Rapid drop in refueling cavity level during refueling.

3.0 Automatic Actions

None

4.0 Immediate Operator Actions

___ 4.1 IF conditions (i.e. radiation levels or rate of cavity level decrease) permit as determined by the SRO in charge of fuel handling, THEN attempt to return any fuel being moved to the reactor vessel or to the upender and lower the upender to its horizontal position.

___ 4.2 Evacuate Containment or Spent Fuel area depending on area where accident has occurred (evacuate Containment and Spent Fuel area if SG nozzle dam failure suspected) and close all access doors.

5.0 Subsequent Operator Actions

___ 5.1 Consider actuating the plant emergency alarm and announce over the public address system for all personnel to report to their designated assembly areas.

___ 5.2 IF fuel damage has occurred in Containment, THEN stop Containment Purge and Mini Purge supply and exhaust fans and close dampers, 1-CP-HV-3196, 3197, 3198 A,B,C,D, 2867 C&D and 2866 C&D.

___ 5.3 IF failure of a Steam Generator primary nozzle dam, Refueling Cavity seal ring failure OR other major Refueling Cavity leakage is suspected, THEN place control switches (BOP) for Containment Sump Pumps in "PULL TO LOCK".

5.4 IF fuel damage has occurred in Containment, THEN perform the following:

___ 5.4.1 Suspend all fuel handling operations in the Spent Fuel Building.

___ 5.4.2 Withdraw the fuel transfer cart to the Spent Fuel Building if the damaged fuel does not interfere with cart movement.

___ 5.4.3 Close the fuel transfer tube gate valve.

___ 5.4.4 Monitor R-2, 11 and 12 for indication of radiation increase to aid in determination of the magnitude of fuel damage.

5.5 IF fuel damage has occurred in the Spent Fuel Building, THEN perform the following:

___ 5.5.1 Close all fuel handling hatches into the Spent Fuel area.

___ 5.5.2 Suspend all fuel handling operations in Containment.

___ 5.5.3 Monitor R-5 and R-25 (A and B) for indication of radiation increase to aid in determination of the magnitude of the fuel damage.

___ 5.6 Initiate FNP-0-EIP-12, Alert, as necessary.

___ 5.7 IF Refueling Cavity leakage is suspected due to possible Refueling Cavity seal ring failure OR to Steam Generator primary nozzle dam failure, THEN verify only one RHR Pump running with its suction lined up to the RCS loops AND closely monitor running pump for evidence of cavitation.

****

CAUTION

Monitor RWST level closely and provide makeup to it as required. Monitor running RHR Pumps for evidence of cavitation and secure RHR Pumps if necessary.

****

___ 5.7.1 Line up one of the RHR Pumps to fill the Refueling Cavity from the RWST using the guidance of FNP-1-SOP-7.0.

___ 5.7.2 IF required, THEN line up a Charging Pump OR the Refueling Water Purification Pump to supply additional makeup water.

___ 5.7.3 Suspend all fuel handling operations in the Spent Fuel Building.

___ 5.7.4 IF conditions (i.e. radiation levels or rate of canal level decrease) permit, as determined by the SRO in charge of fuel handling, THEN attempt to transfer any Fuel Assembly which may be in the upender to the Spent Fuel Pool.

___ 5.7.5 Withdraw the fuel transfer cart to the Spent Fuel Building.

___ 5.7.6 Close the fuel transfer tube gate valve.

___ 5.7.7 IF required, THEN supply makeup to the RWST.

___ 5.8 Initiate FNP-0-EIP-8, Notification Procedure.

___ 5.9 Initiate FNP-0-EIP-14, Re-Entry Procedures.

___ 5.10 Sample and analyze the Refueling Cavity water and Spent Fuel Pool water. IF required, THEN operate the SFPC Purification System per FNP-1-SOP-54.0.

QUESTION RO 2.12/SRO 5.15 (1.00)

Using the attached Table 3 from FRP-I.3, Response to Voids in Reactor Vessel, determine the size in, % pressurizer level, of the entire void if initial RCS pressure was 450 psig and was raised to 550 psig. During the pressure increase, PZR level dropped from 45% to 31%.

a. 14%

b. 65%

c. 79%

d. 84.5%

ANSWER 2.12/5.15 (1.00)

(c)

Reference:

FNP question bank no. 052302F013
FRP-I.3
000074A206 ...(KA'S)

COMMENT

In order to correctly answer this question, the operator must know whether to use initial RCS pressure (450 psig) or final RCS pressure (550) to enter Table 3 that was provided in the test (attached). The instructions for using final RCS pressure to enter the table comes from FNP-1-FRP-I.3 step 13.5.2 (attached). This step was not provided with the exam. No instructions in the stem of the question were provided. This is not a procedure we would expect an operator to memorize.

RECOMMENDATION

Based on the lack of instructions for using the table, this question should be deleted from the exam.

For future use, portions of step 13 of FRP-I.3 should be provided.

Step	Action/Expected Response	Response NOT Obtained
13	Determine minimum allowable pressurizer level during vent.	

****

CAUTION

Pressurizer level must be maintained greater than or equal to 20% {50%}.
IF pressurizer level begins to fall below 20% {50%} during an evolution, THEN stop the evolution.

****

___ 13.1 Verify charging flow control in manual and record one pressurizer level protection channel (LI-459A, 460 or 461) in Table 1 (below).

CHG
FLOW
FK-122
___ controller in manual

Step	Action/Expected Response	Response NOT Obtained

NOTE

During the following RCS pressurization, pressurizer level change is used as an indication of void size. Therefore, charging and letdown flows should NOT be adjusted during the evolution.

___ 13.2 Stop pressurizer spray and turn on pressurizer heaters to raise RCS pressure by 100 psi unless pressurizer level drops to 20% (50%).

__PI-402A (or 403A)

SPR VLV
PVC-444C (D)
__PK-444C controller adjusted

__PK-444D controller adjusted

PRZR
HTR BKUP (VARIABLE)
GRP
__1A on

__1B on

__1C on

__1D on

__1E on

___ 13.3 Record final RCS pressure (PI-402A or 403A), pressurizer level (using same channel recorded above) and complete the Δ pressurizer level in Table 1 (below).

TABLE 1

Pressurizer Level			RCS Pressure
Initial (Step 13.1)	Final (Step 13.3)	Δ Level (Step 13.3)	Final (Step 13.3)
_____% -	_____% =	_____%	_____psig

Step	Action/Expected Response	Response NOT Obtained

****

CAUTION

IF RCS pressure exceeded 2000 psig, THEN reblocking of low pressurizer pressure SI should be performed when permissible.

****

___ 13.4 Restore RCS pressure and pressurizer level to values prior to raising RCS pressure.

CHG
FLOW
__ FK-122 controller manually adjusted as required

SPR VLV
PVC-444C (D)
__ PK-444C controller manually adjusted

__ PK-444D controller manually adjusted

PRZR
HTR BKUP (VARIABLE)
__ GRP 1A, B, C, D or E turned on or off as required

RCS
PRZR AUX SPR
__ CVC-HV-8145 used if required

Step	Action/Expected Response	Response NOT Obtained
___ 13.5	IF pressurizer level remained greater than 20% (50%), THEN obtain minimum allowable pressurizer level during reactor head vent as follows.	13.5 The minimum allowable pressurizer level during the reactor head vent is 20% (50%), proceed to Step 14.
___ 13.5.1	Record existing pressurizer level in Table 2 (below).	
___ 13.5.2	Using the Δ Pressurizer level and final RCS pressure from Table 1, obtain the pressurizer level change required to remove entire void from Table 3 (attached).	
___ 13.5.3	Subtract the required level change from the existing pressurizer level and record in Table 2 (below).	
___ 13.5.4	IF calculated level is less than 20% (50%), THEN minimum allowable level is 20% (50%), otherwise use calculated value.	

TABLE 2

Pressurizer Level

Existing (Step 13.5.1)	Δ Level Required (Step 13.5.2)	Min Level Calculated (Step 13.5.3)
_____% -	_____% =	_____%

TABLE 3
CHANGE IN PRESSURIZER LEVEL REQUIRED TO REMOVE ENTIRE VOID
(SHEET 1 FOR PRZR LEVEL CHANGES 1% - 10%)

OBSERVED CHANGE IN PRESSURIZER LEVEL (%)

		1	2	3	4	5	6	7	8	9	10
	100	1.147	2.294	3.441	4.588	5.735	6.882	8.029	9.176	10.323	11.47
R	200	2.147	4.294	6.441	8.588	10.735	12.882	15.029	17.176	19.323	21.47
C	300	3.147	6.294	9.441	12.588	15.735	18.882	22.029	25.176	28.323	31.47
S	400	4.147	8.294	12.441	16.588	20.735	24.882	29.029	33.176	37.323	41.47
	500	5.147	10.294	15.441	20.588	25.735	30.882	36.029	41.176	46.323	51.47
P	600	6.147	12.294	18.441	24.588	30.735	36.882	43.029	49.176	55.323	61.47
R	700	7.147	14.294	21.441	28.588	35.735	42.882	50.029	57.176	64.323	71.47
E	800	8.147	16.294	24.441	32.588	40.735	48.882	57.029	65.176	73.323	81.47
S	900	9.147	18.294	27.441	36.588	45.735	54.882	64.029	73.176	82.323	91.47
S	1000	10.147	20.294	30.441	40.588	50.735	60.882	71.029	81.176	91.323	101.47
U	1100	11.147	22.294	33.441	44.588	55.735	66.882	78.029	89.176	100.323	111.47
R	1200	12.147	24.294	36.441	48.588	60.735	72.882	85.029	97.176	109.323	121.47
E	1300	13.147	26.294	39.441	52.588	65.735	78.882	92.029	105.176	118.323	131.47
	1400	14.147	28.294	42.441	56.588	70.735	84.882	99.029	113.176	127.323	141.47
P	1500	15.147	30.294	45.441	60.588	75.735	90.882	106.029	121.176	136.323	151.47
S	1600	16.147	32.294	48.441	64.588	80.735	96.882	113.029	129.176	145.323	161.47
I	1700	17.147	34.294	51.441	68.588	85.735	102.882	120.029	137.176	154.323	171.47
G	1800	18.147	36.294	54.441	72.588	90.735	108.882	127.029	145.176	163.323	181.47
	1900	19.147	38.294	57.441	76.588	95.735	114.882	134.029	153.176	172.323	191.47
	2000	20.147	40.294	60.441	80.588	100.735	120.882	141.029	161.176	181.323	201.47

TABLE 3
CHANGE IN PRESSURIZER LEVEL REQUIRED TO REMOVE ENTIRE VOID
(SHEET 2 FOR HIGH LEVEL CHANGES 11% - 20%)

OBSERVED CHANGE IN PRESSURIZER LEVEL (%)

		11	12	13	14	15	16	17	18	19	20
	100	12.617	13.764	14.911	16.058	17.205	18.352	19.499	20.646	21.793	22.94
R	200	23.617	25.764	27.911	30.058	32.205	34.352	36.499	38.646	40.793	42.94
C	300	34.617	37.764	40.911	44.058	47.205	50.352	53.499	56.646	59.793	62.94
S	400	45.617	49.764	53.911	58.058	62.205	66.352	70.499	74.646	78.793	82.94
	500	56.617	61.764	66.911	72.058	77.205	82.352	87.499	92.646	97.793	102.94
P	600	67.617	73.764	79.911	86.058	92.205	98.352	104.499	110.646	116.793	122.94
R	700	78.617	85.764	92.911	100.058	107.205	114.352	121.499	128.646	135.793	142.94
E	800	89.617	97.764	105.911	114.058	122.205	130.352	138.499	146.646	154.793	162.94
S	900	100.617	109.764	118.911	128.058	137.205	146.352	155.499	164.646	173.793	182.94
S	1000	111.617	121.764	131.911	142.058	152.205	162.352	172.499	182.646	192.793	202.94
U	1100	122.617	133.764	144.911	156.058	167.205	178.352	189.499	200.646	211.793	222.94
R	1200	133.617	145.764	157.911	170.058	182.205	194.352	206.499	218.646	230.793	242.94
E	1300	144.617	157.764	170.911	184.058	197.205	210.352	223.499	236.646	249.793	262.94
	1400	155.617	169.764	183.911	198.058	212.205	226.352	240.499	254.646	268.793	282.94
P	1500	166.617	181.764	196.911	212.058	227.205	242.352	257.499	272.646	287.793	302.94
S	1600	177.617	193.764	209.911	226.058	242.205	258.352	274.499	290.646	306.793	322.94
I	1700	188.617	205.764	222.911	240.058	257.205	274.352	291.499	308.646	325.793	342.94
G	1800	199.617	217.764	235.911	254.058	272.205	290.352	308.499	326.646	344.793	362.94
	1900	210.617	229.764	248.911	268.058	287.205	306.352	325.499	344.646	363.793	382.94
	2000	221.617	241.764	261.911	282.058	302.205	322.352	342.499	362.646	382.793	402.94

QUESTION RO 2.15/SRO 5.18 (1.00)

Intermediate Range channel N-35 is severly UNDERcompensated. After a reactor trip from 100% power, the source range channels:

a. will automatically reenergize at P-6, normally 15 to 20 minutes after the trip.

b. must be reenergized by the operator. They normally would have auto energized 15 to 20 minutes after the trip.

c. will automatically reenergize at P-6, normally 20 to 25 minutes after the trip.

d. must be reenergized by the operator. They normally would have auto energized 20 to 25 minutes after the trip.

ANSWER 2.15/5.18 (1.00)

(b)

Reference:

Lesson Plan OPS-52201D EO-2
000032A204 ...(KA's)

QUESTION RO 3.01/SRO 6.01 (1.00)

Which one of the following is an indication of an Intermediate Range channel being undercompensated?

a. Automatic energizing of the source range when the first intermediate range channel drops below the P-6 setpoint.

b. Actual neutron level falls into the source range without indication.

c. Indication remains above 10E5 cps in the source range on both channels.

d. Automatic energizing of the source range when the P-10 interlock was cleared on both channels.

ANSWER 3.01/6.01 (1.00)

(b)

Reference:

FNP Exam Questions from the 1987 Licensed Operator Retraining Program
FNP 1987 License Retraining Objectives Cycle 1, OPS-52201D, item 3 (3.9/3.9)
015000A303 015000K401 ...(KA's)

COMMENT

These two questions rely on the same knowledge in order to be answered correctly. This would penalize the candidate twice if he or she did not know how an intermediate range instrument would respond if under-compensated.

RECOMMENDATION

The use of multiple questions testing the same knowledge should be avoided in the future.

QUESTION RO 2.20/SRO 5.22 (1.00)

With the pressurizer level control selector switch in position "I/II", a failure causes the following plant events: (assume no operator actions)

1. Charging flow reduced to minimum
2. Letdown secured and heaters off
3. Level increase until high presurizer level trip

Using the attached drawing, which of the following failures occurred?

a. LT-459 failed low
b. LT-460 failed low
c. LT-461 failed low
d. LT-459 failed high
e. LT-460 failed high
f. LT-461 failed high

ANSWER 2.20/5.22 (1.00)

(d)

Reference:

FNP question bank no. 052201H09015
000028A202 ...(KA'S)

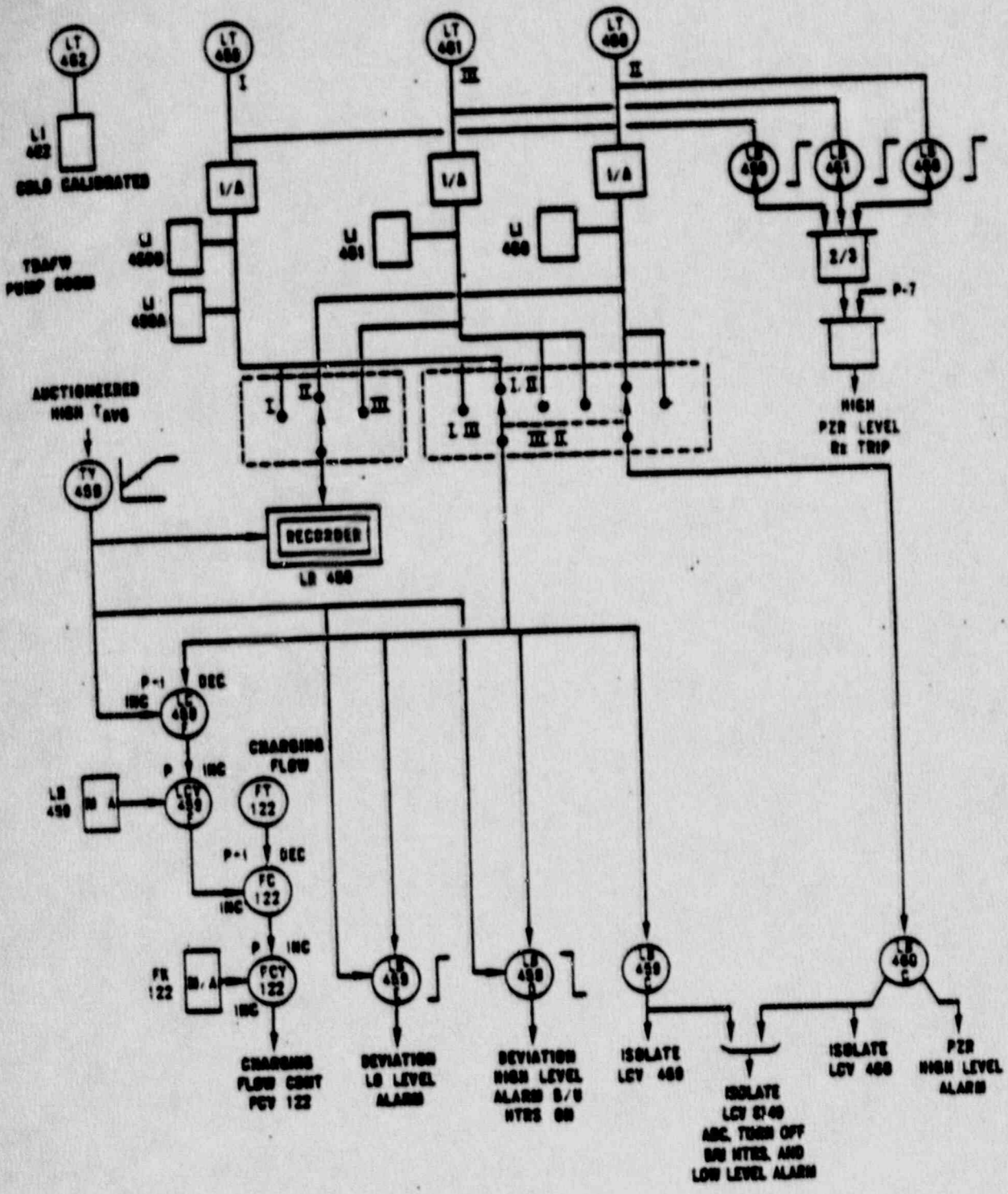
COMMENT

Using the figure provided on the test, pressurizer level protection and control Figure 7 (copy attached), it is clear that LT-459 failing high would cause the events listed to occur in the sequence provided. Choice d. is a correct answer.

If LT-460 were to fail low, it would cause letdown to isolate and turn heaters off (event 2). With letdown isolated, actual pressurizer level would increase. LT-459, sensing the increase, would reduce charging flow to minimum (event 1). Even with minimum flow, pressurizer level would continue to rise until actual level reached the high level setpoint, causing a reactor trip (event 3). The question does not state the events occurred in the sequence listed. With all events occurring for LT-460 failing low, choice b. is also a correct answer.

RECOMMENDATION

Based on two correct answers, the question should be deleted from the exam. The stem of the question should be changed to state that the events occurred in sequence for future use.



PRESURIZER LEVEL PROTECTION AND CONTROL

FIGURE 7

QUESTION RO 2.22/SRO 5.04 (1.00)

Your Unit has experienced a Small Break LOCA accident. Briefly explain why RCS pressure might stabilize at 1300 psig for a long period of time, even though break flow rate is greater than SI flow rate. (Assume the pressurizer has emptied)

ANSWER 2.22/5.04 (1.00)

More energy is being put into the RCS from decay heat than is being removed by energy removal means (1.0)

Reference:

lesson plan 052702A, Obj. 5-004
FNP exam question bank
000009K306 ...(KA'S)

COMMENT

The question does not necessarily solicit the answer key response. WOG Background information for E1 (pages 16 & 25 attached) on page 16 for 1-2 inch breaks discuss the energy balance and energy removal causing pressure to remain elevated. Or page 25 for a 3 inch break (last paragraph), it discusses the RCS going into saturation to stabilize and maintain RCS pressure elevated.

RECOMMENDATION

Accept for full credit the answer key answer, including any reasonable discussion of energy balance. Also accept for full credit any reasonable discussion that discusses the RCS going into saturation without an energy balance discussion.

Breaks ~ 1" < diameter <~ 13-1/2" ($1FT^2$)

For break sizes of one to two-inch in equivalent diameter, the RCS will rapidly depressurize early in the transient, and an automatic reactor trip and safety injection signal will be generated based on low pressurizer pressure. During the early stages of the depressurization, when the system is still full of two-phase liquid, the break flow, which also will be mostly liquid, is not capable of removing all the decay heat. Therefore, the early depressurization is limited by energy removal considerations, and the RCS pressure will temporarily hang up above the steam generator safety valve set pressure, assuming no steam dump is available. The RCS pressure stays at this level in order to provide a temperature difference from primary to secondary so that core heat may be removed by the steam generator. At this energy-balance controlled pressure, however, pumped safety injection flow is less than the break flow, and there is a net loss of mass in the RCS. Voiding throughout the primary side occurs and eventually the RCS begins to drain, starting from the top of the steam generator tubes. The rate of RCS drain is determined by the net loss of liquid inventory, a function of both SI flow and break size.

Prior to the occurrence of draining, heat is removed from the steam generator through continuous two-phase natural circulation, with two-phase mixture flowing over the top of the steam generator tubes. As the draining continues, the natural circulation mode of heat removal as just defined ceases, and core heat is removed through condensation of steam in the steam generator. This method of heat removal is called reflux and is discussed in Reference 2.

The condensation mode of heat removal is almost as efficient as continuous two-phase natural circulation in removing heat. However, condensation heat transfer coefficients may be lower than continuous two-phase natural circulation heat transfer coefficients. Thus, as the steam generator tubes drain, a slight increase in primary system pressure occurs to give a greater ΔT from primary to secondary in order to remove all the decay heat. The steam generator secondary side pressurizes to the safety valve set pressure

This analysis was terminated at 83.3 minutes and does not illustrate the maintenance of long term equilibrium pressure where pumped SI equals break flow, the concept that was described previously. The analysis presented here, however, does represent a stable condition since the core will remain covered and RCS voids are decreasing at the end of the transient. Pressure is stable and the steam generators are not required for heat removal.

This range of break size is large enough that even if maximum safeguards were injected, RCS repressurization would not occur. The characteristic behavior mode would be either as just described, where RCS pressure stabilizes below the steam generator pressure, or as described previously in Category 2, where the RCS pressure stabilizes above the steam generator pressure. The SI flow delivery characteristics and break size will determine the transient behavior mode.

Breaks in the two-inch to thirteen and one-half-inch (13-1/2") diameter range result in significant draining of the RCS and could result in some core uncovery depending on the status of safeguards equipment, break size, and break location. Analyses of cold leg breaks in this category are presented in each plant's SAR to bound the limiting small break for peak clad temperature considerations.

For a three-inch equivalent diameter cold leg break with one train of safeguards equipment operating (one train of pumped safety injection and auxiliary feedwater), the reactor coolant system initially depressurizes rapidly and reactor trip and safety injection are initiated on their respective pressurizer set pressures. The reactor coolant system continues to depressurize relatively rapidly to the pressure where the cold legs become saturated (e.g., P_{sat} at 567°F is ~ 1200 psig). At this point essentially the entire reactor coolant system is at saturation so that core exit voiding and liquid flashing to steam tend to stabilize and maintain pressure. This response is similar to breaks of smaller size where the RCS stabilizes above the steam generator pressure. During this stabilization period liquid is discharged through the break since the RCS has not drained down to the point where steam can be vented through the crossover legs. Since the break is

ENCLOSURE 4

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Alabama Power Company

Facility Docket Nos.: 50-348 and 50-364

Operating Test Administered On: October 24 - 25, 1989

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating test, the following items were observed:

Step 7.6 of FRP-H.1, Response to Loss of Secondary Heat, directs operators to defeat the feedwater isolation signal to main feedwater bypass valves using Attachment 1 to the procedure. This cannot be simulated on the current model. This has been previously identified by the training department, and there are plans to include this in the updated model.