UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

ENCLOSURE 1

EXAMINATION REPORT - 50-302/92-300

Facility Licensee: Florida Power Corporation

Facility Name: Crystal River Nuclear Plant

Facility Docket No.: 50-302

Examinations were administered at the Crystal River Nuclear Plant, Crystal River Florida.

Chief Examiner: Richard S Baldwin — 6/11/92
Richard S. Baldwin — Date Signed

Approved By: Charles A Casto — 6/11/92
Charles A. Casto, Chief — Date Signed
Operator Licensing Section 2
Division of Reactor Safety

SUMMARY

Scope: During the week of May 26, 1992, written and operating examinations were administered to three Senior Reactor Operator (SRO) applicants.

Results: All applicants passed the examinations.

A strength was noted in that the candidates' communication skills exhibited formal repeat backs allowing smooth operations while keeping each other apprised of each individual's area, fostering team work (paragraphs 3d, e).

Weaknesses were noted in the following areas: (1) Performance of off-site dose calculations (paragraph 3a); (2) Lack of use of Annunciator Response procedures during active simulator examinations (paragraph 3b); and (3) Use of place keeping aids while using Abnormal and Emergency Operating Procedures (paragraph 3c).

REPORT DETAILS

1. Facility Employees Attending Exit

   G. Boldt, Vice President, Nuclear Production
   J. Alberti, Manager, Nuclear Plant Operations
   L. Kelly, Director, Nuclear Operations Training
   J. Lind, Licensed Operator Training Manager
   J. Smith, Licensed Operator Training Supervisor
   J. Springer, Simulator Training Supervisor
   T. Miller, Shift Supervisor

2. NRC Personnel Attending Exit

   *R. S. Baldwin, Examiner, DRS
   T. A. Peebles, Chief, Operations Branch, DRS
   C. Tyner, Examiner, EG&G

   *Chief Examiner

3. Discussion

   Operator Performance

   a. A weakness was observed in the candidates' ability to effectively calculate Off-site doses during a release. This was evidenced by the failure rate on a Job Performance Measure (JPM) which required calculation of an Off-site dose based on accident conditions using "Off-site Dose Assessment during Radiological Emergencies" (EM-204A).

   b. A weakness was observed in the candidates' use of Annunciator Response Procedures (ARP). During administration of three simulator scenarios ARPs were used only one time. In at least two instances the use of ARPs would have helped diagnose plant problems.

   c. A weakness was observed in the candidates' use of place keeping aids. It was observed during the simulator examinations that multiple abnormal or emergency procedures could be performed simultaneously. There was no evident means that the candidates' kept their place while using the procedure. One operator was observed using all fingers on one hand to maintain his place.

   d. A strength observed was good communications consisting of formal repeat backs which maintained smooth operations.

   e. In general, crew members maintained each other apprised of their individual areas fostering good team work.

Written exam

A prereview was conducted of the written examination. However, there were six post-examination comments requiring changes. Three of these comments required questions to be deleted, and three changes to the answer key. These changes resulted in a change to the pass decision of one candidate. The facility must be reminded that deletion of questions dilutes the sample plan upon which the examination has been based. Greater effort must be taken in future examinations to preclude question deletion.

4. Exit Meeting

At the conclusion of the site visit, the examiners met with those representatives of the plant staff indicated in paragraph 1 to discuss the results of the examinations and inspection findings. The licensee did not identify as proprietary any material provided to or reviewed by the examiners.

During the administration of simulator scenarios it was determined that Technical Specification 3.1.3.1, based on candidate performance, warrants a technical specification interpretation. Based on the Training Department's comments, a request had been made; however, the Licensing Department had not as yet made a resolution. The facility is reminded that prompt resolution of Technical Specification concerns should be resolved in a timely manner.

In an interoffice correspondence (IOC) from W. Marshall, dated October 3, 1991, to J. Smith and J. Springer concerning Procedure AP-790, "Station Blackout", revealed that this procedure is applicable in Modes 1 thru 4 and not applicable in Modes 5 and 6. This letter stated that an Operations Study Book (OSB) entry was going to be made informing licensed operators that if a Station Blackout occurs in Modes 5 or 6 then the proper procedure to enter would be AP-360, "Loss of Decay Heat Removal". The Training Department updated lesson plans and conducted operator requalification as well as initial candidate training concerning this topic. However, Operations did not follow up with the OSB as stated in their IOC. This topic appeared on the written examination which required an additional answer to be accepted as an additional correct question. The facility is reminded that when procedural guidance is required for procedural inadequacies they should be promulgated to the operators as soon as they are discovered. It is apparent that the feedback mechanism had failed by not issuing the OSB.

It is apparent from post-written examination comments, concerning verification of natural circulation, what the required delta temperature should be to ensure Tc is approximately equal to Tsat. It was recommended the facility determine what temperature difference Operations expects the operators to use in order to meet this condition.

The cooperation of the facility training staff in reviewing and administering the examinations was noted and appreciated.

The licensee did not identify as proprietary any of the material used or reviewed by the examiners.

ENCLOSURE 2

SIMULATOR FIDELITY REPORT

Facility Licensee: Florida Power Corporation

Facility Name: Crystal River Nuclear Plant

Facility Docket No.: 50-302

Operating Tests Administered On: May 27 and 28, 1992

This form is to be used only to report observations. These observations do not constitute, in and of themselves, 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 solely in response to these observations.

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

ITEM	DESCRIPTION
Make-up system	MUV-31 would not operate in automatic.
	Make-up tank pressure and temperature increased with out external input.
Main turbine	Main turbine megawatt output increased with out reason.

ENCLOSURE 3

Florida Power
CORPORATION

May 28, 1992
TRA92 0036

Mr. Rick Baldwin
Operator Licensing Section
Region II
U. S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street, NW
Atlanta, GA 30323

Subject: Crystal River Unit 3
May 26, 1992 - NRC Initial License Exams

Dear Mr. Baldwin:

In accordance with the current practice for examination reviews after NRC Initial Operator Examinations, please find enclosed our review of the technical content of the May 26, 1992 Senior Reactor Operator Examination given at Crystal River Unit 3. We are including our comments and recommended action for each question under review.

If you desire any further information, please contact Mr. Johnie Smith, Nuclear Operations Training Supervisor, at (904) 795-0504, ext. 107. Thank you for your attention to this very important matter.

Sincerely,

Larry Kelley

Larry C. Kelley
Director, Nuclear Operations

JGS/LCK/lb

Attachment

QUESTION: 007 (1.00)

The plant has been shutdown for two (2) weeks and is making preparations for a startup to commence at 5:00pm on 06/15/92.

Which ONE of the following sets of Estimated Critical Position (ECP) calculations are acceptable?

a. ECP #1: -2.68% delta K/K TIME: 6:45am 06/15/92

   ECP #2: -2.45% delta K/K TIME: 7:00am 06/15/92

b. ECP #1: -1.95% delta K/K TIME: 1:00pm 06/14/92

   ECP #2: -1.90% delta K/K TIME: 1:10pm 06/14/92

c. ECP #1: -2.15% delta K/K TIME: 11:50pm 06/14/92

   ECP #2: -2.25% delta K/K TIME: 12:20am 06/15/92

d. ECP #1: -1.35% delta K/K TIME: 3:00pm 06/15/92

   ECP #2: -1.50% delta K/K TIME: 2:00pm 06/15/92

Answer: c
Reference: OP-210, "Reactor Startup", Rev. 30, Page 10
Rot 5-2, "Startup Operating Procedures", RO & SRO L.O. 5.d

[2.6/3.1]

194001A108 ..(KA's)

COMMENT

In order to answer this question, the operator must know that independent ECP calculations must agree within .1% ΔK/K and that the time limit between successive ECP's should not exceed 24 hours. Both pieces of information are administrative requirements found within OP-210 (see attached). It is not the policy of CR-3 to require operators to memorize the information contained within notes and step details of operating procedures. Information of this type is subject to change and is therefore provided to the operator by the procedure at the time of use. Procedures of this type are required to be in the possession of the person(s) preforming the task and signed as completed. Therefore, memorization of this type of information will not improve the outcome of the task.

RECOMMENDATION

We recommend that this question be deleted from this exam and not be used for future exams.

HIT FCP QUES 001

4.2 CRITICALITY ON CONTROL RODS (Cont'd)

ACTIONS	DETAILS
NOTE: This may be N/A if this is not the first approach to criticality and core reactivity has not changed significantly from last ECP: In any event, the time between successive ECP calculations should be as soon as possible and should not exceed 24 hours.	
4.2.3 Perform Two (2) Independent ECP Calculations and record in space provided ECP #1 performed by ______ ECP #2 performed by ______	o ___ Complete (2) ECP Worksheets per Enclosure 1, by different Operators ______/______ Initial/Date
4.2.4 Ensure both ECP Calculations are in agreement prior to Reactor Startup	o Record results on <u>Section 1</u> of Enclosure 3 o Results should be within $\pm$ 0.1% Δ k/k o Attach a copy of any Computer calculations used to determine Xenon or Samarium reactivities to Enclosure 1 ______/______ Initial/Date
4.2.5 Ensure predicted Critical Rod Positions are within Limits	o Record on SP-390 ______/______ Initial/Date

QUESTION # 026 (1.00)

The Crystal River Tech Specs for Emergency Core Cooling Systems (ECCS) identifies a SUBSYSTEM of ECCS as: one High Pressure Injection (HPI) pump, one Low Pressure Injection (LPI) pump and which ONE of the following?

a. One Reactor Building Spray Pump

b. One Decay Heat Cooler

c. One Core Flood Tank

d. The Borated Water Storage Tank

ANSWER: 026 (1.00)

b.

REFERENCE:

CR Tech Spec 3.5.2, Page 3/4 5-3

ROT 4-13, "Engineered Safeguards Actuation System", Rev. 5, L.O. - B10

[3.6/4.2]

006000G011 ..(KA's)

COMMENT

This question also requires knowledge which does not improve the operator's ability to safely operate a nuclear power plant. In this case, the operator must know which of the listed components are included within a particular STS with the HPI and LPI pumps under the heading of an ECCS subsystem. All components listed are covered by STS. All components except the RB Spray pump are part of the ECCS. Both the Decay Heat Cooler and the BWST are listed within the referenced STS, however, the BWST is listed as "an OPERABLE flow path from the borated water storage tank..." (see attached). From this perspective, answers B, C, and D are all correct. Again, it is not the policy at CR-3 to require the operator to memorize information of this type to the degree required to answer this question.

RECOMMENDATION

We recommend that this question be deleted from this exam and not be used for future exams.

ATT FOR QUES - 026

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} > 280°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

QUESTION 042 (1.00)

The Pressurizer Code Safety Valves are designed to prevent a Reactor Coolant System (RCS) pressure in excess of 110% of design pressure.

Which ONE of the following transients could impose this maximum pressure transient on the RCS?

a. A full High Pressure Injection actuation at normal operating pressure.

b. A loss of feed water at 100% power with the PORV failing to operate.

c. A control rod withdrawal accident from zero percent power.

d. A complete load reject from 100% power without a reactor trip.

ANSWER: 042 (1.00)

c.

REFERENCE:

ROT 4-01, "Primary Systems", Rev. 11, Page 6, RO & SRO L.). - 3 & 6b

[3.0/3.5]

002000K612 ..(KA's)

COMMENT

This question requires the candidate to know the specific transient which was used as a basis for the design of the RCS code safety valves. This information is found in the referenced lesson plan under a detailed description of the pressurizer and related components (relief valves). Much of the information provided is included to enhance the required material. It is not intended that all information in any lesson plan be committed to memory. The referenced objectives for this question state "Describe the alarms, indications, control functions, and protective functions associated with RCS pressure" and "Describe operation of the following valves:

a. Power Operated Relief Valve
b. Pressurizer Code Safety Valves
c. RCS High Point Vents
d. Pressurizer Vents"

Neither objective is intended to imply that the operator be able to state the basis of design for the listed components.

RECOMMENDATION

We recommend that this question be deleted from this exam and not be used for future exams.

COURSE: SYSTEM TECHNOLOGY

LESSON: PRIMARY SYSTEMS

TERMINAL OBJECTIVE: (RO and SRO)

At the completion of this lesson the student shall be able to discuss the operation, design, flow paths, and operating requirements for the following systems:

1. Reactor Coolant System
2. Makeup and Purification System
3. Decay Heat Removal System
4. Core Flooding System
5. Reactor Building Spray System

CHAPTER 1: REACTOR COOLANT SYSTEM

LEARNING OBJECTIVES: (RO and SRO)	REFERENCE NUMBER:	ADDRESSED BY:
1. State the functions of the reactor coolant system.	0020101009	1.1
2. Draw a simplified one line diagram of the Reactor Coolant System as shown in FIGURE 1 of this lesson.	0020101001	FIG. 1
3. Describe the alarms, indications, control functions, and protective functions associated with RCS pressure.	0000501001 0100101004 0100101007	1.4.3 2.1.3 2.2.2 2.2.3 2.2.5 2.4
4. Describe operation of the PZR. Include the cause / effect of a pressurizer insurge and outsurge.	0100101004	1.2 1.3 1.4.3
5. Explain how to manually control PZR heaters and spray.	0100101001 0100101009 0100401001 0100501002	2.2.2 2.2.3 2.2.4

COURSE: SYSTEM TECHNOLOGY

LESSON: PRIMARY SYSTEMS

LEARNING OBJECTIVES: (RO & SRO)	REFERENCE NUMBER:	ADDRESSED BY:
6. Describe operation of the following valves: a. Power Operated Relief Valve b. Pressurizer Code Safety Valves c. RCS High Point Vents d. Pressurizer Vents	0100101006 0100101007 0100101008 0100201003 0100501001	1.4.3 2.2.5 2.2.6 2.2.8
7. Describe how the RCS is degassed.	0020101013	4.3.5
8. Describe the alarms, interlocks, control functions and protective functions associated with PZR level.	0000501001 0110101001 0110101003 0110401001	2.1.1
9. Describe a RCP seal package and indications of normal and abnormal RCP seal operation.	0030101002 0030401002	1.4.4
10. List the interlocks that must be satisfied to start a RCP.	0030101001 0030101002	2.3.3
11. Describe how to bypass RCP start interlocks.	0030501001	2.3.3
12. Identify the Limiting Conditions for Operation of the following when given a copy of Technical Specifications. a. Coolant Loops and Circulation b. Code Safety Valves c. Power Operated Relief Valve d. Pressurizer e. RCS Leakage Detection Systems f. RCS Leakage g. RCS Chemistry h. RCS Activity i. RCS Heatup and Cooldown limits j. PZR Pressure/Temperature limits k. RCS Structural Integrity l. RCS Vents	0020101011 0020101012 0020401001 0030401002 0100101001 0350401001 1190301015 3410103036	5.0 STS

The pressurizer is a vertical cylindrical vessel with a bottom entry 10 inch surge line penetration connected to the "A" reactor coolant hot leg piping, removable electric heaters in its lower section, and a 2.5 inch water spray nozzle in its upper section to permit system pressure adjustments. It is protected from thermal effects by thermal sleeves in the surge and spray lines, and by an internal diffuser above the surge pipe entrance to the pressurizer. The spray nozzle inside the pressurizer is a cast stainless steel unit screwed to the spray piping and adjusted for a vertical spray pattern. The pressurizer is made of carbon steel for strength and clad with stainless steel for corrosion resistance. The pressurizer immersion heaters are arranged in three heater bundles located in the lower portion of the vessel. Each bundle contains 39 heater elements (total 117). Thirteen electric circuits serve each bundle. This is accomplished by connecting three heaters per circuit. This electrical lineup allows selected heaters to be used to obtain desired water temperature without energizing the entire bundle. The total electric heater capacity is 1638 KW at 480 VAC. Each heater element can supply 14 KW along its 6 2/3 foot length. The heaters are arranged into five banks for control. Heater banks "A" and "B" are sized to provide enough heating capacity to compensate for normal heat losses and spray valve bypass flow during steady state operation. Banks "C","D", and "E" provide additional heating during startup and load changes.

PRESSURIZER HEATER DATA

Total number of elements..........................	117
Element rating..	14 KW
Element length..	9.245 ft.
Number of assemblies...............................	3
Elements per assembly.............................	39
Assembly rating.......................................	546 KW
Overall rating..	1638 KW

Two pressurizer code safety valves (RCV-8 & 9) are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valves inlet and outlets are flanged to allow removal for maintenance and setpoint testing. The total design capacity of each code safety valve is 317,973 lbm/hr. The capacity is determined by the maximum pressure transient imposed on the RCS (rod withdrawal from zero power). The code safety valves are sized to prevent a pressure in excess of 110% of design pressure. The set pressure of the code safety valves is 2500 psig.

QUESTION: 055 (1.00)

The Integrated Control System (ICS) Turbine Bypass Valve (BPV) hand/auto station is in "Hand" with the bypass valves closed.

Which ONE of the following will fully open the bypass valves?

a. Steam generator pressure is 1045 psig.

b. Steam generator pressure is 1025 psig.

c. Turbine header pressure is 1000 psig with a reactor trip.

d. turbine header pressure is 875 psig with a turbine trip.

Answer: 055 (1.00)

a

Reference:

Rot 4-14, "Integrated Control System", Rev. 3, Pages 13 & 14, RO & SRO L.O. - 3c

[2.9/2.9]

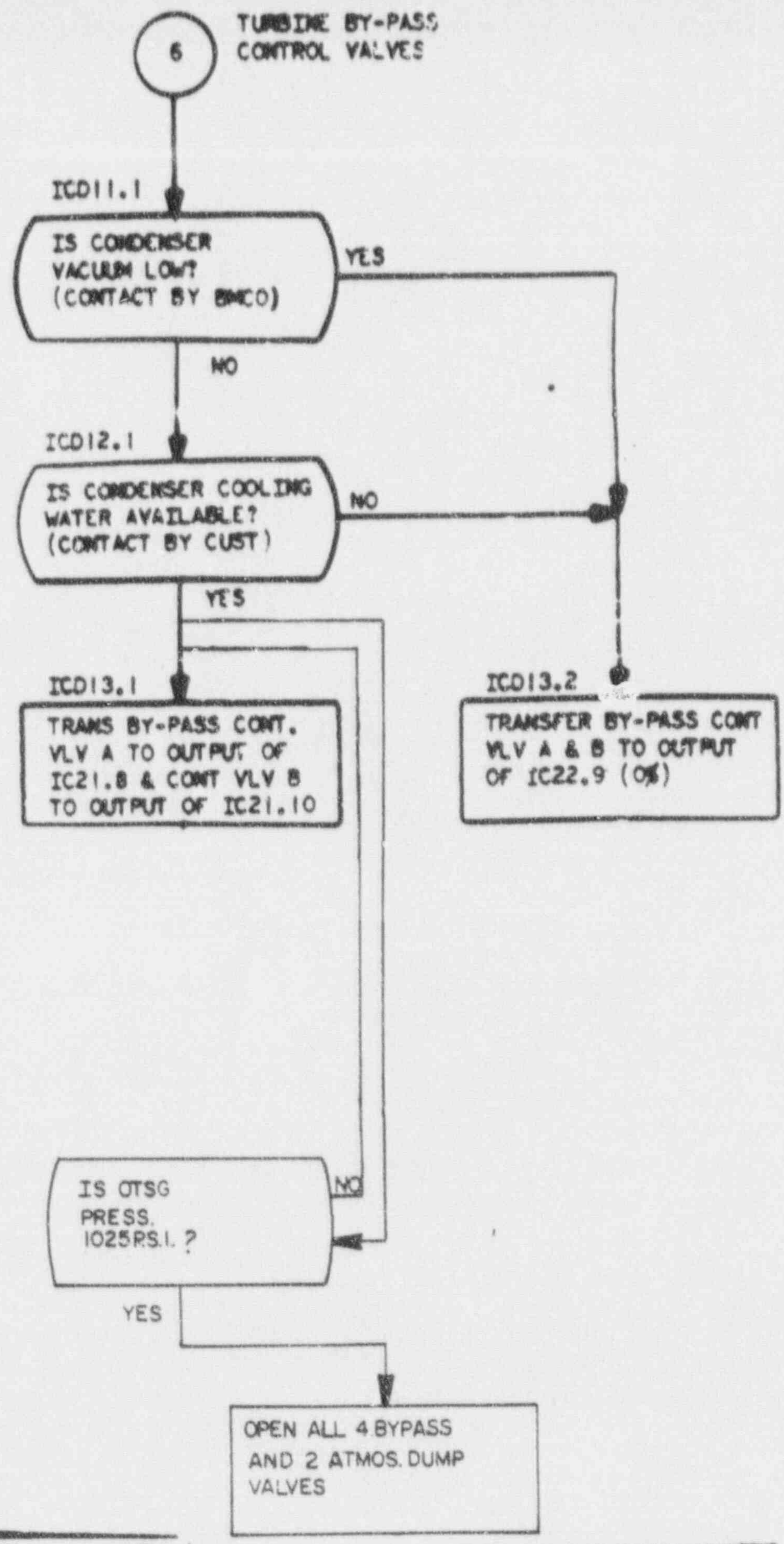
041020A305 ..(KA's)

COMMENT

The purpose of the turbine bypass valve overpressure protection circuit is to open the valves when OTSG pressure exceeds a specific value (1025 psig). This is true of any relief valve. The pressure value at which the valve is fully open is not normally an operator concern once he has determined that the valve has opened at the correct value and is attempting to control pressure. The information used for this question is again information which was added to enhance the lesson plan and not intended as required knowledge. The objectives listed in this lesson plan and training provided to the operators have stressed the primary function of the valves to open at the setpoint of 1025 psig which corresponds to choice "B". As an additional consideration, the logic strings for the ICS show an additional circuit which will open the bypass valves fully at the 1025 psig setpoint (see attached). These logic drawings are given to the operators for training. Based on the above, it is reasonable to assume that an operator would select the 1025 setpoint as the full open value.

RECOMMENDATION

Accept answer "B" as correct.



BY	CK	APP	DATE
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4

3

REV	DATE	
E	[illegible]	KEEK
F	[illegible]	REV/SET AGAIN
G	[illegible]	[illegible]

BABCO
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6

XACCEPTE
ACCEPTE
NOT AC
REVIEW FROM DEVIATION FOR
ENGINEER
ENG MGR

QUESTION # 069 (1.00)

Which ONE of the following are the plant operational modes that require use of AP-790, "Station Black-out" during a station blackout?

AP-790 shall be used when the plant is in:

a. any of the six operational modes.

b. operational modes 1 through 4 only.

c. operational modes 1 through 3 only.

d. operational modes 1 and 2 only.

ANSWER: 069 (1.00)

b.

REFERENCE:

ROT 5-80, "AP-790 Station Black-Out", Rev. 3, Page 2, L.O. - B1

[3.6/3.7]

000055G007 ..(KA's)

COMMENT

This question could be answered from either of two perspectives. First, the procedure entry conditions as written imply that this procedure is applicable in all modes (since there are no mode restrictions listed). From this perspective, the candidate would select "A" (any of the six operational modes). However, a recent concern regarding the use of this procedure in modes 5 and 6 has led to the issuance of a letter to training (attached) stating that the actions of this procedure are designed to be used during modes one through four only and that procedure revisions are under review. This information was included in the referenced lesson plan (attached). The combination of the above places the candidate in the undesirable position of answering the question in reference to strict adherence to the procedure as it exist today or according to the way the procedure will be following review and revision.

RECOMMENDATION

We recommend either response "A" or "B" be accepted as correct or that this question be deleted from this exam and that it be verified against the procedure prior to use on future exams to ensure that revisions have been made as per the referenced letter.

TRAINING DEPT.
CONTROLLED COPY

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

NUCLEAR OPERATIONS TRAINING DEPARTMENT

PROGRAM: REPLACEMENT OPERATOR TRAINING

COURSE: OPERATIONS AND ADMINISTRATION

LESSON: AP-790 STATION BLACK-OUT

LESSON NO: ROT-5-80

REVISION NUMBER: 3

Prepared By: C.M. CROSTEN 1-8-92
/Date

Reviewed By: [signature] 1/9/92
Nuclear Training
Academic Specialist
/Date

Approved By: Johnnie Smith 2-11-92
Nuclear Training
Supervisor
/Date

IC-1 ____________________
/Date

IC-2 ____________________
/Date

IC-3 ____________________
/Date

EXPLANATION OF STEPS

1.0 ENTRY CONDITIONS

This procedure is written for a total loss of AC power to the Crystal River Unit #3 operating facility. This requires the simultaneous loss of all offsite power feeds (Unit #3 start-up transformer, Unit #3 unit transformer and the "Off-site" transformer) and a failure of both of the onsite 4160V Emergency Diesel Generators. This condition leaves the plant with only a minimum of DC powered equipment and the steam driven Emergency Feedwater pump for plant control and stabilization. While not specified by the entry conditions, this procedure is designed for use in operational modes 1 through 4 only. During a loss of power in modes 5 or 6, guidance for mitigation of the transient will be given via AP-360, Loss of Decay Heat Removal (ref. #3).

009771

INTEROFFICE CORRESPONDENCE

Florida Power CORPORATION

Nuclear Operations — Office | NR3B — MAC | 230-4286 — Telephone

SUBJECT: Use of AP-790 and AP-360

TO: J. G. Smith
J. L. Springer

DATE: October 3, 1991
OP91-052

The purpose of this IOC is to clarify AP usage for combatting a Station Blackout Event (SBO). While conducting a verification and validation of AP-360, Loss of Decay Heat Removal, it was revealed that the operating staff would use AP-790 to "lead" them through the event even if the event occurred while in mode 5. Their conviction was strong enough to warrant not only this IOC but an OSB entry which is soon to follow.

Although the Entry Conditions don't specifically address mode constraints on procedure applicability, AP-790, Station Blackout, is only intended for implementation from a higher mode than 5 or 6. NRC and NUMARC only addressed SBO from a pre-event condition of Mode 1, 100% power and all systems in a ready/standby state. These assumed conditions are no longer valid in modes 5 or 6. Our SBO mitigating procedure (AP-790) was developed as a function of the NUMARC position (87-00). The actions contained within AP-790 enable us to cope with a SBO event based on EFW availability and possibly the CFTs.

AP-360, Loss of Decay Heat Removal, was designed to provide the guidance needed to ensure core cooling during a low mode event, when OTSG cooling is most likely not available.

I am evaluating the use of a discriminator within AP-790 that sends the operating staff to AP-360 if pre-event temperatures were < 200°F. As a result of some concerns raised at a Shutdown Risk Task Force meeting I am also evaluating the need for a "Shutdown SBO" procedure, which would contain some of the information presently located within AP-360.

Until procedural changes are generated we are asking that you reinforce the use of AP-790 for Modes 1 through 4 and AP-360 for Modes 5 and 6 to cope with SBO events.

If you have any questions please contact Paul Fleming at extension 4286. Thank you for your support.

Paul V. Fleming
Assistant Nuclear Shift Supervisor

W. Marshall

W. M. Marshall
Nuclear Operations Superintendent

PVF/91-35
xc: T. Miller

QUESTION: 083 (1.00)

The plant has had a loss of all running Reactor Coolant Pumps. All expected automatic actions have occurred and adequate subcooling margin exists. The operator is using AP-530, "Natural circulation", to verify natural circulation exists.

Which ONE of the following conditions is a confirmation that natural circulation flow has been established?

a. Tcold is approximately 10 degrees F above OTSG Tsat and is steady.

b. Incore temperatures are increasing with Thot, with a 9 degrees F differential.

c. There is a 70 degree F, and increasing, difference between incore temperature and Tcold.

d. Thot, Tcold and incore temperatures do not change when OTSG pressure is changed.

Answer: 083 (1.00)

b.

Reference:

AP-[illegible]0, "Natural Circulation", Rev. 9, Pages 11 and 12

Rot 5-23, "AP-530 Natural Circulation", Rev. 5, L.O. - No Learning Objective Identified

[4.4/4.5]

000015A121 ..(KA's)

COMMENT:

The distractors for this question are paraphrased from a list of indicators used to confirm natural circulation listed in AP-530 (see attached). The first indication listed is "Tc ≈ Tsat of OTSGs". The first distractor states "Tcold is approximately 10 degrees F above OTSG Tsat and is steady". There is no reference which defines "approximately equals" therefore the operator must decide if this should be something less than 10 degrees. Since the stem of this question does not indicate that OTSG Tsat is changing, there is no reason to say that Tc should be changing since Tcold should follow OTSG Tsat. This distractor therefore represents a condition which is reasonable during natural circulation if a 10 degree differential is considered approximately equal. Another indication listed in the AP is "Incores follow Th within 10°F". The distractor for this indication states "Incore temperatures are increasing with Thot, with a 8 degree F differential". In this distractor the temperature differential is less than that required by the AP, however, if natural circulation has been established, the operator would first expect Thot and incores to be either steady or decreasing. For these to increase, either natural circulation is still developing (ΔT between Th and Tc not steady), or the amount of heat removal by the OTSG's does not match heat production by the reactor. Thus, both distractors could be indications of the development of natural circulation and neither alone is an indication of steady state natural circulation.

RECOMMENDATION

We recommend that either A or B be accepted as correct or that this question be deleted from this exam and be reworded prior to use on future exams.

3.3	<u>IF</u> adequate subcooling margin does <u>NOT</u> exist, <u>THEN</u>: o Raise OTSG levels to 95% using EFW, o Start full HPI, o GO TO AP-380, ES Actuation, beginning with Step 3.5.

Table 2: Nat Circ Cooldown Rates

RCS Tc	°F/hr
> 280°F	≤ 10
280 to 150°F	≤ 5
< 150°F	≤ 2.5
*	≤ 50

* Only to be used if RCS PRESS is maintained above Nat Circ cooldown curve.

3.10	<u>IF</u> RCPs becomes available, <u>AND</u> OTSG heat removal is available, <u>THEN</u> ensure PORV is closed <u>AND</u> start 1 RCP/loop, <u>AND</u> GO TO Step 3.26 in this procedure.

Table 3: Indications of Nat Circ

Tc ≈ Tsat of OTSGs
ΔT (incore - Tc) develops and stabilizes
Incores follow Th within 10°F
Th, Tc, and incores lower when OTSG PRESS is lowered

ENCLOSURE 4

NRC RESOLUTION OF FACILITY COMMENTS

Comment

1. Question 7 SRO: Comment accepted. The facility is reminded that pre-examination review should be technical in nature in order to prevent question deletion. This question will be deleted from the examination and total point value adjusted accordingly.

2. Question 26 SRO: Comment not accepted. It is felt that an SRO should know from memory what plant components will affect the operability of one of only two ECCS subsystems. This question, and supporting K/A, were used to determine the operators' knowledge of plant components comprising a train, or subsystem, of ECCS when the operability of that entire system may be affected by the failure of individual components. The answer key will not be changed.

3. Question 42 SRO: Comment accepted. The facility is reminded that pre-examination review should be technical in nature in order to prevent question deletion. This question will be deleted from the examination and total point value adjusted accordingly.

4. Question 56 SRO: Comment accepted. The answer key will be changed to reflect that "b" is the only correct answer.

5. Question 69 SRO: Comment accepted. The answer key will be changed to reflect that "a" and "b" will be accepted as correct answers.

6. Question 83 SRO: Comment accepted. The facility is reminded that pre-examination review should be technical in nature in order to prevent question deletion. This question will be deleted from the examination and total point value adjusted accordingly.