U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249
License Nos:	DPR-19; DPR-25
Report Nos:	50-237/97304(OL); 50-249/97304(OL)
Licensee:	Commonwealth Edison Company (ComEd)
Facility:	Dresden Station Units 1, 2, and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	July 28 - August 6, 1997
Examiners:	M. Bielby, Chief Examiner D. Muller, Examiner R. Doornbos, Examiner J. Ellis, Examiner (Certification) T. Jones, Examiner in Training J. Munro, NRR HOLB (Auditor)
Approved by:	Melvyn Leach, Chief, Operator Licensing Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Dresden Station, Units 1, 2, and 3
NRC Examination Reports 50-237/97304; 50-249/97304

An NRC developed and licensee reviewed initial operator licensing examination was administered to four Senior Reactor Operator and three Reactor Operator applicants. One of the individuals applying for a Senior Reactor Operator license was previously licensed at Dresden on Units 2 and 3. The examination process incorporated a one week on-site validation period for review of the examination by facility personnel.

Examination Results:

- All seven applicants passed all portions of their respective examinations.
- One Reactor Operator applicant was not issued a license based on termination of employment at Dresden Station subsequent to the examination.
- One Senior Reactor Operator applicant not previously licensed was issued a Reactor Operator license due to eligibility concerns identified by the NRC.

Examination Summary:

- NRC examiners identified a discrepancy between a Dresden Safe Shutdown Procedure and equipment improperly retired in place (VIO 50-237/97304-01; 50-249/97304-01) (Section O2.1).
- More attention to detail needs to be applied toward validation of training and examination material (Section O3.1).
- The licensee performed a thorough technical review of the written examination (Section O5.2).
- Applicants demonstrated good overall communications, briefings, and command and control of emergency procedures during simulator scenarios (Section O5.3).
- Applicants demonstrated instances of inattention to detail and failure to follow procedures during dynamic simulator scenarios (Section O5.3).
- Applicants demonstrated lack of knowledge areas and poor procedural adherence during administration of selected JPMs and associated questions (Section O5.3).
- The licensee made a positive contribution to examination security. (Section O5.4)
- The simulator exhibited a large number of fidelity issues, limited malfunction capability and contributed to a lost evaluation event (Sections O5.2.b(3) and (4), O5.3.b(5), O5.5).

Report Details

I. Operations

O2 Operational Status of Facilities and Equipment

O2.1 Improper Equipment Retirement

a. Scope

On July 18, 1997, NRC examiners and licensee instructors performed a Job Performance Measure (JPM) walkdown of Dresden Safety Shutdown Procedure (DSSP), 0200-T3, Revision 4, "2/3 Diesel Generator Local Manual Start," as part of the validation process for an initial operator license examination.

b. Observations and Findings

Procedure step G.16.b required verification of diesel generator cooling water flow at the local flow indicator, FI-2/3-3941-882A, for the 2/3 Diesel Generator Cooling Water Pump. NRC examiners observed the local flow instrument was labeled "Retired In Place" and questioned licensee training staff members, consisting of an onshift RO and an instructor that was a former onshift SRO, about the discrepancy between the procedure step requirement and indicator label. The licensee's initial response was the instrument could be used for flow indication, but a new remote indication in the turbine building should be used for surveillances because it was calibrated. Furthermore, the licensee identified the remote indication read total cooling water flow and the local indication read only cooling water flow to the diesel. The licensee agreed there was reason to believe a discrepancy with either the procedure or the label existed. The licensee was concerned about addressing the issue because of examination security, however, the examiners clearly stated that whatever changes were necessary, should be completed to ensure the DSSP procedure was correct.

On July 30, 1997, the JPM examination was administered to the license applicants and the NRC examiners observed that neither the DSSP procedure nor the local instrument label had been changed. On July 31, 1997, NRC examiners discussed the issue with licensee examination staff members, including a representative from the Operations Department. On August 6, 1997, during the initial operator license examination exit, the issue was identified as a potential violation. The licensee immediately initiated a Problem Identification Form (PIF) D1997-016108 to track the problem.

On August 13, 1997, the licensee provided the results of its investigation. The report stated that the local flow indicator had been inadvertently retired in place and removed from the maintenance calibration schedule. The licensee determined that the local flow indicator was still within its calibration period, and subsequently calibrated and returned it to the normal calibrated schedule. The flow instrument placard was replaced with one that read, "Do not use this FI for IST data".

The PIF investigation identified that in February 1997, a system engineer had incorrectly concluded the new turbine building indicator modification had replaced the local flow indicator. Actually, the new turbine building flow indicator modification was installed in

1994 as a method to measure total cooling water flow to the 2/3 Diesel Generator, LPCI, and HPCI room coolers, because the local flow indicator only measured flow to the 2/3 Diesel Generator. The PIF investigation documented a procedural non-compliance in that the system engineer failed to execute the steps of DAP 21-09, "Control of Retired Plant Equipment," Revision 1. The system engineer inappropriately determined that the "Retired in Place" label was the action to take without using any available input to his decision process and without determining whether a procedure or process existed for task performance. The engineer's failure to comply with the requirements of DAP 21-09 was a violation of Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings."

The PIF investigation further documented that operability of the 2/3 diesel generator was unchallenged as a result of the event. In addition to the previously mentioned corrective actions, the PIF investigation documented the Engineering supervisor was scheduled to counsel the system engineer in accordance with station policy (NTS # 237-200-97-04021), by October 24, 1997. No similar industry events involving retired components were identified during the licensee's investigation. However, the PIF investigation identified one example of a component being inappropriately retired in place at Dresden one month prior to distribution of DAP 21-09, Revision 0. The PIF investigation identified that since creation of DAP 21-09, there have been nine PIFs which documented components no longer in use and identified that consideration should be given to permanent retirement of components in service. The PIF investigation stated the NRC identified event stood as the only procedural non-compliance for proper retirement of components.

c. Conclusions

A violation was identified for procedural non-compliance because the system engineer failed to obtain the required engineering and configuration reviews before deciding to have a "Retired In Place" label prepared and posted above the flow indicator for the 2/3 Diesel Generator. (VIO 50-237/97304-01; 50-249/97304-01).

O3 Operations Procedures and Documentation

O3.1 Operations Procedures Deficiencies

a. Scope

NRC examiners evaluated usage and adequacy of procedures during examination validation and administration conducted the weeks of July 14 and 28, and August 4, 1997.

b. Observations and Findings

Procedure errors, weaknesses and inconsistencies were identified based on examiner observations of operator performance. The licensee wrote procedure revision requests for recognized errors.

(1) Minor Procedure Errors

- Wrong indicator designation. DOP 0202-01, "Reactor Recirculation System Startup," Revision 19; page 11, steps G.20.f and g, indicator should be PI-262 vice PI-260.

- Steps out of order. DOS 0500-05, "Calculation of Core Thermal Power," Revision 10; step I.3.f should come after step I.3.h.

(2) Procedure Weaknesses

Electrical procedures did not contain a caution for locally checking buses for faults before cross-tying them to restore electrical power. Abnormal procedure DOA 6500-01, Revision 12, "4KV Bus Failure," references procedures DAP 07-34, "4KV Electrical Distribution Management," and DOP 6700-02, "Transferring 480 Volt Busses;" however, none of the procedures contain a caution for checking local buses for faults such as over current prior to cross-tying them. During scenario #2, a local over current trip on Bus 23-1 caused a loss of Bus 28 and inability of the emergency diesel generator to energize Bus 28. The shift supervisor initially considered restoration of electrical power to Bus 28 by cross-tying it with Bus 29. The order was not given because another operator identified that a potential over current could exist on one of the local busses and they should send someone out to investigate the bus status prior to re-energizing it. Examiners observed the validation crew consider performing the same action of cross-tying a faulted bus. Examiners validated that cross-tying busses 28 and 29 in this situation would have resulted in a scram with group 1 isolation. Since the scenario was designed as a 100% anticipated transient without scram (ATWS), a more significant challenge to containment would have resulted based on the higher reactor power with main steam isolation valve (MSIV) closure.

Examiners identified the reactor recirculation procedure referenced parameter values that were not nominal for starting a second reactor recirculation pump (RRP). DOP 0202-01, revision 19, "Reactor Recirculation System Startup," step G.20, gives reference values for reactor recirculation motor generator (RRMG) set drive motor and pump motor parameters after a RRP start. These values did not appear to be nominal for starting a second RRP when operating in single loop. JPM 1 required applicants to restart the tripped "B" RRP with "A" RRP already running. When applicants performed step G.20, they compared the motor parameter indications of the "B" RRP with the running "A" RRP to verify proper operation because the procedure referenced values were significantly different.

(3) Procedure Inconsistency

Procedures for unloading emergency diesel generators were inconsistent. Neither procedure provided an adequate load margin from which to open the output breaker to prevent an inadvertent reverse power trip. DOP 6500-09, revision 7, Bus 24-1 To Bus 34-1 Tie Breaker Operation Utilizing U2(3) D/G, step G.1.g.(8), directed the operator to unload the Unit 2 emergency diesel generator and open the output breaker. DOP 6600-03, revision 10, Diesel Generator 2(3)

Shutdown, directed unloading the emergency diesel generator by reducing load to zero, and then opening the output breaker. JPM 5 directed the applicant to realign the electrical power supply to Bus 24-1 from the Unit 2 emergency diesel generator to Bus 34-1. On two separate occasions, applicants unloaded the emergency diesel generator to zero KW and inadvertently allowed the output breaker to trip on reverse power.

c. Conclusions

A significant number of procedural errors was noted during the validation of the examination. Since these procedures are used frequently during training sessions with operations department personnel, it is indicative of a lack of questioning attitude by station operators and trainers concerning operating procedures. Additional attention to identifying and correcting procedural errors as they are detected by station personnel is warranted.

O5 Operator Training and Qualification

O5.1 General Comments

Operator initial licensing examinations were developed and administered by NRC examiners. The licensee performed a technical review and assisted the NRC with validation of the examination material at the Dresden Station.

O5.2 Examination Preparation and Validation

a. Scope (NUREG-1021)

NRC examiners prepared the examination using guidance prescribed in NUREG 1021, "Operator Licensing Examiner Standards," Revision 7, Supplement 1. The licensee performed a technical review, and assisted the NRC with onsite validation of the examination material the week of July 14, 1997.

b. Observations and Findings

The licensee performed an extensive review of the written examination during the validation week. Four licensed operators took the written examination, and approximately ten training and operations personnel reviewed individual questions. Conversely, the licensee only assigned four individuals to validate the operating examination. NRC examiners revised the operating examination material during the validation week; however, written examination comments were not completed by the licensee until late in the week, so incorporation of those comments was delayed until completion of the operating examination administration.

(1) Written Examination

The licensee's written examination comments were numerous and not completed until the last day of the onsite examination validation. The licensee provided comments on all 127 questions. More than two thirds of the comments were considered editorial changes to the question stem or distractors, or requested use of station terminology to improve clarity. As a result, NRC examiners and licensee staff agreed to delay the written examination administration two days to allow time to review and incorporate those comments, and to allow a final review by the licensee.

The licensee requested deletion of 22 questions primarily based on procedure revisions, plant modifications, inappropriate level of operator knowledge, more than one correct answer or double jeopardy reasons. NRC examiners determined that 14 of those questions required replacement or rewrite. Many of the question inaccuracies were because the licensee had streamlined their procedure revision process. A significant number of station procedures were combined, deleted or rewritten during the examination preparation period. The licensee suggested replacement questions of like knowledge or ability; however, NRC examiners either wrote entirely new questions, or significantly modified the licensee's suggested questions. All of the licensee's requested editorial changes were incorporated.

Three handouts were required for the RO examination and five handouts were required for the SRO examination. The handouts consisted of specific tables, curves, technical specifications, and one emergency operating procedure for reactor pressure vessel flooding with entry conditions and setpoints removed.

(2) <u>Administrative Job Performance Measures</u>

No significant comments or changes were identified by the licensee. Comments ranged from question clarification to minor technical comments to match the licensee's learning objectives. The licensee failed to identify all required isolation valves for tag out of the Reactor Water Cleanup (RWCU) Auxiliary pump seal replacement. During examination administration, applicants identified additional vent and drain valves. Overall, the licensee's comments enhanced the quality of the administrative examination.

(3) <u>Operating Job Performance Measures</u>

The licensee identified appropriate technical issues and provided reasonable alternatives for questions and tasks. On two different occasions examiners solicited the licensee's assistance in developing solutions to problems identified while validating JPMs:

- During validation of a simulator JPM for unloading the emergency diesel generator, examiners requested that the Operations Department representative provide pass/fail criteria for the JPM in lieu of available procedural guidance (DOP 6500-09 and DOP 6600-03, Section O3.1 of this report). The provided criteria was that emergency diesel generator electrical load should be reduced to less than 200 KW and the breaker manually opened prior to tripping on Reverse Power.

- The format of a JPM had to be changed because of limited availability of validated simulator malfunctions for the Standby Gas Treatment (SBGT) system. Examiners had originally written a JPM that required an operator response to trip the 2/3B SBGT fan during a surveillance. However, the licensee identified they had no validated simulator malfunctions to perform the event. Consequently, examiners rewrote the JPM to have the inoperable 2/3A SBGT running to perform a post modification test, and require the operator to respond to an incomplete Group II isolation. The operator was expected to identify the incomplete isolation and that the running 2/3A SBGT was inoperable. The operator was expected to start the operable 2/3B SBGT.

During examination preparation a knowledge weakness was demonstrated in the area of EDG start logic. The trainers and operators assigned to validate JPM 9, Question 1, did not understand a portion of the EDG start logic and initially challenged the technical validity of the question. Upon further review of the EDG training material, learning objectives, and design documentation, the staff determined the NRC provided answer was correct. This weakness was further demonstrated during applicant examinations as none of the applicants were able to provide the correct answer without consulting a reference. One applicant was unable to determine the correct answer with references available. This knowledge deficiency was referred to the Dresden Station NRC resident staff since it involved safety related equipment.

(4) Dynamic Simulator Scenarios

The simulator had a comparatively limited range of malfunctions and initial conditions for scenario development. The licensee identified that generally only master controllers vice individual components could be failed. The licensee suggested ways to adequately incorporate events requested by the examiners for the appearance of smooth scenario transition. However, the process required additional time to develop the final scenario. Overall, the final simulator scenarios were satisfactorily validated with good support from the licensee, but at the expense of requiring additional validation time.

c. Conclusions

The licensee performed a rigorous review of the written examination and provided a large number of non-technical comments. A majority of the technical comments resulted from "out of revision" reference material supplied to NRC examiners very early in the written examination preparation process. Although less rigorous, the operating examination received satisfactory licensee attention during the validation week. A significant number of licensee comments improved clarity and corrected technical errors. Development of scenario events required additional time because of the limited ability to fail individual simulator components. Licensee support was good throughout the examination validation.

O5.3 Examination Administration

a. Scope (NUREG-1021)

NRC examiners administered written and operating examinations to three RO and four SRO applicants during the weeks of July 28 and August 4, 1997 using guidance prescribed in NUREG 1021, "Operator Licensing Examiner Standards," Revision 7, Supplement 1.

The examination consisted of the following:

- A 100 question multiple choice written portion focused on a broad spectrum of plant system design and operation.
- A walkthrough portion focused on administrative topics and operating JPMs related to control room and inplant systems.
- A dynamic, performance-based simulator scenario portion focused on integrated plant operations.

b. Observations and Findings

(1) Written Examination

The NRC developed written examination was considered technically accurate with two exceptions. One post examination comment developed by the facility, and listed in Enclosure 4, identified a typographical error in the RO Answer Key. NRC resolution to that comment is listed in Enclosure 5. Additionally, an NRC post examination review identified that SRO Question 60 had a second correct answer. The question and possible answers were:

> In reference to the Standby Liquid Control System, a MINIMUM of 600 ppm boron in the reactor core is required to be injected within 60 minutes. Which ONE of the following statements is correct concerning the boron injection?
>
> a. A 600 ppm concentration will provide at least a 3% delta K Shutdown Margin during cold xenon free conditions.
>
> b. The 100 minute maximum time requirement is necessary to provide adequate mixing, AND prevent "chugging" in the core.
>
> c. The boron solution in the core results in changing the moderator temperature coefficient from positive to negative.
>
> d. The required 600 ppm boron concentration, in the core, includes a 25% additional margin to accommodate improper mixing.

The answer key listed 'd' as the correct answer. However, the post NRC examination review identified that answer 'a' was also correct. Applicant's written examinations were graded to reflect the two post examination corrections.

NRC examiners identified applicant knowledge weaknesses in the following four areas based on the number of incorrect answers:

(1) RO Question 3 / SRO Question 3: Two of three RO's and two of four SRO's missed the question. The question asked for the basis of not needing the RWM (Rod Worth Minimizer) above 20% reactor power.

(2) RO Question 51: Three of three RO's missed the question. The question asked for the Safety Parameter Display System (SPDS) color and position indication for Automatic Depressurization System (ADS) valves after actuation and reactor depressurization has occurred.

(3) RO Question 63 / SRO Question 61: Three of three RO's and one of four SRO's missed the question. The question required candidates to recognize plant conditions for single loop operation, and calculate the required 'Flow Biased Neutron Flux - High' setpoint using Technical Specification Table 2.2.A-1 from a set of reference material provided.

(4) SRO Question 52: Three of four SRO's missed the question. The question asked for the unit load requirement which limits personnel entries to Nitrogen -16 affected areas.

(2) Administrative Job Performance Measures / Questions

All applicants demonstrated satisfactory performance of administrative tasks and response to questions. Individual weaknesses, none of which were generic, were identified and documented by examiners.

(3) Simulator and Inplant Walkthrough JPMs / Questions

Overall JPM performance by applicants was satisfactory. Examiners identified generic areas of lack of knowledge, failing to meet licensee's expectations and poor procedural adherence:

- A JPM question asked for two primary safety concerns of starting an idle reactor recirculation pump. Applicants demonstrated good knowledge of the reactivity effects, but were deficient about knowledge of thermal effects.

- A JPM question identified plant operation at 100% rated thermal power with all turbine controls in their normal power alignment, except for the Electro-hydraulic Governor Control which was not in operation. The question asked what main turbine valves would go closed after a Power Load Unbalance. Applicants demonstrated unfamiliarity with the main turbine steam bypass and pressure control based on a majority of responses that only identified the turbine control valves, but did not mention the turbine combined intermediate valves.

- A JPM task required applicants to restore offsite electrical power to Bus 24-1 and electrically unload the Unit 2 emergency diesel generator from service in accordance with DOP 6500-09, Bus 24-1 TO 34-1 Tie Breaker Operation Utilizing U2(3) D/G. Although a majority of applicants satisfactorily performed the task, two applicants caused the tie breaker to trip on reverse power. Their actions were contrary to plant procedures and licensee's expectations.

(4) Dynamic Simulator Examination

Overall, applicants demonstrated satisfactory performance when implementing and directing Emergency Operating Procedure (EOP) actions during dynamic scenarios. They effectively used three-way communications, demonstrated good command and control, and conducted informative crew briefings that contributed to good teamwork; however, some individual performance was inconsistent. NRC evaluators made the following observations of individual applicants:

Inattention to detail:

- during a normal evolution involving a control rod withdrawal to establish 100% rod control line, an applicant used continuous rod withdrawal and withdrew a rod past its target position.

- during a normal evolution involving torus pump down due to high water level, an applicant started the operation and walked away for seven minutes until alerted to low torus water level by an annunciator.

- during an abnormal event involving loss of an electrical bus, an applicant initially mis-diagnosed the lost bus, although the applicant corrected the diagnosis some time later after reviewing panels.

- during a major transient involving an off gas explosion with release, two applicants identified the event, but failed to order evacuation of the turbine building.

Failure to follow procedures:

- during a major transient involving a primary system discharging into the reactor building, an applicant directed an Emergency Depressurization after identifying one (vice the procedure required two) area exceeded the Max Safe Level.

- during an abnormal event involving loss of an electrical bus, an applicant failed to reposition two major breakers in accordance with the procedure until they were identified by another operator.

- during a major transient involving decreasing reactor water level, an applicant did not monitor critical parameters in a timely manner, and failed to enter DEOP 100 until level was well below the entry condition.

(5) Loss of Simulator Event Malfunction

Examiners identified a weakness in the licensee's setup of repeated scenarios. The simulator did not have a switch check feature, and the licensee did not use a checkoff sheet to verify switch positions. The licensee failed to verify re-position of a feedwater reactor water level select switch from the initial scenario run. Consequently, when the scenario was repeated for a second crew of applicants, the instrument malfunction that was dependant on the select switch position did not occur and an applicant evaluation was lost. As a result, examiners had to create and validate another instrument event to run after completion of the scenario set in order to obtain a complete applicant evaluation.

c. Conclusions

Applicants were well prepared to take the written examination as indicated by the high average score of 89%. Applicants were well prepared for the administrative portion of the operating examination as evidenced by no failures and lack of identification of generic weaknesses. Overall JPM performance by applicants was satisfactory; however, examiners identified areas of lack of knowledge, failing to meet licensee's expectations, and procedural adherence. Overall operator performance during scenarios was satisfactory although examiners identified instances of lack of attention to detail and procedural adherence. One simulator event malfunction was lost due to lack of attention to detail by the licensee.

O5.4 Examination Security

a. Scope (NUREG-1021)

NRC examiners maintained possession of all examination material hard copies during development, validation and administration as prescribed by NUREG 1021, "Operator Licensing Examiner Standards," Revision 7, Supplement 1. The licensee controlled reference documents, simulator security, and personnel security agreements as prescribed by their Nuclear Training Administrative Policy (NTAP), Revision 0.

b. Observations and Findings

The licensee's NTAP procedure required posting and locking simulator doors, breaking simulator communication links to remote screens, controlled access to reference material and controlled instructor involvement in training with applicants. The examiners noted the licensee's examination preparation room had a dedicated computer, combination filing cabinet, and keylock entrance door. Commencing the first day of onsite NRC examination validation, the licensee implemented an additional security measure to clearly distinguish between license applicants and personnel signed onto the security agreement. Individuals in each group were required to wear colored badges. Applicants wore red badges with white lettering, while security agreement personnel wore a reverse color scheme badge. Security agreement personnel also carried pocket sized laminated cards describing their examination security responsibilities.

c. Conclusions

There were no breaches of examination security identified during development, validation and administration. The licensee's innovation of wearing badges was considered a strength in their examination security program. NRC examiners identified two potential weaknesses with the licensee's examination preparation room security. Duplicate keys to the entrance door and knowledge of filing cabinet combination numbers become difficult to control over time due to personnel changes. The licensee was urged to maintain strict control of these items or develop an alternate method to ensure examination security in this area.

O5.5 Simulator Fidelity

a. Examination Scope (NUREG-1021)

NRC examiners observed performance of the plant specific simulator during job performance measure walkthroughs and dynamic scenario sets using guidance contained in NUREG-1021, "Operator Licensing Examiner Standards," Revision 7,

Supplement 1. This fidelity review was conducted in relation to simulation tasks performed on the plant specific simulator during the examination process.

b. Observations and Findings

A number of fidelity problems were observed by examiners during validation and administration of the examination and are described in the Simulator Fidelity Report, Enclosure 6 of this report. Prior to starting the first day of examination scenarios the licensee identified the simulator rod worth minimizer (RWM) had become inoperable. Since RWM operability did not affect the initial scenario events, the examination continued and the RWM was identified to the applicants as being out of service during their turnover. The RWM was returned to operable by the start of the second scenario.

c. Conclusions

The plant specific simulator was able to mimic major plant parameters during a variety of malfunctions and plant conditions. All of the examiner observed fidelity issues had been previously identified by the licensee. However, examiners observed the number and type of fidelity issues appear to be increasing when compared to the previous NRC license examination.

V. Management Meetings

X1 Exit Meeting Summary

NRC examiners presented the team's observations and findings to members of the licensee's management on August 6, 1997. The licensee acknowledged the findings presented. No proprietary information was identified during the examination or at the exit meeting.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

E. Carroll, Regulatory Assurance NRC Coordinator
P. DiGiovanna, PWR Operations Training Supervisor
T. Eason, Operations Training Supervisor
R. Holbrook, Training Manager
S. Kuczynski, Shift Operations Supervisor
T. Riley, Regulatory Assurance Supervisor
R. Sitts, Assistant Operations Staff Supervisor
R. Weidner, ISEG Engineering

NRC

B. Dickson, Resident Inspector
D. Roth, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-237/97304-01; 50-249/97304-01	VIO	Failure to comply with the requirements of DAP 21-09.

Closed

50-237/97304-01; 50-249/97304-01	VIO	Failure to comply with the requirements of DAP 21-09.

Enclosure 3

WRITTEN EXAMINATIONS AND ANSWER KEYS (SRO/RO)

Enclosure 4

FACILITY COMMENTS

NRC RESOLUTION OF FACILITY COMMENTS

RO Examination Question 85:

NRC Resolution: Comment was accepted, the original answer on the answer key was a typographical error. Answer key changed from 'b' to 'a'.

Enclosure 6

SIMULATION FACILITY REPORT

Facility Licensee: Dresden Station, Units 1 & 2

Facility Licensee Docket No: 50-237, 50-249

Operating Tests Validated: July 14 - 18, 1997; Administered: July 28 - August 1, 1997

The following documents observations made by the NRC examination team during the initial license examination validation and administration. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance 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 tests, the following items were observed:

ITEM	DESCRIPTION
RWM	Hardware problem. RWM became inoperable after simulator placed in RUN. Same type of problem documented in report 94-01(OL).
RRMG temperature recorder indication	RRP speed controller indication When RRMG cooling water support system failed, high temperature alarms annunciate for motor or generator windings (902-4 E-4), but recorder (TR 2-262-19A and B) points 1 and 2 don't change.
Main Turbine control valve #1	Jumpered in plant but not in simulator.
Main Turbine vacuum	Only decreases after MSIVs shut. With loss of SJAEs, continued to stay normal for rest of scenario (30 minutes).
RRP 2A(2B) #1 seal leakoff alarm	Didn't receive alarm 902-4 F-3 (F-7), 2A (2B) RECIRC PP #2 SEAL LEAKOFF, when exceed setpoint of .25 gpm.
CRD accumulator alarms	No accumulator alarms received after losing CRD for 20 minutes.
AGAF alarms	Process computer alarmed continuously.
Rod K-8	Full core display indicating '4-' vice '48'.
RRP start parameters	Did not get nominal motor parameter values listed in procedure, DOP 0202-1, RESTARTING A RRP. When starting second RRP, the % speed controller indication would only go to a minimum of '29%' vice the procedure required value of '28%' (DOP 0202-01, RESTARTING A RRP, step G.10)

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
REACTOR OPERATOR LICENSE
REGION 3

CANDIDATE'S NAME: **MASTER EXAMINATION**

FACILITY: Dresden 2 & 3

REACTOR TYPE: BWR-GE3

DATE ADMINISTERED: August 6, 1997

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
100.0	FINAL GRADE	___ %	TOTALS

All work done on this examination is my own. I have neither given nor received aid.

Candidate's Signature

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. 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.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

ANSWER KEY

MASTER COPY

MULTIPLE CHOICE

001	a	023	b
002	d	024	b
003	b	025	b
004	b	026	d
005	b	027	a
006	a	028	d
007	c	029	a
008	d	030	a
009	d	031	a
010	c	032	a
011	d	033	a
012	c	034	b
013	d	035	a
014	a	036	a
015	c	037	a
016	a	038	a
017	a	039	c
018	d	040	d
019	a	041	d
020	a	042	b
021	a	043	a
022	b	044	b
		045	b

ANSWER KEY

046	a
047	d
048	b
049	b
050	a
051	b
052	b
053	b
054	d
055	b
056	d
057	b
058	d
059	d
060	b
061	c
062	b
063	a
064	b
065	c
066	c
067	b
068	b
069	a
070	c
071	c
072	a
073	b
074	b
075	c
076	c
077	c
078	c
079	c
080	a
081	b
082	b
083	c
084	b
085	~~b~~ a JPEllis 8/18/97 typo - no technical change
086	a
087	b
088	b
089	a
090	c
091	b

ANSWER KEY

092	c
093	d
094	b
095	a
096	b
097	d
098	d
099	c
100	a

(********** END OF EXAMINATION **********)

Master Copy

QUESTION: 001 (1.00)

In accordance with DAP 7-27, which ONE of the following is the PRIMARY method to perform an independent verification on a manually throttled valve?

Assume that the valve is installed in a system with a local flow indication controlled by the valve AND the valve has a rising stem.

a. Observing the initial valve operator's action in positioning the throttled valve.

b. Observing flow indication through the throttled valve's system during system lineup.

c. Performing a second valve operation to verify position.

d. Performing an independent visual check of the valve position by comparing the actual valve position with the required valve position.

8/6/97

Start time: 0802
Stop time: 1202

QUESTION: 002 (1.00)

Unit 2 is shutdown after a scram. RPV pressure is 700 psig AND reactor water level can NOT be determined. HPCI is injecting into the vessel. Torus water level is going down and is presently 11.5 feet.

Under these conditions HPCI must be secured because . . .
(Select ONE of the following)

a. HPCI speed will be less than 2000 rpm.

b. HPCI suction line will become uncovered.

c. HPCI system leak is lowering Torus level.

d. HPCI exhaust discharge will pressurize the Torus.

QUESTION: 003 (1.00)

Per the Technical Specifications, the Rod Worth Minimizer (RWM) is NOT required to be operable above 20% reactor power because at higher power levels . . .
(Select ONE of the following)

a. the higher power production in the upper core region minimizes control rod worth.

b. increased voiding of the core limits the reactivity addition rate of a dropped control rod.

c. the reactivity change associated with a dropped control rod becomes insignificant when compared with the overall core reactivity.

d. reactivity changes are less pronounced for a dropped control rod because the EHC system maintains RCS pressure essentially constant.

QUESTION: 004 (1.00)

Fuel loading is in progress. The first three fuel assemblies of a cell are fully seated in the correct core locations and are in the correct orientation. The fourth assembly loaded in this cell is inadvertently oriented 180 degrees out from its correct position, but is fully seated into its correct core location.

If the reactor was operated in this condition, then the incorrect fuel assembly orientation will affect . . . (Select ONE of the following)

a. TIP operation.

b. proper control rod blade operation.

c. core flow through the fuel assembly.

d. core bypass flow to cool the control rod blade AND incore instrumentation.

QUESTION: 005 (1.00)

What is the purpose of a Turbine trip caused by a Main Generator 86 device trip?
(Select ONE of the following)

a. Prevent stator overheating.

b. Provide overspeed protection.

c. Provide reverse power protection.

d. Prevent last stage bucket overheating.

QUESTION: 006 (1.00)

The decision to maintain primary containment integrity without regard to the status of maintaining adequate core cooling must be made when . . .
(Select ONE of the following)

a. Torus bottom pressure can not be maintained below 65 psig.

b. Drywell temperature can not be maintained below 281 degrees F.

c. Torus water level AND RPV pressure can not be maintained below the ADSV Tail Pipe Level Limit.

d. Torus bottom pressure can not be maintained below the Pressure Suppression Pressure for the present Torus water level.

QUESTION: 007 (1.00)

Given the following plant conditions:

- L1220 indicates 230 amps
- L1221 indicates 250 amps
- L0302 indicates 255 amps
- 345 kV BT 4-8 CB is open
- All ring Bus breakers are closed
- Main Generator is at 750 MWe AND 50 MVARs

Do to an OOS error, the amplidyne is inadvertently de-energized, causing the voltage regulator to shift to Manual. Based on these conditions, what is the actions, if any, that must be taken?
(Select ONE of the following)

a. Operation in this condition can continue indefinitely. No reports or actions are required.

b. Inform BPO that the voltage regulator is in Manual, then continue operation in the present region.

c. Inform BPO that the voltage regulator is in Manual. Raise reactive loading to at least 90 MVARs.

d. Inform BPO that the voltage regulator is in Manual. Reactive loading will automatically raise to approximately 90 MVARs.

QUESTION: 008 (1.00)

The following conditions exist on Unit 3:

- a reactor startup is in progress
- all IRM's onscale on Range 2
- all four SRM's are being withdrawn by the operator
- reactor power is held constant during SRM withdrawal
- SRM 21 indicates 111 cps
- SRM 22 indicates 96 cps
- SRM 23 indicates 115 cps
- SRM 24 indicates 109 cps

Which ONE of the following actions will occur?

a. All SRM detector drive motors will de-energize.

b. A SRM "HI/INOP" rod withdraw block will be generated.

c. An IRM DOWNSCALE rod withdraw block will be generated.

d. A SRM "Wrong Detector Position" rod withdraw block will be generated.

QUESTION: 009 (1.00)

In accordance with DOP 1600-22, Drywell Entry (Initial or At Power), which ONE of the following is the MAXIMUM reactor power level at which a Drywell entry can be performed?

a. 5%

b. 10%

c. 25%

d. 40%

QUESTION: 010 (1.00)

In accordance with DAP 07-05, Operating Charts, Logs and Records, which ONE of the following items shall be entered into the Degraded Equipment Log?

a. Technical Specification equipment that activates an LCO.

b. Equipment that has been modified by a temporary alteration.

c. Instrumentation that will hamper implementation of Dresden Emergency Operating Procedures.

d. DATR equipment that is defined by the Probabilistic Risk Analysis (PRA) study as "critical" equipment.

QUESTION: 011 (1.00)

The Unit 2 Service Air to Instrument Air cross-tie valve automatically opened after the Main Instrument Air receiver pressure dropped.

After restoration of Main Instrument Air receiver pressure, the cross-tie valves . . .
(Select ONE of the following)

a. will auto close.

b. must be manually cranked closed.

c. will close after depressing the reset pushbutton on the side of the valve.

d. will close after depressing the reset pushbutton next to the Main Instrument Air receiver.

QUESTION: 012 (1.00)

The following alarms are received:

Panel 902(3)-4, H-18, DRYWELL FLOOR DRN SUMP LVL HI-HI
A-17, DRYWELL EQUIP SUMP LVL HI-HI

A decision is made to enter DOA 0040-01, Slow Leak.

Which ONE of the following items is an Immediate Operator Action?

a. Evacuate all personnel from Containment.

b. Initiate a load reduction to stabilize EHC response.

c. Maintain water level with the Feedwater Control system.

d. If Reactor Building exhaust radiation level exceeds 2 mrem/hr, then initiate Reactor Building Ventilation System isolation AND manually initiate SBGT.

QUESTION: 013 (1.00)

If Bus 24-1 becomes de-energized AND is NOT re-energized, which ONE of the following describes the response of MCC 28-7 AND MCC 29-7?

a. MCC 28-7 AND MCC 29-7 remain energized without losing power.

b. MCC 28-7 will de-energize AND MCC 29-7 will remain energized.

c. MCC 28-7 will remain energized AND MCC 29-7 will de-energize.

d. MCC 28-7 AND MCC 29-7 lose power AND re-energize after a time delay.

QUESTION: 014 (1.00)

Given the following plant conditions:

- 100% steady state power on Unit 2 for an extended period of time

Then, the following events occur on Unit 2:

- a full MSIV isolation AND associated reactor scram occurs
- an ERV does NOT completely reseat following operation

The Shift Manager decides to initiate the MAXIMUM allowable Technical Specification cooldown in order to reach cold shutdown. Assuming that a uniform cooldown is initiated at the MAXIMUM allowable cooldown rate from the MAXIMUM allowable LCO steam dome pressure, how many minutes will it take to reach the point that Shutdown Cooling can be placed into service?
(Select ONE of the following)

a. 118 minutes

b. 114 minutes

c. 110 minutes

d. 105 minutes

QUESTION: 015 (1.00)

Given the following plant conditions for Unit 2:

- One control rod is withdrawn AND immovable
- The plant has been operating in Single Loop for 2 days per DGP-03-03, Single Recirculation Loop Operation.

What is the current limit on MCPR?
(Select ONE of the following)

a. 1.07

b. 1.08

c. 1.09

d. 1.10

QUESTION: 016 (1.00)

Unit 3 is at 100% reactor power when a total loss of service water occurs.

What is the effect on a running train of the SBGT system? (Select ONE of the following)

a. Shortened retention of iodine in charcoal.

b. Standby SBGT train auto-starts on low flow.

c. SBGT system damper 2/3-7509 (Cross-Tie damper) fails closed.

d. SBGT system damper 3-7503 (Reactor Building Inlet damper) fails open.

QUESTION: 017 (1.00)

Core flow is NOT directly proportional to recirculation loop drive flow.

Which ONE of the following describes the reason?

a. As reactor power rises, the void fraction increases the hydraulic resistance, thus causes a reduction in core flow.

b. As reactor power rises, the overall change in the boiling boundary increases resistance to flow, thus causing a rise in core flow.

c. As reactor power rises, power production is pushed lower AND to the outside of the core, thus causing a reduction in core flow in these regions AND a reduction in overall core flow.

d. Regardless of reactor power level, core flow is a function of the dp across the flow orifices, thus any other hydro-dynamic effects are insignificant with respect to this dp.

QUESTION: 018 (1.00)

Unit 3 is shutdown with refueling operations in progress when the "REFUEL FLOOR RAD HI" annunciator alarm is received. The Refuel Floor Lo Range ARM is indicating that the refueling floor radiation level is 35 mrem/hr.

Which ONE of the following automatic actions will occur?

a. The SBGT system will automatically start.

b. Unit 2 AND 3 Reactor Building Ventilation isolates.

c. Upward movement of the main refueling hoist will be inhibited.

d. Upward movement of the main hoist on the Reactor Building Overhead Crane is inhibited.

QUESTION: 019 (1.00)

A transient on Unit 3 has resulted in a release of radioactive contaminants to the Reactor Building. Current plant conditions are as follows:

- Reactor Building Ventilation Exhaust Radiation indicates 7 mr/hr
- Reactor Building to atmosphere differential pressure has risen to +0.2 inches of water, and is steady
- Isolation Condenser area temperature = 165 degrees F AND rising
- the other Reactor Building temperatures are near normal

Which ONE of the following will occur?

a. SBGT system will autostart.

b. Reactor Building Ventilation fans will trip on overpressure.

c. Control room ventilation will switch to emergency recirc mode.

d. Reactor Building blow-out panels will blow out to relieve overpressure condition.

QUESTION: 020 (1.00)

When raising Reactor Cavity level during flooding for refueling operations, what action is required when level is 2 feet above the vessel flange?
(Select ONE of the following)

a. Check for any leaks on the bellows seals.

b. Insert the removable gate between the reactor well AND the fuel storage pool.

c. Perform the surveillance on refuel interlocks (DOS 800-1), AND allow Health Physics to take Radiation Surveys.

d. Remove the blocks on the dryer-separator pit to allow flooding of the pit along with the reactor well.

QUESTION: 021 (1.00)

Given the following plant conditions:

- Unit 3 is at 100% reactor power
- the process computer is out of service for the next 30 to 40 minutes for periodic system maintenance
- a partial loss of power occurs which causes a reactor scram AND de-energizes the full core display AND rod worth minimizer CRT

In accordance with DGP 02-03, which ONE of the following Immediate Operator Actions is the operator required to perform?

a. Initiate ARI.

b. Verify APRM's downscale.

c. Insert the SRMs AND verify that a negative period exists.

d. Insert the IRMs AND verify that reactor power is decreasing with a negative period.

QUESTION: 022 (1.00)

Given the following plant conditions:

- Unit 2 is at 100% reactor power
- the Feed Water Level Control System (FWLCS) is in Master Auto with normal reactor water level setpoint
- a narrow range (NR) level transmitter, NOT selected on RX LVL SELECT, fails downscale

The automatic plant response will be . . .
(Select ONE of the following)

a. HPCI receives a start signal.

b. FWLCS will maintain reactor water level at the normal reactor water level setpoint.

c. FWLCS will overfeed the reactor vessel AND a high reactor water level reactor trip will occur.

d. feedwater level setpoint Auto Runback occurs reducing the reactor water level setpoint to 1/2 of original value.

QUESTION: 023 (1.00)

The following plant conditions exist on Unit 3:

- Reactor shutdown, all rods in
- Drywell pressure is 2.5 psig
- Drywell temperature is 405 degrees F
- Reactor pressure is 50 psig
- Reactor water level is -70 inches

Which ONE of the following is the reason for exiting DEOP 0100, "Reactor Control", AND entering DEOP 0400-01, "RPV Flooding"?

a. It is the alternate method to feed the RPV to maintain the RPV water level above -70 inches.

b. The RPV water level instruments can not be relied upon to assure adequate core cooling.

c. RPV pressure is low enough to allow the use of Core Spray to submerge the core to maintain adequate core cooling.

d. Drywell temperature is above the saturation curve which requires flooding of the RPV to quench reactor pressure.

QUESTION: 024 (1.00)

An unisolable reactor coolant leak on Unit 2 has resulted in a reactor scram AND a rapid rise in drywell pressure. The following additional plant conditions exist:

- Reactor building ventilation radiation is at 3 mr/hr
- Reactor pressure is 775 psig
- Drywell temperature is 155 degrees F
- All rods are inserted
- Drywell pressure is 3 psig AND rising
- HPCI is maintaining reactor water level at 10 inches

Which ONE of the following DEOP(s) are required to be executed?

a. DEOP 0200-01 (Primary Containment Control) only.

b. DEOP 0100 (Reactor Control) AND DEOP 0200-01 (Primary Containment Control).

c. DEOP 0100 (Reactor Control), DEOP 0200-01 (Primary Containment Control) AND DEOP 300-01 (Secondary Containment Control).

d. DEOP 0200-01 (Primary Containment Control) AND DEOP 300-01 (Secondary Containment Control).

QUESTION: 025 (1.00)

Which ONE of the following components is designed to limit Xenon AND Krypton levels that reach the Off Gas Charcoal Adsorber?

a. recombiner.

b. holdup pipe.

c. cooler condenser.

d. charcoal absorber prefilter.

QUESTION: 026 (1.00)

Torus cooling has just been placed in service on Unit 3.

Which ONE of the following describes the purpose of the LPCI heat exchanger differential pressure controller?

a. The CCSW heat exchanger INLET valve is throttled to maintain CCSW pressure 20 psig LOWER than LPCI system pressure to prevent contaminating the torus.

b. The LPCI heat exchanger INLET valve is throttled to maintain LPCI system pressure 20 psig HIGHER than CCSW pressure to prevent contaminating the torus.

c. The LPCI heat exchanger OUTLET valve is throttled to maintain LPCI system pressure 20 psig LOWER than CCSW pressure to prevent inadvertent release of radioactive effluents.

d. The CCSW heat exchanger OUTLET valve is throttled to maintain CCSW pressure 20 psig HIGHER than LPCI system pressure to prevent inadvertent release of radioactive effluents.

QUESTION: 027 (1.00)

Unit 2 is operating at 100% rated power when a spurious scram on RPS Channel A occurs. The operator observes the following:

- the indicating lights for scram solenoid groups A1, A2, AND A3 are ILLUMINATED.
- the indicating light for scram solenoid group A4 is EXTINGUISHED.

Which ONE of the following states the percentage of control rods that will insert into the core, if the operator inadvertently depresses the RPS Channel B manual scram button?

a. 25%

b. 50%

c. 75%

d. 100%

QUESTION: 028 (1.00)

The Scram Discharge Instrument Volume High Level Scram signal is NORMALLY bypassed . . .
(Select ONE of the following)

a. by installation of electrical jumpers.

b. automatically, 10 seconds after placing the mode switch to SHUTDOWN.

c. by placing the DISCH VOL HI WTR BYPASS keylock switch in the BYPASS position while the Reactor Mode switch is in STARTUP.

d. by placing the DISCH VOL HI WTR BYPASS keylock switch in the BYPASS position while the Reactor Mode switch is in REFUEL OR SHUTDOWN.

QUESTION: 029 (1.00)

Select the potential detrimental effects of a hydrogen-oxygen burn in the primary containment from the following:

1. Uncontrolled release of radioactivity.
2. Pressures exceeding the structural capability of the Drywell.
3. Damage to plant equipment required for the shutdown of the reactor plant.
4. Pressures excursions exceeding the capacity of the Torus to Reactor Building vacuum breakers.

Select ONE of the following combinations:

a. 1, 2, 3

b. 2, 3, 4

c. 1, 3, 4

d. 1, 2, 4

QUESTION: 030 (1.00)

Given the following plant conditions on Unit 2:

- Reactor power is 100%
- all reactor water level instruments are Operable

It has been observed that over the last couple of shifts that the medium range Yarway reactor water level instruments on the 2202-5 and 2206 racks are drifting high. All control room indications are steady.

If the drifting continues it will ....
(Select ONE of the following)

a. have no effect on RPS.

b. result in the high reactor water level scram occurring at a level higher than designed.

c. result in the low reactor water level scram occurring at a higher level than designed.

d. result in the low reactor water level Group 1 isolation occurring at a lower level than designed.

QUESTION: 031 (1.00)

Given the following plant conditions:

- Unit 3 is at 100% reactor power
- all equipment is operable except for the A channel of medium range (MR) reactor water level instrumentation, MR transmitter LT 3-263-57A, has failed downscale

The following alarm is received:

903-5, C-13, CHANNEL A/B RPV PRESS HI-HI

(which is due to the failure of RPV Pressure transmitter 3-263-55D upscale)

What will the plant response be?
(Select ONE of the following)

a. Full reactor scram.

b. Trips the Recirc MG Set drive motor breakers.

c. Only Group I, II, III AND IV "B" scram pilot solenoid valves will de-energize.

d. Only Group I, II, III AND IV "A" scram pilot solenoid valves will de-energize.

QUESTION: 032 (1.00)

The design of the scram discharge volume (SDV) high-high level input into RPS assures that . . .
(Select ONE of the following)

a. the SDV can accommodate water discharged after a scram.

b. an exhaust path for drive water flow is provided following a scram.

c. operation of the CRD system and associated piping will not interfere with other safety related equipment operation following a scram.

d. an extension to the reactor coolant system pressure boundary is provided in the event that a drive water return line is leaking reactor water back through the CRD mechanism.

QUESTION: 033 (1.00)

What will cause the Medium Range Level indication to fail full downscale?
(Select ONE of the following)

a. A break occurs in the variable leg.

b. A break occurs in the reference leg.

c. A bellows leak occurs in the DP cell.

d. A rapid depressurization of the reactor occurs.

QUESTION: 034 (1.00)

During normal 100% reactor power operation, it is noted that Secondary Containment differential pressure is at 0 inches of water AND going up.

Which ONE of the following states the adverse consequences that will occur if this situation is allowed to continue?

a. The torus could be overpressurized in the event of a Loss of Coolant Accident (LOCA).

b. In the event of a design basis Loss of Coolant Accident (LOCA) an unmonitored release could occur.

c. The differential pressure across access doors would limit the access to the Secondary Containment.

d. Higher inleakage to the Secondary Containment would occur due to the excessive load being placed on the Reactor Building vent and purge system.

QUESTION: 035 (1.00)

Unit 3 is operating at 100% reactor power when the "3A RECIRC PP CONT SEAL FLOW HI" computer point alarm is received.

Other indications on the 3A recirc pump are as follows:

- No. 1 seal pressure 990 psig AND steady
- No. 2 seal pressure 785 psig AND rising

Which ONE of the following is the cause of the alarm?

a. A failure of the No. 1 seal.

b. A failure of the No. 2 seal.

c. Plugging of the No. 1 internal restricting orifice.

d. Plugging of the No. 2 internal restricting orifice.

QUESTION: 036 (1.00)

Unit 2 is conducting a plant startup at approximately 35% reactor power when the "RPIS SYS INOP" annunciator is received.

The NSO must . . .
(Select ONE of the following)

a. stop any power changes in progress AND investigate the failure.

b. attempt to move the selected control rod to verify failure of the RPIS.

c. return the reactor to hot shutdown within 24 hours by normal reactor shutdown.

d. continue raising reactor power as instructed AND call Instrument Maintenance for assistance.

QUESTION: 037 (1.00)

Given the following plant conditions:

- A LOCA has occurred at 100% power on Unit 2

If the LPCI Loop Select Logic senses a break in the "B" recirc loop, COINCIDENT with 2 psig in the Drywell, which ONE of the following does NOT occur?

a. The "A" recirc loop suction valve shuts.

b. The system "B" LPCI injection valves shut.

c. The "A" recirc loop discharge valve shuts.

d. The "B" recirc loop discharge valve remains open.

QUESTION: 038 (1.00)

Which ONE of the following is the reason for stabilizing reactor pressure below 1060 psig following a reactor scram and subsequent entry into DEOP 100, Reactor Control?

a. To avoid ADS valves lifting.

b. To permit RPV pressure control with the bypass valves.

c. To maintain cooldown rate below 100 degrees F per hour.

d. To ensure proper operation of reactor water level instruments.

QUESTION: 039 (1.00)

You have been directed to pull a fuse for an Out-Of-Service. There is NO local fuse label listing type or amperage of the fuse. To verify the correct fuse was installed, you must.... (Select ONE of the following)

a. check the electrical print for the proper type fuse.

b. check the System Electrical Checklists to verify the fuse information.

c. compare the pulled fuse to the specifications in EWCS.

d. inform the Label Coordinator of the missing label AND have him verify the fuse.

QUESTION: 040 (1.00)

Fuel movements are being performed on Unit 2. The Unit 2 NSO and Fuel Handlers cannot agree as to which move is to be performed next.

According to Unit 2 Master Refueling Procedure, DFP 800-1, which ONE of the following personnel is to be contacted for further assistance?

a. Shift Manager.

b. Fuel Handling Supervisor.

c. Nuclear Materials Custodian.

d. Control Room Nuclear Observer.

QUESTION: 041 (1.00)

Given the following plant conditions on Unit 2:

- An ATWS from a CRD hydraulic lock is in progress.
- ADS valves are being cycled to control reactor pressure.
- Torus temperature is 114 degrees F.
- Drywell pressure is 3.5 psig.
- Current reactor power is 10%

Which ONE of the following is the prescribed reactor water level band?
(Select ONE of the following)

a. +8 AND +48 inches

b. -143 AND +48 inches

c. -143 inches AND a level to which it was lowered.

d. -164 inches AND a level to which it was lowered.

QUESTION: 042 (1.00)

Which ONE of the following is an indication of a failed jet pump?

a. A rise in core thermal power.

b. A drop in core plate differential pressure.

c. A rise in main generator electrical output.

d. A drop in individual recirc pump flow for a given speed.

QUESTION: 043 (1.00)

Unit 2 is operating at approximately 85% reactor power when a reduction in condenser vacuum commences. In accordance with the Immediate Operator actions of DOA 3300-02, you must . . .

(Select ONE of the following)

a. trip hydrogen addition.

b. start the mechanical vacuum pump.

c. raise Gland Seal Steam pressure to 10 psig.

d. raise Steam Jet Air Ejector steam supply pressure to 150 psig.

QUESTION: 044 (1.00)

Which ONE of the following will occur if all 24/48 VDC is lost on Unit 3?

a. ATS panel trouble alarm.

b. Will experience a half scram.

c. Core Spray system minimum flow valves will not operate.

d. Wide range reactor water level indication becomes inoperable.

QUESTION: 045 (1.00)

Which ONE of the following will occur if the Unit 3 Essential Service System Bus is lost?

a. Complete Group II AND III isolations will occur.

b. Reactor feed pump minimum flow valves fail open.

c. Core spray system minimum flow valve will not operate.

d. Wide range reactor water level indication becomes inoperable.

QUESTION: 046 (1.00)

An unisolable steam leak in the turbine building has resulted in a high radioactivity level in the turbine building. The Unit Supervisor wishes to monitor the release rate to the environment.

Which ONE of the following will accomplish that?

a. Starting Turbine Building Ventilation if shutdown.

b. Securing Turbine Building Ventilation if operating.

c. Starting SBGT AND aligning it to the Turbine Building.

d. Starting Reactor Building Ventilation AND aligning it to the Turbine Building.

QUESTION: 047 (1.00)

Isolation of a primary system leak is required by DEOP 300-1 AND 300-2, in order to limit radioactive discharge.

Under these conditions, the term "Primary System" refers to any system....
(Select ONE of the following)

a. containing reactor coolant.

b. for which the ASME "N" stamp is issued.

c. connected to the RPV that contains radioactive water.

d. connected to the RPV that has a reduced leak rate if RPV pressure is lowered.

QUESTION: 048 (1.00)

During Unit 2 100% reactor power operation, the following events occur:

- DAN 923-1 C-1, "U2 RBCCW PP TRIP"
- DAN 902-3 A-13, "Drywell Pressure Hi"
- DAN 902-4 G-17, "Drywell Atmosphere Temp Hi"
- Standby RBCCW pump can NOT be started

Sixty seconds later:

- DAN 923-1 D-1, "U2 RBCCW Press Lo"
- Drywell temperature is 170 degree F AND slowly rising
- Drywell pressure is 2.1 psig AND slowly rising

What Immediate Operator Actions must you take?
(Select ONE of the following)

a. Verify reactor scram and/or manually scram the reactor, AND verify automatic isolation of RBCCW.

b. Verify reactor scram, trip Reactor Recirc pumps, AND isolate RBCCW to the Drywell.

c. Trip the Reactor Recirc pumps, verify core flow has dropped to less than 45 Mlbm/hr, AND enter DGP 2-1, "Normal Unit Shutdown" AND DEOP 200-2.

d. Reduce Reactor Recirc pump speeds, verify power/flow region, enter DEOP 100 AND 200-1, AND assess if any plant equipment damage is imminent before manually scramming the reactor.

QUESTION: 049 (1.00)

Unit 3 has lost its CRD pumps. The Shift Manager determined that it will require CRD pump discharge cross-tie operation between Units 2 and 3.

Which ONE of the following is the MAXIMUM CRD pump current that is allowed during this evolution?

a. 30 amps

b. 34 amps

c. 40 amps

d. 44 amps

QUESTION: 050 (1.00)

A Group II Isolation signal exists AND the Containment Vent valves are required to be operated. The reactor Mode Switch is in "SHUTDOWN".

What condition(s) is(are) required to open valves 1601-61 (Torus 2 inch Vent) AND 1601-63 (Vent to SBGT)?
(Select ONE of the following)

a. Vent Isol Signal Bypass switch in the "Torus" position.

b. Vent Isol Signal Bypass switch in the "Drywell" position.

c. Mode Switch in "RUN" AND Vent Isol Signal Bypass switch in the "Torus" position.

d. Mode Switch in "RUN" AND Vent Isol Signal Bypass switch in the "Drywell" position.

QUESTION: 051 (1.00)

A transient has occurred causing an ADS initiation signal AND all ADS valves open as expected. As a result of the transient, the reactor has fully depressurized. Assuming proper operation of the ADS valves for the present plant condition, the operator should expect the SPDS block for the ADS valves to indicate . . .
(Select ONE of the following)

a. "RED" AND "OPEN"

b. "RED" AND "CLOSED"

c. "GREEN" AND "OPEN"

d. "GREEN" AND "CLOSED"

QUESTION: 052 (1.00)

Which ONE of the following explains how ARM instruments are identified as "DEOP" instruments?

a. The labels have a black dot.

b. The instruments have purple labels.

c. The words "DEOP" are found on the label.

d. The instrument identifier is written in red.

QUESTION: 053 (1.00)

As the off-going Unit 2 NSO preparing for turnover, you have completed the Unit 2 Log and reviewed and initialed the surveillance performed during your shift.

Which ONE of the following is NOT required of you for proper shift turnover?

a. Perform a face to face turnover with the oncoming NSO.

b. Discuss applicable DEL items with the Unit Supervisor.

c. Complete appropriate portions of the NSO shift turnover checklist.

d. Remain on the unit until the oncoming NSO has received a satisfactory turnover AND is fully aware of existing conditions.

QUESTION: 054 (1.00)

According to DAP 12-04, the Radiation Protection Manager AND Operations Manager must make all efforts to eliminate power entries into Nitrogen-16 areas.

At what unit load does this first occur?
(Select ONE of the following)

a. 200 megawatts

b. 250 megawatts

c. 300 megawatts

d. 350 megawatts

QUESTION: 055 (1.00)

A licensed operator, as defined by 10CFR55, must meet several requirements to maintain his/her license.

Which ONE of the following is included in these requirements in order to maintain an ACTIVE status of his/her NRC license?

a. The licensee shall have an annual medical examination.

b. Pass a comprehensive requalification written examination AND an annual operating test.

c. Must apply for renewal of license at least 30 days prior to the five year expiration date (based on date of license issuance)

d. The licensee shall actively perform the functions of the licensed position on a minimum of seven 8-hour shifts or five 12-hour shifts per calendar year.

QUESTION: 056 (1.00)

While operating at 100% reactor power, feedwater regulating valve (FWRV) 2A is in service controlling reactor water level. The air line supplying FWRV 2A ruptures AND air is rapidly lost to the operator.

Which ONE of the statements below identifies the response of the FWRV?

a. The valve fails full open, but the speed is limited by the hydraulic damper.

b. The valve fails full closed, but the speed is limited by the hydraulic damper.

c. The valve fails full open immediately since it uses air to close, AND spring pressure to open.

d. The valve would "lock up" in its present position, due to the actuation of the air lock valve.

QUESTION: 057 (1.00)

Unit 3 is at 100% reactor power with feedwater in automatic control when the selected reactor water level transmitter (Narrow Range "A") fails off scale high. The Bailey System indicates "bad quality".

The Control System will ...
(Select ONE of the following)

a. automatically shift to Medium Range "A".

b. automatically shift to Narrow Range "B".

c. lock up at the last valid reactor water level signal sensed from Narrow Range "A".

d. reduce feedwater flow until an alternate reactor water level instrument has been manually selected.

QUESTION: 058 (1.00)

The automatic emergency start of the diesel generator bypasses some of the protective trips for the diesel and/or the diesel generator supply breaker.

Which ONE of the following statements is NOT correct concerning the diesel generator trip bypasses?

a. The ECCS auto start bypasses the diesel engine low lube oil pressure trip.

b. The undervoltage auto start bypasses the diesel generator output breaker overcurrent trip.

c. The ECCS auto start bypasses the diesel generator output breaker reverse power trip.

d. The undervoltage auto start bypasses the diesel generator high differential current trip.

QUESTION: 059 (1.00)

Unit 2 is operating at 75% reactor power with the 'A' Recirc Pump MG set scoop tube locked out for maintenance.

Which ONE of the following describes the preferred Recirc Flow Control System lineup under these conditions?

Controller Positions

	MASTER	'A' RECIRC	'B' RECIRC
a.	automatic	automatic	manual
b.	manual	manual	automatic
c.	manual	automatic	automatic
d.	manual	manual	manual

QUESTION: 060 (1.00)

During a normal control rod insertion, what prevents drive water from recirculating back into the cooling water header? (Select ONE of the following)

a. The system cooling water header pressure is greater than drive water pressure.

b. The cooling water supply header to the hydraulic control unit includes a check valve.

c. The control rod drive stabilizing valve closes on an "Insert" signal thus isolating the cooling water header.

d. The cooling water header isolation valve (104) to the hydraulic control unit closes when "Insert" is selected.

QUESTION: 061 (1.00)

A valid initiation signal for SBGT is received. Train "A" is in PRIMARY and Train "B" is in STANDBY. Under the existing circumstances, which ONE of the following will initiate Train "B" of the SBGT System?

a. Drywell radiation level of 100 R/hr.

b. Refueling Floor radiation level of 100 mr/hr.

c. Low flow condition on Train "A" for 20 seconds.

d. Reactor Building Ventilation radiation level of 4 mr/hr.

QUESTION: 062 (1.00)

The MINIMUM Boron concentration that will bring the reactor from full reactor power to a 3% delta K or more subcritical condition with all rods withdrawn in less than 100 minutes is ...
(Select ONE of the following)

a. 550 ppm

b. 600 ppm

c. 650 ppm

d. 700 ppm

QUESTION: 063 (1.00)

Given the following Unit 2 plant conditions:

- Reactor Power is 1516 MWt
- "A" Recirculation Pump Trip
- Calculated Total Core Flow is 58 million lbs/hr
- Loop B Recirculation Flow is 55%

Which ONE of the following is the setpoint for the APRM Flow Biased Neutron Flux High Trip?

a. 92.8%

b. 93.5%

c. 95.9%

d. 97.8%

QUESTION: 064 (1.00)

While making a tour of the control room back panels, you notice an alarm light on a RBM channel on top of panel 902-37. The light is labeled, "REF APRM DOWNSCALE".

What RBM function is associated with this alarm? (Select ONE of the following)

a. Rod Block.

b. Automatically bypasses RBM.

c. Bypasses Rod Insert Blocks.

d. Indication that the reference APRM is at 40% reactor power.

QUESTION: 065 (1.00)

One of the refueling requirements at Dresden is to check the "one-rod-out" interlock for the Reactor Manual Control System (RMCS) prior to performing refueling operations.

Which ONE of the following statements correctly describes the proper operation of the "one-rod-out" interlock?

a. Selecting a rod which is fully inserted initiates a Rod Block.

b. A selected rod is fully withdrawn (notch 48); at which time a Rod Block is initiated.

c. A selected rod is withdrawn to any position (02 to 48); a second rod can be selected, but not withdrawn.

d. A selected rod can be fully withdrawn (notch 48); a second rod initiates a Rod Block if withdrawn to notch 02.

QUESTION: 066 (1.00)

Which ONE of the following statements correctly describes the operation of the INTERCEPT valves on a Main Turbine Overspeed?

a. A drop in pressure in the Moisture Separator to 105 psig will cause the Intercept Valves to go closed to prevent turbine overspeed.

b. Intercept Valves 2, 4 AND 6 fully close at 103% turbine speed, while valves 1, 3 AND 5 are not closed until 105% turbine speed.

c. As turbine speed is reduced the Intercept valves 1, 3 AND 5 will re-open, while valves 2, 4, AND 6 will ramp open after the first group is 90% open.

d. Once activated by turbine speed, all Intercept Valves remain closed until turbine speed drops to 50% at which time all intercept valves ramp open.

QUESTION: 067 (1.00)

Given the following set of plant conditions:

- Load set is 90%
- Load limit is 100%
- 100% reactor power
- Recirc flow in master manual
- Equalizing header pressure is 950 psig
- Pressure set is 920
- Max combined flow is 105%
- 100% rated core flow

Which ONE of the following would be the current control valve/bypass valve lineup?

a. control valve demand 100%, bypass valve demand 0% open.

b. control valve demand 90%, bypass valve demand 10% open.

c. control valve demand 80%, bypass valve demand 20% open.

d. control valve demand 70%, bypass valve demand 30% open.

QUESTION: 068 (1.00)

Dresden Unit 2 has just experienced a loss of offsite power concurrent with +2.5 psig Drywell pressure. The diesels auto start as designed.

Which ONE of the following states the timed starting sequence for the emergency bus equipment?

a. The diesel generator breaker closes within 10 seconds, then the first LPCI pump starts followed by the core spray pump 5 seconds later followed by the second LPCI pump 5 seconds later.

b. The diesel generator breaker closes within 10 seconds, then the first LPCI pump starts followed by the second LPCI pump 5 seconds later followed by the core spray pump 5 seconds later.

c. When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes, then the first LPCI pump starts followed by the second LPCI pump 5 seconds later followed by the core spray pump 5 seconds later.

d. When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes, then the first low pressure coolant injection (LPCI) pump starts followed by the core spray pump 5 seconds later followed by the second LPCI pump 5 seconds later.

QUESTION: 069 (1.00)

Which ONE of the following will NOT cause a trip of the RPS EPA breakers?

a. Over frequency

b. Under frequency

c. Over voltage

d. Under voltage

QUESTION: 070 (1.00)

The reactor was operating at approximately 5% reactor power following a refueling outage when a scram occurred due to a spurious main steamline isolation.

Which ONE of the following pressure control methods will minimize the inventory loss from the reactor vessel?

a. ADS valves

b. IC vent valve venting.

c. RWCU in Recirculation Mode

d. HPCI in the pressure control mode.

QUESTION: 071 (1.00)

The Intermediate Range Monitors are reading "10" on Range 9.

Which ONE of the following is the percent reactor power correlating to this IRM reading?

a. 0.04%

b. 0.40%

c. 4.0%

d. 40.0%

QUESTION: 072 (1.00)

Given the following methods for initiating LPCI, which ONE of the following does NOT require operation of the 316 Torus keylock switch in order to open Torus cooling valves?

a. Manual LPCI initiation.

b. Drywell Pressure greater than 2 psig.

c. Low Low Water Level AND 8.5 minute time.

d. Reactor Low Pressure AND Low Low Water Level.

QUESTION: 073 (1.00)

Which ONE of the following conditions will result in the closure of the Chimney Isolation Valve?

a. ONE main steam line radiation monitor is alarming high AND ONE offgas radiation monitor has been alarming high for 15 minutes.

b. ONE offgas radiation monitor is alarming downscale AND ONE offgas radiation monitor has been alarming high-high for 15 minutes.

c. ONE main steam line radiation monitor is alarming high-high AND ONE offgas radiation monitor has been alarming high-high for 15 minutes.

d. ONE main steam line radiation monitor has been alarming high for 15 minutes AND ONE main steam line radiation monitor is alarming downscale.

QUESTION: 074 (1.00)

The Cram Arrays for this fuel cycle are as follows:

Array 1 - four rods at position 16
Array 2 - four rods at position 20
Array 3 - four rods at position 24
Array 4 - four rods at position 28

Unit 2 is operating at 95% reactor power. The 2D2 Heater Drains trip with subsequent drop in the feedwater temperature. The NSO reduces recirculation flow. APRM High alarms are received for Channel 1 AND 5.

Which ONE of the following actions can the NSO take in regards to control rod movement?

a. All rods in Cram Array 1 are continuously inserted to position 8.

b. One rod in Cram Array 1 is continuously inserted to position 0.

c. All rods in Cram Arrays are continuously inserted to position 16.

d. Two rods in Cram Array 2 are continuously inserted to position 0.

QUESTION: 075 (1.00)

DEOP 400-1, "RPV Flooding", has been entered during an ATWS condition, and reactor pressure is 450 psig.

What is the MINIMUM number of ADS valves required to be OPEN to ensure ADEQUATE CORE COOLING by submergence or steam cooling?
(Select ONE of the following)

a. 2

b. 3

c. 4

d. 5

QUESTION: 076 (1.00)

The Reactor Building Ventilation AND the SBGT Systems experience a total failure of their respective fans.

Which ONE of the following states the mechanism which occurs as the reactor building pressurizes to prevent structural damage of the Reactor Building?

a. At a pressure of 2.7" water gage, the SBGT outlet damper opens to equalize the pressure to the outside atmosphere.

b. At a pressure of 0.5 psi in the reactor building, the reactor building to torus vacuum breakers relieve pressure to the torus.

c. At a pressure of 0.5 psi in the reactor building, the blowout panels part from the beams on the refueling floor to equalize pressure to the outside atmosphere.

d. At a pressure of 2.2" water gage, the normal ventilation supply AND exhaust dampers open to equalize the pressure to the outside atmosphere.

QUESTION: 077 (1.00)

Which ONE of the following conditions will result in an automatic start of the condensate/condensate booster pump selected to STANDBY?

a. Reactor feed pump suction pressure is 190 psig.

b. Condensate Booster pump discharge pressure is 300 psig.

c. An operating condensate/condensate booster pump trips off.

d. The operating condensate/condensate booster pump suction pressure has been greater than or equal to 105 psig for at least 8 seconds.

QUESTION: 078 (1.00)

Select the ONE reason that DEOP 400-5, Failure to Scram, directs operators to terminate boron injection when there is a reduction in SBLC tank level to 27 percent.

a. Cavitation of the SBLC pumps will be prevented.

b. The hot shutdown boron weight has been injected.

c. The cold shutdown boron weight has been injected.

d. A positive temperature coefficient will be prevented.

QUESTION: 079 (1.00)

In which ONE of the following locations is transmitting with a portable radio PROHIBITED?

a. HPCI Pump Room

b. Radwaste Mezzanine

c. Auxiliary Computer Room

d. Station Blackout Building

QUESTION: 080 (1.00)

The Unit 2 Isolation Condenser automatically initiated during a reactor transient. The Unit Supervisor determines that the Isolation Condenser is NOT required even though an initiation signal is still present.

Which ONE of the following must be performed to assure that the RX INLET ISOL valve (MO 1301-3) will remain CLOSED?

a. Place the valve control switch in PULL-TO-LOCK.

b. Cycle the ISOL COND RESET switch in BOTH directions.

c. Place the RX INLET ISOL VLV HAND/RESET switch in HAND.

d. Hold the valve control switch closed at least 5 seconds after the full close indication has been received.

QUESTION: 081 (1.00)

With Recirculation Pump 2B running, an attempt has been made to start the 2A Recirculation pump. As the attempt is made, the 2A Recirc M-G Set Drive Motor breaker trips and the alarm is received. Further investigation reveals that the Scoop Tube Brake has actuated.

Which ONE of the following has caused the failure of the pump to start?

a. Low feedwater flow.

b. High lube oil temperature.

c. Reactor water level at +5 inches.

d. Pump speed mismatch greater than 10%.

QUESTION: 082 (1.00)

During a surveillance on APRM Channel 4 the Function Switch on the meter unit is placed in the COUNT position. The meter indicates 90%.

Which ONE of the following is the MAXIMUM number of LPRM(s) that have/has been bypassed from input into APRM Channel 4?

a. 1

b. 2

c. 3

d. 4

QUESTION: 083 (1.00)

Which ONE of the following conditions will automatically trip the reactor recirculation motor generator set field breakers after a 9 second time delay?

a. Drywell pressure is 6.8 psig.

b. Reactor pressure is 1190 psig.

c. Reactor water level is -64 inches.

d. ARI manual pushbuttons are armed AND depressed.

QUESTION: 084 (1.00)

During a reactor transient the reactor water level has dropped to -73 inches and drywell pressure is 1.82 psig. The ADS 8.5 minute timer has actuated.

Which ONE of the following will cause the 8.5 minute timer to reset?

a. Depress the TIMER RESET pushbutton.

b. - Reactor water level rises to -42 inches.

c. Place the ADS INHIBIT switch in INHIBIT.

d. Drywell pressure is reduced to 1.45 psig.

QUESTION: 085 (1.00)

Given the following plant conditions on Unit 2:

- An ATWS has occurred.
- Power is lost to MCC 28-1.
- The Unit Supervisor has directed the NSO to initiate SBLC.
- The NSO positions the keylock switch to position "SYS 2 & 1".

Which ONE of the following describes the expected SBLC system operation?

a. Pump 2A OFF; Pump 2B ON; One Squib valve open.

b. Pump 2A ON; Pump 2B OFF; One Squib valve open.

c. Pump 2A ON; Pump 2B ON; Both Squib valves open.

d. Pump 2A OFF; Pump 2B OFF; NO Squib valves open.

QUESTION: 086 (1.00)

Which ONE of the following are DC powered AND must energize to operate?

a. ARI valves.

b. Scram dump valves.

c. Scram pilot solenoid valves.

d. SDV vent AND drain pilot valves.

QUESTION: 087 (1.00)

Reactor startup is in progress on Unit 2 with reactor power at 22%. Prior to startup of the Hydrogen Addition system, which ONE of the following actions is required to prevent an inadvertent reactor scram?

a. Lower the H2/FW RATIO SET setpoint to zero.

b. Place the BYP MSL HI HI RAD SIGNAL ON/OFF switches in the ON position.

c. Place the BYP MSL HI HI RAD SIGNAL ON/OFF switches in the OFF position.

d. Isolate hydrogen injection to all but one operating condensate booster pump.

QUESTION: 088 (1.00)

Unit 3 is operating at 100% reactor power. An Equipment Attendant reports a fire on the Isolation Condenser floor. The Unit Supervisor directs that DOA 10-10, Fire/Explosion, be entered.

Which ONE of the following Immediate Operator Actions are you REQUIRED to perform?

a. Scram the reactor.

b. Initiate the Plant Fire Siren.

c. Immediately implement Plant Assembly.

d. Immediately notify the Coal City Fire Protection District.

QUESTION: 089 (1.00)

Unit 2 is operating at 80% reactor power, with RWCU in it's normal operating lineup.

Which ONE of the following will cause MO 2-1201-07, REACTOR RETURN VALVE, to isolate?

a. Reactor water level is +4 inches.

b. Pressure Regulating Station outlet pressure is 155 psig.

c. Auxiliary Pump Cooler discharge water temperature is 145 deg. F.

d. Non-Regenerative Heat Exchanger outlet temperature is 148 deg. F.

QUESTION: 090 (1.00)

Which ONE of the following will cause Fuel Storage Pool Pump 2A to trip?

a. Undervoltage on Bus 29.

b. Pump suction pressure is 8 psig.

c. Filter inlet pressure is 152 psig.

d. Skimmer Surge Tank water level is 20 inches.

QUESTION: 091 (1.00)

Unit 2 was maintaining reactor water level at +30 inches prior to a loss of coolant accident. 5 seconds following the accident, reactor water level is -2 inches on all level indications and drywell pressure is 2.4 psig.

Which ONE of the following is the reactor water level setpoint signal from the Feedwater Level Control system?

a. 0 inches -

b. +5 inches

c. +15 inches

d. +30 inches

QUESTION: 092 (1.00)

Unit 2 has just shutdown after a 12 month operating period. Shutdown cooling has been placed in service with the 2A AND 2B Shutdown Cooling (SDC) pumps running aligned to the reactor. The reactor water temperature is 214 deg. F. Which ONE of the following actions would be taken if the reactor coolant cooldown rate needed to be raised?

a. Start the 2C SDC pump.

b. Throttle the RBCCW Outlet Valve, MO 2-3704.

c. Open the SDC pump discharge valves on the operating SDC pumps to the desired position.

d. From fully closed, open the SDC pump suction valve to the specified position within 20 to 25 seconds.

QUESTION: 093 (1.00)

On the full core display, during normal full reactor power operations, all 177 rod drift lights come on AND the scram valves indicate closed. You also recognize four (4) scram relay white lights are off.

Which ONE of the following Immediate Operator Actions must be taken?

a. Investigate a power loss from the Instrument Bus.

b. Press the scram buttons AND place the mode switch to SHUTDOWN.

c. Verify control rods have inserted (< 02) using the Rod Worth Minimizer CRT or OD-7.

d. Direct the High Voltage Operator to Auxiliary Electrical Equipment Room to investigate cause of the power loss.

During exam, told the candidates that the "four scram relay lights that are off are all in the same channel."

QUESTION: 094 (1.00)

Unit 2 is operating at 60% reactor power. Plant indications show that condenser vacuum is slowly dropping.

Which ONE of the following can NOT be used to maintain condenser vacuum?

a. Reduce reactor power.

b. Start the Mechanical Vacuum Pump.

c. Start the Standby Circulating Water Pump.

d. Isolate the Hydrogen AND Oxygen injection systems.

QUESTION: 095 (1.00)

Given the following plant conditions on Unit 3:

- Reactor power is 80%
- All ECCS systems are in Standby Readiness
- CRD Pump 3A is in service
- No surveillance testing is in progress at this time

Unit 3 experiences a partial loss of 125VDC power. Computer point C181, "125VDC Reserve Bus 3B-1", indicates Bus failure.

Which ONE of the following is immediately available for emergency core cooling initiated from the control room?

a. HPCI

b. CRD Pump 3B

c. LPCI Pump 3C

d. Core Spray 3B

QUESTION: 096 (1.00)

Which ONE of the following plant conditions requires entry into DEOP 300-2, Radioactive Release Control?

a. Reactor building vent radiation is 6 mr/hr.

b. Offsite release rate has been 20 x ODCM for 30 minutes.

c. Failure of the MSIVs to close on high radiation signal.

d. Two reactor building area radiation levels are above max safe levels.

QUESTION: 097 (1.00)

Unit 3 is operating at 60% rated power with CRD pump 3A tagged out of service for maintenance. A trip of CRD pump 3B results in two control rod "ACCUMULATOR TROUBLE" alarms. The two control rods are NOT at notch 00.

Which ONE of the following is the reason that DOA 0300-01, Control Rod Drive System Failure, directs the operator to immediately scram the reactor?

a. To prevent damage to the control rod drive mechanisms due to overheating.

b. To prevent criticality in a control cell if the two rods, which are both in a three by three rod array, fail to scram.

c. To prevent reactor coolant back leakage into the CRD hydraulic accumulators AND the subsequent high area radiation conditions.

d. To ensure sufficient control rods can be fully inserted to shutdown the reactor before additional CRD problems can occur.

QUESTION: 098 (1.00)

Both units are operating at 100% reactor power when an electrical fault results in a complete loss of Division II 125 VDC power on Unit 2.

Which ONE of the following describes how this failure will affect plant operations?

a. The 2B Recirc Pump will trip.

b. Unit 2 ADS logic will not operate.

c. Unit 2 Isolation Condenser will initiate.

d. Diesel Generator 2 will not be able to start.

QUESTION: 099 (1.00)

The reason that the feed pumps trip on high reactor water level is to prevent ...
(Select ONE of the following)

a. Jet pump damage.

b. Feed pump damage.

c. Main steam line piping AND hanger damage.

d. Reactor Recirculation pump suction nozzle damage.

QUESTION: 100 (1.00)

While at 90% power condenser vacuum is observed to be decreasing.

Which ONE of the following states the expected plant response that will occur if vacuum further decreases without operator action?

a. The turbine will trip after the reactor scrams at 21" Hg vacuum.

b. The turbine will trip at 20" Hg vacuum which will cause a reactor scram.

c. The turbine will trip at 21" Hg vacuum resulting in a generator load reject.

d. The turbine will trip AND the reactor will scram from a turbine control valve fast closure at 20" Hg vacuum

(********** END OF EXAMINATION **********)

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 001 (1.00)

REFERENCE:

1. DAP 7-27, Rev. 13, Independent Verification, p. 9 of 12.

   294001K101

QUESTION: 002 (1.00)

REFERENCE:

1. DLP 206L-S1, Rev. 03, HPCI.

   206000K605

QUESTION: 003 (1.00)

REFERENCE:

1. DLP 201L-S6, Rev. 04, Rod Worth Minimizer.
2. Technical Specification 3.4, page 3/4.3-18, Amend. No. 3/16/88.

   201006K105

QUESTION: 004 (1.00)

REFERENCE:

1. DLP 232L-S1, Rev. 03, Nuclear Fuel, page 16 of 30.

   234000K505

QUESTION: 005 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev. 03, Main Turbine.

   295005K304

QUESTION: 006 (1.00)

REFERENCE:

1. DLP 295L-S2, Rev. 1, Primary Containment Control and Primary Containment Hydrogen Control, DEOP's 200-1 and 200-2.

   295024K101

QUESTION: 007 (1.00)

REFERENCE:

1. DOP 6400-8, Rev. 7, Figure 7.

294001A108

QUESTION: 008 (1.00)

REFERENCE:

1. DLP 215L-S4, Rev. 03, Source Range Monitoring System.

215004A104

QUESTION: 009 (1.00)

REFERENCE:

1. DOP 1600-22, Rev. 5, page 3 of 11.

294001K102

QUESTION: 010 (1.00)

REFERENCE:

1. DAP 7-05, Rev. 16, Operating Charts, Logs and Records.

294001A106

QUESTION: 011 (1.00)

REFERENCE:

1. DLP 278L-S1, Rev. 4, Instrument Air System.
2. DOA 4700-01, Rev. 17, Instrument Air System Failure.

295019A101

QUESTION: 012 (1.00)

REFERENCE:

1. DOA 0040-01, Rev. 15, Slow Leak.

295027G010

QUESTION: 013 (1.00)

REFERENCE:

1. DOA 6600-01, Diesel Generator Failure, Rev. 09.
2. DLP 262L-S1, Auxiliary Power.

262001K406

QUESTION: 014 (1.00)

REFERENCE:

1. Technical Specifications 3/4.2.A, 3/4.6.K, and 3/4.6.L.

290002A204

QUESTION: 015 (1.00)

REFERENCE:

1. Technical Specification 3/4.6.1 and Safety Limit 2.1.B.

259002G005

QUESTION: 016 (1.00)

REFERENCE:

1. DOA 4700-01, Rev. 17, Instrument Air System Failure, page 10 of 29.
2. DOA 3900-01, Rev. 09, Loss of Cooling by the Service Water System, page 4.
3. DLP 261L-S1, Rev. 5, Standby Gas Treatment (SBGT).

261000K405

QUESTION: 017 (1.00)

REFERENCE:

1. DLP 215L-S5, Rev. 05, Average Power Range Monitoring, p. 16A.

202001A412

QUESTION: 018 (1.00)

REFERENCE:

1. DFP 0850-01, Rev. 03, Section B.3, p. 2.
2. DAN 902(3)-3 B-1, Rev. 04, "REFUEL FLOOR RAD HI".
3. DLP 272L-S2, Rev. 1, Process Radiation Monitoring.

295023K201

QUESTION: 019 (1.00)

REFERENCE:

1. DLP 272L-S2, Rev. 1, Process Radiation Monitoring, p. 18 of 27.
2. DEOP 300-1, Rev. 03, Secondary Containment Control.

295035K201

QUESTION: 020 (1.00)

REFERENCE:

1. EDE Letter 87-344, response to SOER 85-1, Haddam Neck Incident, Reactor Cavity Seal Failure.
2. DOP 1900-03, Rev. 20, Reactor Cavity, Dryer/Separator Storage Pit & Fuel Pool Level Control, page 9.

295023A204

QUESTION: 021 (1.00)

REFERENCE:

1. DGP 02-03, Rev. 29, Reactor Scram, Immediate Operator Actions.

295006G011

QUESTION: 022 (1.00)

REFERENCE:

1. DOA 0600-01, Rev. 17, Transient Level Control, page 4 of 14.

295009A102

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 023 (1.00)

REFERENCE:

1. DEOP 0100-00, Rev. 04, Reactor Control.
2. DEOP 0400-01, Rev. 04, RPV Flooding.

295031K101

QUESTION: 024 (1.00)

REFERENCE:

1. DEOP 0100, Rev. 04, Reactor Control.
2. DEOP 0200-00, Rev. 03, Primary Containment Control.

295010G011

QUESTION: 025 (1.00)

REFERENCE:

1. DLP 271L-S1, Rev. 2, Off-Gas System, page 16A of 31A.

271000K508

QUESTION: 026 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 04, Low Pressure Coolant Injection (LPCI), p. 12.
2. DLP 277L-S1, Rev. 03, Containment Cooling Water System.
3. DAN 902(3)-3 A-8, "3A LPCI HX DP LO". Rev. 03.

219000K504

QUESTION: 027 (1.00)

REFERENCE:

1. DLP 212L-S1, Rev. 3, RPS, page 7.
2. DOA 0500-02, Rev. 01, Partial 1/2 or Full Scram Actuation.

212000K104

QUESTION: 028 (1.00)

REFERENCE:

1. DLP 212L-S1, Rev. 3, Reactor Protection System, page 22.
2. DGP 02-03, Rev. 29, Reactor Scram, Step 13.c, page 8.

212000K412

QUESTION: 029 (1.00)

REFERENCE:

1. DLP 295L-S2, Rev. 1, DEOPs 200-1 & 200-2, Primary Containment Control and Primary Containment Hydrogen Control.

294001K115

QUESTION: 030 (1.00)

REFERENCE:

1. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

212000K202

QUESTION: 031 (1.00)

REFERENCE:

1. DLP 216L-S2, Rev. 0, Analog Trip System.
2. DLP 212L-S1, Rev. 3, Reactor Protection System.
3. DAN 903-5, C-13, Rev. 04, CHANNEL A/B RPV PRESS HI-HI.
4. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

212000K502

QUESTION: 032 (1.00)

REFERENCE:

1. DLP 212L-S1, Rev. 3, Reactor Protection System, page 20.

212000K106

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 033 (1.00)

REFERENCE:

1. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

216000K507

QUESTION: 034 (1.00)

REFERENCE:

1. DLP 223L-S1, Rev. 2, Containment Systems.

290001A202

QUESTION: 035 (1.00)

REFERENCE:

1. DLP 202L-S1, Rev. 4, Recirculation System.

202001A109

QUESTION: 036 (1.00)

REFERENCE:

1. DAN 902(3)-5 G-3, "RPIS SYS INOP".
2. DLP 201L-S2.

214000K303

QUESTION: 037 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 04, LPCI, page 16.

203000K411

QUESTION: 038 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, DEOP 100, Reactor Control, p. 35.

295007G007

QUESTION: 039 (1.00)

REFERENCE:

1. DAP 21-07, Rev. 2, Control/Maintenance of Fuses & the Fuse List, page 6.

   294001K107

QUESTION: 040 (1.00)

REFERENCE:

1. DFP 0800-01, Rev. 28, Refueling Procedure, G.7, page 7 of 14.

   294001A112

QUESTION: 041 (1.00)

REFERENCE:

1. DLP 295L-S8, Rev. 1, DEOP 400-5.

   295037G012

QUESTION: 042 (1.00)

REFERENCE:

1. DOA 0201-01, Rev. 05, Jet Pump Failure, page 2.

   295001A205

QUESTION: 043 (1.00)

REFERENCE:

1. DOA 3300-02, Rev. 13, Loss of Condenser Vacuum, page 3 of 8, Step C.7, Note: Question delineates differences in actions between Units.

   295002G007

QUESTION: 044 (1.00)

REFERENCE:

1. DOA 6900-01, Rev. 8, and Modification M12-3-95-003, changed ATS panel trouble alarm from 24/48 VDC to 125 VDC.

   295004K303

QUESTION: 045 (1.00)

REFERENCE:

1. DOA 6800-01, Rev 13, Loss of Power to Essential Service System Bus or Instrument Bus, page 2 of 14.
2. DLP 259L-S1, Rev. 02, Condensate/Feedwater.
3. DLP 223L-S1, Rev. 1, PCIS.
4. DLP 209L-S1, Rev. 5, Core Spray System.

   295004K303

QUESTION: 046 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, DEOP 300-2.

   295017K302

QUESTION: 047 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, DEOP 300-1 and 300-2.
2. DEOP 0100, Rev. 6.

   295017G012

QUESTION: 048 (1.00)

REFERENCE:

1. DOA 3700-01, Rev. 14, Loss of Cooling By Reactor Building Closed Cooling Water (RBCCW) System, Immediate Operator Actions.
2. DLP 208L-S1, Rev. 02, Reactor Building Closed Cooling Water (RBCCW), page 10A, "potential breach of primary containment".

   295018G011

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 049 (1.00)

REFERENCE:

1. DOP 300-19, Rev. 03, CRD Pump Discharge Cross-Tie Operation.
2. DLP 201L-S1, Control Rod Drive Hydraulic System.

295022G007

QUESTION: 050 (1.00)

REFERENCE:

1. DLP 223L-S1, Containment Systems, Rev. 2.

295020K203

QUESTION: 051 (1.00)

REFERENCE:

1. DOP 9900-205.

294001A115

QUESTION: 052 (1.00)

REFERENCE:

1. DAN 902(3)-3 A-1, Rev. 08, page 1 of 3.

294001A113

QUESTION: 053 (1.00)

REFERENCE:

1. DAP 07-02, Rev. 31, page 19 of 26.

294001A102

QUESTION: 054 (1.00)

REFERENCE:

1. DAP 12-04, Control of Access to High Radiation Areas, Rev. 27, p. 5.

294001K114

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 055 (1.00)

REFERENCE:

1. DAP 07-47, Rev. 01, NRC License Active Status Maintenance and Reactivation.
2. 10CFR55, parts 51, 53, 55, 57 and 59.
3. IE Notice 94-14, Supplement 1, Failure to Implement Requirements for Biennial Medical Examinations and Changes in Licensed Operator Medical Conditions, dated April 14, 1997.

294001A102

QUESTION: 056 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, pages 7 and 21 of 31.

259002K601

QUESTION: 057 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, Page 27 of 31.

259002K605

QUESTION: 058 (1.00)

REFERENCE:

1. DLP 264L-S1, Rev. 3, Emergency Diesel Generators Control and Operation, pages 17 and 18 of 37.

264000K402

QUESTION: 059 (1.00)

REFERENCE:

1. DOP 0202-12, Rev. 10, Recirculation Pump Motor Generator Set Scoop Tube Operation, Section G.1.d and G.1.e, pages 4 and 5 of 8.
2. DLP 202L-S2, Rev. 04, Recirculation Flow Control.

202002A408

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 060 (1.00)

REFERENCE:

1. DLP 201L-S1, Control Rod Drive Hydraulic System.

201001G007

QUESTION: 061 (1.00)

REFERENCE:

1. DLP 261L-S1, Rev. 5, SBGT, page 11A.

261000A301

QUESTION: 062 (1.00)

REFERENCE:

1. DLP 211L-S1, Rev. 02, Standby Liquid Control (SBLC), page 5A.

211000K301

QUESTION: 063 (1.00)

REFERENCE:

1. Technical Specification Table 2.2.A-1.

215005A104

QUESTION: 064 (1.00)

REFERENCE:

1. DLP 215L-S2, Rev. 03, RBM System, p. 5A.

215002A305

QUESTION: 065 (1.00)

REFERENCE:

1. DOS 800-1, Rev. 18, Refueling Interlock Checks.
2. DLP 201L-S2, Reactor Manual Control System (RMCS) and Rod Position Indication System (RPIS).
3. Technical Specification 3.10.I.

234000A302

QUESTION: 066 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev. 03, Main Turbine, page 13A of 55A.

245000A312

QUESTION: 067 (1.00)

REFERENCE:

1. DLP 241L-S1, Rev. 03, EHC Pressure Control and Logic.

245000A312

QUESTION: 068 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 4, Low Pressure Coolant Injection, p. 23.

262001K602

QUESTION: 069 (1.00)

REFERENCE:

1. DLP 262L-S5, Rev. 2, Low Voltage AC Distribution, Attachment B.
2. DOP 0500-03, Rev. 06, Reactor Protection System Power Supply Operation, Section F.3.d, page 4 of 14.

212000K601

REACTOR OPERATOR WRITTEN EXAM

QUESTION: 070 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, Reactor Control (DEOP 100), page 37.

295025A206

QUESTION: 071 (1.00)

REFERENCE:

1. DLP 215L-S3, Rev. 2, Intermediate Range Monitoring, page 15A.

215003A401

QUESTION: 072 (1.00)

REFERENCE:

1. Brand new question.
2. DLP 203L-S1, Rev. 4, Low Pressure Coolant Injection.

219000A414

QUESTION: 073 (1.00)

REFERENCE:

1. DGA-16, Rev. 07, Coolant High Activity/Fuel Element Failure, p. 4.

295017K301

QUESTION: 074 (1.00)

REFERENCE:

1. DGP 03-04, Rev. 24, Control Rod Movements.

295014A103

QUESTION: 075 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, RPV Flooding, DEOP 400-1 Lesson Plan, page 11A & 12A.
2. DEOP 0400-01, Rev. 04, RPV Flooding.

295031K101

QUESTION: 076 (1.00)

REFERENCE:

1. DLP 223L-S1, Rev. 2, Containment Systems, page 27A.

290001K402

QUESTION: 077 (1.00)

REFERENCE:

1. DLP 259L-S1, Rev. 02, Condensate/Feedwater, p. 20.

256000A302

QUESTION: 078 (1.00)

REFERENCE:

1. DLP 295L-S8, Rev. 1, Failure to Scram (DEOP 400-5), p. 44A of 45A.

295037A203

QUESTION: 079 (1.00)

REFERENCE:

1. DAP 01-11, Rev. 02, In Plant Communications Systems, page 2.
2. DAP 07-02, Rev. 26, Conduct of Shift Operations, page 14.

294001A104

QUESTION: 080 (1.00)

REFERENCE:

1. DOP 1300-2, Rev. 10, Automatic Operation of the Isolation Condenser, E.2, page 4.

207000A405

QUESTION: 081 (1.00)

REFERENCE:

1. DLP 202L-S3, Rev. 1, Recirc MG Set and Auxiliaries, page 26 and Table 1.

202002A303

QUESTION: 082 (1.00)

REFERENCE:

1. DLP 215L-S5, Rev. 05, APRM's, Attachment B, p. 3A.

215005A403

QUESTION: 083 (1.00)

REFERENCE:

1. DLP 212L-S2, Rev. 0, ATWS/ARI, p. 4.

212000K502

QUESTION: 084 (1.00)

REFERENCE:

1. DLP 218L-S1, Rev. 03, Automatic Depressurization System, Fig 3.

218000K403

QUESTION: 085 (1.00)

REFERENCE:

1. DLP 211L-S1, Rev. 02, SBLC, p. 9A, 11A, and 16A.

211000K603

QUESTION: 086 (1.00)

REFERENCE:

1. DLP 201L-S1, Control Rod Drive Hydraulic System, page 21.

201001K205

QUESTION: 087 (1.00)

REFERENCE:

1. DGP 01-01, Rev. 73, Unit Startup, Section G.117, p 52.

272000K501

QUESTION: 088 (1.00)

REFERENCE:

1. DOA 10-10, Rev. 05, Fire/Explosion, p. 2.

286000G002

QUESTION: 089 (1.00)

REFERENCE:

1. DOP 1200-01, Rev. 23, RWCU Operation During Startup and Shutdown.

204000K404

QUESTION: 090 (1.00)

REFERENCE:

1. DLP 233L-S1, Rev. 12, Fuel Pool Cooling, p. 15.

233000K601

QUESTION: 091 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, page 25.

295006K202

QUESTION: 092 (1.00)

REFERENCE:

1. DOP 1000-03, Rev. 21, Shutdown Cooling Mode of Operation, pages 10, 11 and 12.

295021K103

QUESTION: 093 (1.00)

REFERENCE:

1. DOA 6800-01, Rev. 13, Loss of Power to the ESS Bus or Instrument Bus, page 3 of 14.
2. DAN 902(3)-8 F-8, Rev. 03, ESS UPS TROUBLE.

295003K103

QUESTION: 094 (1.00)

REFERENCE:

1. DOA 3300-02, Rev. 13, Loss of Condenser Vacuum, page 3.

295002K207

QUESTION: 095 (1.00)

REFERENCE:

1. DOA 6900-T1, Rev. 7. Page 23, Bus 34-1 supplies breaker control power to LPCI Pump 3C and Core Spray Pump 3B. Page 24, Bus 34 supplies control power to CRD Pump 3B. Page 25, line item 20, HPCI loses normal power feed and transfers to alternate feed.

295004A102

QUESTION: 096 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, Radiation Release Control (DEOP 300-02), page 25A.

295038G011

QUESTION: 097 (1.00)

REFERENCE:

1. DOA 0300-01.
2. Technical Specification 3.3.G.

295022K301

QUESTION: 098 (1.00)

REFERENCE:

1. DOA 6900-02, Rev. 05, Failure of Unit 2 125 VDC Power Supply, page 4.

295004A202

QUESTION: 099 (1.00)

REFERENCE:

1. DLP 259L-S1, Rev. 02, Condensate/Feedwater, p. 25.

256000K313

QUESTION: 100 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev 03, Main Turbine.
2. DLP 212L-S1, Rev 03, Reactor Protection System.

295002K103

Written Examination Summary - Applicant Handouts

Reactor Operator

1. DOP 6400-8, Rev. 7, Figure 7, "Regulator: Out of Service"
2. Steam Tables and calculator
3. Technical Specification Table 2.2.A-1
4. DEOP 0400-01. Rev. 4, RPV Flooding (entry condition setpoints removed)

RO Total: Steam tables, calculator, and three pages of handouts

Senior Reactor Operator

1. DOP 6400-8, Rev. 7, Figure 7, "Regulator: Out of Service"
2. Steam Tables and calculator
3. Tech Spec 3.5.A, page 3/4.5-3
4. DEOP 0100, Tables 0100-C, 0100-E, 0100-F and 0100-G
5. EPIP 0200-T11. Rev. 7, page 1 of 89
6. Technical Specification Table 2.2.A-1 (entry condition setpoints removed)

SRO Total: Steam Tables, calculator, and eight pages of handouts

U. S. NUCLEAR REGULATORY COMMISSION
SITE SPECIFIC EXAMINATION
SENIOR REACTOR OPERATOR LICENSE
REGION 3

CANDIDATE'S NAME: MASTER EXAMINATION

FACILITY: Dresden 2 & 3

REACTOR TYPE: BWR-GE3

DATE ADMINISTERED: August 6, 1997

INSTRUCTIONS TO CANDIDATE:

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires a final grade of at least 80%. Examination papers will be picked up four (4) hours after the examination starts.

TEST VALUE	CANDIDATE'S SCORE	%	
100.0	______ FINAL GRADE	___ %	TOTALS

All work done on this examination is my own. I have neither given nor received aid.

____
Candidate's Signature

1. Cheating on the examination will result in a denial of your application and could result in more severe penalties.

2. After you complete the examination, sign the statement on the cover sheet indicating that the work is your own and you have not received or given assistance in completing the examination.

3. To pass the examination, you must achieve a grade of 80 percent or greater.

4. The point value for each question is indicated in parentheses after the question number.

5. There is a time limit of 4 hours for completing the examination.

6. Use only black ink or dark pencil to ensure legible copies.

7. Print your name in the blank provided on the examination cover sheet and the answer sheet.

8. Mark your answers on the answer sheet provided and do not leave any question blank.

9. If the intent of a question is unclear, ask questions of the examiner only.

10. 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.

11. When you complete the examination, assemble a package including the examination questions, examination aids, and answer sheets and give it to the examiner or proctor. Remember to sign the statement on the examination cover sheet.

12. After you have turned in your examination, leave the examination area as defined by the examiner.

A N S W E R K E Y

MASTER COPY

MULTIPLE CHOICE

001	a
002	d
003	b
004	b
005	b
006	a
007	c
008	b
009	c
010	c
011	d
012	a
013	c
014	b
015	a
016	a
017	a
018	a
019	a
020	b
021	b
022	b
023	b
024	b
025	a
026	a
027	a
028	a
029	a
030	b
031	a
032	a
033	a
034	a
035	d
036	c
037	b
038	a
039	d
040	d
041	a
042	b
043	b
044	a
045	d

ANSWER KEY

046	b
047	c
048	a
049	b
050	b
051	b
052	d
053	b
054	d
055	b
056	d
057	d
058	b
059	c
060	d or a JDE 8/18/97 (see next page)
061	a
062	b
063	c
064	c
065	b
066	c
067	b
068	a
069	c
070	c
071	b
072	b
073	c
074	c
075	a
076	c
077	c
078	a
079	b
080	b
081	c
082	b
083	b
084	c
085	d
086	b
087	a
088	b
089	d
090	c
091	a

ANSWER KEY

092 a

093 c

094 d

095 c

096 c

097 c

098 d

099 b

100 a

Question No. 60

After a post-exam review of the written exam results, RIII OLB determined that this question has two correct answers.

(********** END OF EXAMINATION **********)

MASTER COPY

QUESTION: 001 (1.00)

In accordance with DAP 7-27, which ONE of the following is the PRIMARY method to perform an independent verification on a manually throttled valve?

Assume that the valve is installed in a system with a local flow indication controlled by the valve AND the valve has a rising stem.

a. Observing the initial valve operator's action in positioning the throttled valve.

b. Observing flow indication through the throttled valve's system during system lineup.

c. Performing a second valve operation to verify position.

d. Performing an independent visual check of the valve position by comparing the actual valve position with the required valve position.

8/6/97

Start time : 0802

Stop time : 1202

QUESTION: 002 (1.00)

Unit 2 is shutdown after a scram. RPV pressure is 700 psig AND reactor water level can NOT be determined. HPCI is injecting into the vessel. Torus water level is going down and is presently 11.5 feet.

Under these conditions HPCI must be secured because . . .
(Select ONE of the following)

a. HPCI speed will be less than 2000 rpm.

b. HPCI suction line will become uncovered.

c. HPCI system leak is lowering Torus level.

d. HPCI exhaust discharge will pressurize the Torus.

QUESTION: 003 (1.00)

Per the Technical Specifications, the Rod Worth Minimizer (RWM) is NOT required to be operable above 20% reactor power because at higher power levels . . .
(Select ONE of the following)

a. the higher power production in the upper core region minimizes control rod worth.

b. increased voiding of the core limits the reactivity addition rate of a dropped control rod.

c. the reactivity change associated with a dropped control rod becomes insignificant when compared with the overall core reactivity.

d. reactivity changes are less pronounced for a dropped control rod because the EHC system maintains RCS pressure essentially constant.

QUESTION: 004 (1.00)

Fuel loading is in progress. The first three fuel assemblies of a cell are fully seated in the correct core locations and are in the correct orientation. The fourth assembly loaded in this cell is inadvertently oriented 180 degrees out from its correct position, but is fully seated into its correct core location.

If the reactor was operated in this condition, then the incorrect fuel assembly orientation will affect . . . (Select ONE of the following)

a. TIP operation.

b. proper control rod blade operation.

c. core flow through the fuel assembly.

d. core bypass flow to cool the control rod blade AND incore instrumentation.

QUESTION: 005 (1.00)

What is the purpose of a Turbine trip caused by a Main Generator 86 device trip?
(Select ONE of the following)

a. Prevent stator overheating.

b. Provide overspeed protection.

c. Provide reverse power protection.

d. Prevent last stage bucket overheating.

QUESTION: 006 (1.00)

The decision to maintain primary containment integrity without regard to the status of maintaining adequate core cooling must be made when . . .
(Select ONE of the following)

a. Torus bottom pressure can not be maintained below 65 psig.

b. Drywell temperature can not be maintained below 281 degrees F.

c. Torus water level AND RPV pressure can not be maintained below the ADSV Tail Pipe Level Limit.

d. Torus bottom pressure can not be maintained below the Pressure Suppression Pressure for the present Torus water level.

QUESTION: 007 (1.00)

Given the following plant conditions:

- L1220 indicates 230 amps
- L1221 indicates 250 amps
- L0302 indicates 255 amps
- 345 kV BT 4-8 CB is open
- All ring Bus breakers are closed
- Main Generator is at 750 MWe AND 50 MVARs

Do to an OOS error, the amplidyne is inadvertently de-energized, causing the voltage regulator to shift to Manual. Based on these conditions, what is the actions, if any, that must be taken?
(Select ONE of the following)

a. Operation in this condition can continue indefinitely. No reports or actions are required.

b. Inform BPO that the voltage regulator is in Manual, then continue operation in the present region.

c. Inform BPO that the voltage regulator is in Manual. Raise reactive loading to at least 90 MVARs.

d. Inform BPO that the voltage regulator is in Manual. Reactive loading will automatically raise to approximately 90 MVARs.

QUESTION: 008 (1.00)

Given the following plant conditions:

- Current time 0915
- 100% reactor power on Units 2 AND 3
- Personal OOS 97P-05 in place from 0730 to 1030 (projected)

A request was made to operate a Personal Out-Of-Service (OOS) 97P-05 boundary isolation valve to support draining an adjacent section of piping.

Which ONE of the following reasons would be used to REJECT this request?

a. The equipment manipulations were specified on the Personal OOS form.

b. The lead worker has released the equipment for manipulation in his absence from site.

c. The Unit Supervisor (or designee) authorized removal of the Personal OOS card.

d. The Personal OOS, if this valve were to be operated, would be needed for an additional four hours.

QUESTION: 009 (1.00)

In accordance with DAP 07-05, Operating Charts, Logs and Records, which ONE of the following items shall be entered into the Degraded Equipment Log?

a. Technical Specification equipment that activates an LCO.

b. Equipment that has been modified by a temporary alteration.

c. Instrumentation that will hamper implementation of Dresden Emergency Operating Procedures.

d. DATR equipment that is defined by the Probabilistic Risk Analysis (PRA) study as "critical" equipment.

QUESTION: 010 (1.00)

The following alarms are received:

Panel 902(3)-4, H-18, DRYWELL FLOOR DRN SUMP LVL HI-HI
A-17, DRYWELL EQUIP SUMP LVL HI-HI

A decision is made to enter DOA 0040-01, Slow Leak.

Which ONE of the following items is an Immediate Operator Action?

a. Evacuate all personnel from Containment.

b. Initiate a load reduction to stabilize EHC response.

c. Maintain water level with the Feedwater Control system.

d. If Reactor Building exhaust radiation level exceeds 2 mrem/hr, then initiate Reactor Building Ventilation System isolation AND manually initiate SBGT.

QUESTION: 011 (1.00)

If Bus 24-1 becomes de-energized AND is NOT re-energized, which ONE of the following describes the response of MCC 28-7 AND MCC 29-7?

a. MCC 28-7 AND MCC 29-7 remain energized without losing power.

b. MCC 28-7 will de-energize AND MCC 29-7 will remain energized.

c. MCC 28-7 will remain energized AND MCC 29-7 will de-energize.

d. MCC 28-7 AND MCC 29-7 lose power AND re-energize after a time delay.

QUESTION: 012 (1.00)

Given the following plant conditions:

- 100% steady state power on Unit 2 for an extended period of time

Then, the following events occur on Unit 2:

- a full MSIV isolation AND associated reactor scram occurs
- an ERV does NOT completely reseat following operation

The Shift Manager decides to initiate the MAXIMUM allowable Technical Specification cooldown in order to reach cold shutdown. Assuming that a uniform cooldown is initiated at the MAXIMUM allowable cooldown rate from the MAXIMUM allowable LCO steam dome pressure, how many minutes will it take to reach the point that Shutdown Cooling can be placed into service?
(Select ONE of the following)

a. 118 minutes

b. 114 minutes

c. 110 minutes

d. 105 minutes

QUESTION: 013 (1.00)

Given the following plant conditions for Unit 2:

- One control rod is withdrawn AND immovable
- The plant has been operating in Single Loop for 2 days per DGP-03-03, Single Recirculation Loop Operation.

What is the current limit on MCPR?
(Select ONE of the following)

a. 1.07

b. 1.08

c. 1.09

d. 1.10

QUESTION: 014 (1.00)

Given the following plant conditions:

- All Unit 2 Technical Specification equipment is operable.
- Unit 2 is at 100% reactor power.

A Fragnet is being developed to perform HPCI steam trap drain line repairs. The work requires the HPCI system to be isolated and drained. Restoration and surveillance testing following maintenance will take 8 hours. In accordance with DAP 18-08, Guideline for the Performance of Online Maintenance, what is the MAXIMUM time that you can allow the Maintenance Department to work on this system?
(Select ONE of the following)

a. 72 hours

b. 160 hours

c. 168 hours

d. 252 hours

QUESTION: 015 (1.00)

Unit 3 is at 100% reactor power when a total loss of service water occurs.

What is the effect on a running train of the SBGT system? (Select ONE of the following)

a. Shortened retention of iodine in charcoal.

b. Standby SBGT train auto-starts on low flow.

c. SBGT system damper 2/3-7509 (Cross-Tie damper) fails closed.

d. SBGT system damper 3-7503 (Reactor Building Inlet damper) fails open.

QUESTION: 016 (1.00)

The MSIV's have two closing times associated with their automatic closure.

What is the purpose of the smaller of these two times? (Select ONE of the following)

a. Limit the peak heat flux generated by an MSIV isolation.

b. Limit the drop in reactor water level AND thus limit the loss of inventory during a LOCA.

c. Limit the release of fission products from containment during a gross fuel failure.

d. Limit the damage to the MSIV valve seats during closure to allow a leak tightness assumed in the design analysis.

QUESTION: 017 (1.00)

Core flow is NOT directly proportional to recirculation loop drive flow.

Which ONE of the following describes the reason?

a. As reactor power rises, the void fraction increases the hydraulic resistance, thus causes a reduction in core flow.

b. As reactor power rises, the overall change in the boiling boundary increases resistance to flow, thus causing a rise in core flow.

c. As reactor power rises, power production is pushed lower AND to the outside of the core, thus causing a reduction in core flow in these regions AND a reduction in overall core flow.

d. Regardless of reactor power level, core flow is a function of the dp across the flow orifices, thus any other hydro-dynamic effects are insignificant with respect to this dp.

QUESTION: 018 (1.00)

A transient on Unit 3 has resulted in a loss of water from the Torus.

Current plant conditions are:

- Torus level is 7.0 feet AND steady
- Torus bottom pressure is 3.5 psig
- Torus water temperature is 145 degrees F

Which ONE of the following statements correctly describes the effect that this Torus water level has on the operation of ECCS pumps for reactor water level control.

a. Discontinue operation of the LPCI or the Core Spray pumps to prevent cavitation on low suction pressure.

b. Continued operation of HPCI is allowed if the suction is aligned to the CST to provide adequate suction pressure.

c. Total LPCI system flow must be limited to 20,000 gpm with four pumps operating to ensure adequate NPSH.

d. Total Core Spray system flow must be limited to 10,000 gpm with two pumps operating to ensure adequate NPSH.

QUESTION: 019 (1.00)

Given the following plant conditions:

- Unit 3 is at 100% reactor power
- the process computer is out of service for the next 30 to 40 minutes for periodic system maintenance
- a partial loss of power occurs which causes a reactor scram AND de-energizes the full core display AND rod worth minimizer CRT

In accordance with DGP 02-03, which ONE of the following Immediate Operator Actions is the operator required to perform?

a. Initiate ARI.

b. Verify APRM's downscale.

c. Insert the SRMs AND verify that a negative period exists.

d. Insert the IRMs AND verify that reactor power is decreasing with a negative period.

QUESTION: 020 (1.00)

Given the following plant conditions:

- Unit 2 is at 100% reactor power
- the Feed Water Level Control System (FWLCS) is in Master Auto with normal reactor water level setpoint
- a narrow range (NR) level transmitter, NOT selected on RX LVL SELECT, fails downscale

The automatic plant response will be . . .
(Select ONE of the following)

a. HPCI receives a start signal.

b. FWLCS will maintain reactor water level at the normal reactor water level setpoint.

c. FWLCS will overfeed the reactor vessel AND a high reactor water level reactor trip will occur.

d. feedwater level setpoint Auto Runback occurs reducing the reactor water level setpoint to 1/2 of original value.

QUESTION: 021 (1.00)

The following plant conditions exist on Unit 3:

- Reactor shutdown, all rods in
- Drywell pressure is 2.5 psig
- Drywell temperature is 405 degrees F
- Reactor pressure is 50 psig
- Reactor water level is -70 inches

Which ONE of the following is the reason for exiting DEOP 0100, "Reactor Control", AND entering DEOP 0400-01, "RPV Flooding"?

a. It is the alternate method to feed the RPV to maintain the RPV water level above -70 inches.

b. The RPV water level instruments can not be relied upon to assure adequate core cooling.

c. RPV pressure is low enough to allow the use of Core Spray to submerge the core to maintain adequate core cooling.

d. Drywell temperature is above the saturation curve which requires flooding of the RPV to quench reactor pressure.

QUESTION: 022 (1.00)

An unisolable reactor coolant leak on Unit 2 has resulted in a reactor scram AND a rapid rise in drywell pressure. The following additional plant conditions exist:

- Reactor building ventilation radiation is at 3 mr/hr
- Reactor pressure is 775 psig
- Drywell temperature is 155 degrees F
- All rods are inserted
- Drywell pressure is 3 psig AND rising
- HPCI is maintaining reactor water level at 10 inches

Which ONE of the following DEOP(s) are required to be executed?

a. DEOP 0200-01 (Primary Containment Control) only.

b. DEOP 0100 (Reactor Control) AND DEOP 0200-01 (Primary Containment Control).

c. DEOP 0100 (Reactor Control), DEOP 0200-01 (Primary Containment Control) AND DEOP 300-01 (Secondary Containment Control).

d. DEOP 0200-01 (Primary Containment Control) AND DEOP 300-01 (Secondary Containment Control).

QUESTION: 023 (1.00)

Which ONE of the following components is designed to limit Xenon AND Krypton levels that reach the Off Gas Charcoal Adsorber?

a. recombiner.

b. holdup pipe.

c. cooler condenser.

d. charcoal absorber prefilter.

QUESTION: 024 (1.00)

Following a LPCI initiation signal, the EARLIEST that the LPCI motor operated Heat Exchanger Bypass valves can be manually closed from the control room is...
(Select ONE of the following)

a. 20 seconds.

b. 30 seconds.

c. 5 minutes.

d. 8.5 minutes.

QUESTION: 025 (1.00)

Select the potential detrimental effects of a hydrogen-oxygen burn in the primary containment from the following:

1. Uncontrolled release of radioactivity.

2. Pressures exceeding the structural capability of the Drywell.

3. Damage to plant equipment required for the shutdown of the reactor plant.

4. Pressures excursions exceeding the capacity of the Torus to Reactor Building vacuum breakers.

Select ONE of the following combinations:

a. 1, 2, 3

b. 2, 3, 4

c. 1, 3, 4

d. 1, 2, 4

QUESTION: 026 (1.00)

Given the following plant conditions on Unit 2:

- Reactor power is 100%
- all reactor water level instruments are Operable

It has been observed that over the last couple of shifts that the medium range Yarway reactor water level instruments on the 2202-5 and 2206 racks are drifting high. All control room indications are steady.

If the drifting continues it will ....
(Select ONE of the following)

a. have no effect on RPS.

b. result in the high reactor water level scram occurring at a level higher than designed.

c. result in the low reactor water level scram occurring at a higher level than designed.

d. result in the low reactor water level Group 1 isolation occurring at a lower level than designed.

QUESTION: 027 (1.00)

Given the following plant conditions:

- Unit 3 is at 100% reactor power
- all equipment is operable except for the A channel of medium range (MR) reactor water level instrumentation, MR transmitter LT 3-263-57A, has failed downscale

The following alarm is received:

903-5, C-13, CHANNEL A/B RPV PRESS HI-HI

(which is due to the failure of RPV Pressure transmitter 3-263-55D upscale)

What will the plant response be?
(Select ONE of the following)

a. Full reactor scram.

b. Trips the Recirc MG Set drive motor breakers.

c. Only Group I, II, III AND IV "B" scram pilot solenoid valves will de-energize.

d. Only Group I, II, III AND IV "A" scram pilot solenoid valves will de-energize.

QUESTION: 028 (1.00)

The design of the scram discharge volume (SDV) high-high level input into RPS assures that . . .
(Select ONE of the following)

a. the SDV can accommodate water discharged after a scram.

b. an exhaust path for drive water flow is provided following a scram.

c. operation of the CRD system and associated piping will not interfere with other safety related equipment operation following a scram.

d. an extension to the reactor coolant system pressure boundary is provided in the event that a drive water return line is leaking reactor water back through the CRD mechanism.

QUESTION: 029 (1.00)

What will cause the Medium Range Level indication to fail full downscale?
(Select ONE of the following)

a. A break occurs in the variable leg.

b. A break occurs in the reference leg.

c. A bellows leak occurs in the DP cell.

d. A rapid depressurization of the reactor occurs.

QUESTION: 030 (1.00)

During normal 100% reactor power operation, it is noted that Secondary Containment differential pressure is at 0 inches of water AND going up.

Which ONE of the following states the adverse consequences that will occur if this situation is allowed to continue?

a. The torus could be overpressurized in the event of a Loss of Coolant Accident (LOCA).

b. In the event of a design basis Loss of Coolant Accident (LOCA) an unmonitored release could occur.

c. The differential pressure across access doors would limit the access to the Secondary Containment.

d. Higher inleakage to the Secondary Containment would occur due to the excessive load being placed on the Reactor Building vent and purge system.

QUESTION: 031 (1.00)

Unit 3 is operating at 100% reactor power when the "3A RECIRC PP CONT SEAL FLOW HI" computer point alarm is received.

Other indications on the 3A recirc pump are as follows:

- No. 1 seal pressure 990 psig AND steady
- No. 2 seal pressure 785 psig AND rising

Which ONE of the following is the cause of the alarm?

a. A failure of the No. 1 seal.

b. A failure of the No. 2 seal.

c. Plugging of the No. 1 internal restricting orifice.

d. Plugging of the No. 2 internal restricting orifice.

QUESTION: 032 (1.00)

Unit 2 is conducting a plant startup at approximately 35% reactor power when the "RPIS SYS INOP" annunciator is received.

The NSO must . . .
(Select ONE of the following)

a. stop any power changes in progress AND investigate the failure.

b. attempt to move the selected control rod to verify failure of the RPIS.

c. return the reactor to hot shutdown within 24 hours by normal reactor shutdown.

d. continue raising reactor power as instructed AND call Instrument Maintenance for assistance.

QUESTION: 033 (1.00)

Given the following plant conditions:

- A LOCA has occurred at 100% power on Unit 2

If the LPCI Loop Select Logic senses a break in the "B" recirc loop, COINCIDENT with 2 psig in the Drywell, which ONE of the following does NOT occur?

a. The "A" recirc loop suction valve shuts.

b. The system "B" LPCI injection valves shut.

c. The "A" recirc loop discharge valve shuts.

d. The "B" recirc loop discharge valve remains open.

QUESTION: 034 (1.00)

Which ONE of the following is the reason for stabilizing reactor pressure below 1060 psig following a reactor scram and subsequent entry into DEOP 100, Reactor Control?

a. To avoid ADS valves lifting.

b. To permit RPV pressure control with the bypass valves.

c. To maintain cooldown rate below 100 degrees F per hour.

d. To ensure proper operation of reactor water level instruments.

QUESTION: 035 (1.00)

Given the following plant conditions:

- Drywell pressure is 3.2 psig.
- Primary containment hydrogen concentration is 4.5%
- Primary containment oxygen concentration is 3.5%

Which ONE of the following states the additional conditions necessary to allow containment venting?

a. Nitrogen purge must be established.

b. Oxygen concentration must be lowered below 3.5%.

c. Hydrogen concentration must be lowered below 3.5%.

d. Offsite doses must be expected to remain below LCO levels.

QUESTION: 036 (1.00)

You have been directed to pull a fuse for an Out-Of-Service. There is NO local fuse label listing type or amperage of the fuse. To verify the correct fuse was installed, you must.... (Select ONE of the following)

a. check the electrical print for the proper type fuse.

b. check the System Electrical Checklists to verify the fuse information.

c. compare the pulled fuse to the specifications in EWCS.

d. inform the Label Coordinator of the missing label AND have him verify the fuse.

QUESTION: 037 (1.00)

Which ONE of the following personnel has the authority to terminate a fire watch?

a. Fire watch.

b. Shift Manager/designee.

c. Discretion of the Work Group Supervisor.

d. Safety & Property Loss Prevention Team Supervisor.

QUESTION: 038 (1.00)

Which ONE of the following is an example of a Temporary Alteration which requires tracking per DAP 05-08, "Control of Temporary System Alterations"?

a. Plugging of a floor drain.

b. Removal of a fuse in accordance with an Out-of-Service.

c. Any alteration installed in equipment that is out-of-service.

d. Lifting a wire for troubleshooting in accordance with an approved test procedure.

QUESTION: 039 (1.00)

Fuel movements are being performed on Unit 2. The Unit 2 NSO and Fuel Handlers cannot agree as to which move is to be performed next.

According to Unit 2 Master Refueling Procedure, DFP 800-1, which ONE of the following personnel is to be contacted for further assistance?

a. Shift Manager.

b. Fuel Handling Supervisor.

c. Nuclear Materials Custodian.

d. Control Room Nuclear Observer.

QUESTION: 040 (1.00)

Given the following plant conditions on Unit 2:

- An ATWS from a CRD hydraulic lock is in progress.
- ADS valves are being cycled to control reactor pressure.
- Torus temperature is 114 degrees F.
- Drywell pressure is 3.5 psig.
- Current reactor power is 10%

Which ONE of the following is the prescribed reactor water level band?
(Select ONE of the following)

a. +8 AND +48 inches

b. -143 AND +48 inches

c. -143 inches AND a level to which it was lowered.

d. -164 inches AND a level to which it was lowered.

QUESTION: 041 (1.00)

Unit 2 is operating at approximately 85% reactor power when a reduction in condenser vacuum commences. In accordance with the Immediate Operator actions of DOA 3300-02, you must . . .

(Select ONE of the following)

a. trip hydrogen addition.

b. start the mechanical vacuum pump.

c. raise Gland Seal Steam pressure to 10 psig.

d. raise Steam Jet Air Ejector steam supply pressure to 150 psig.

QUESTION: 042 (1.00)

Which ONE of the following will occur if all 24/48 VDC is lost on Unit 3?

a. ATS panel trouble alarm.

b. Will experience a half scram.

c. Core Spray system minimum flow valves will not operate.

d. Wide range reactor water level indication becomes inoperable.

QUESTION: 043 (1.00)

Which ONE of the following will occur if the Unit 3 Essential Service System Bus is lost?

a. Complete Group II AND III isolations will occur.

b. Reactor feed pump minimum flow valves fail open.

c. Core spray system minimum flow valve will not operate.

d. Wide range reactor water level indication becomes inoperable.

QUESTION: 044 (1.00)

An unisolable steam leak in the turbine building has resulted in a high radioactivity level in the turbine building. The Unit Supervisor wishes to monitor the release rate to the environment.

Which ONE of the following will accomplish that?

a. Starting Turbine Building Ventilation if shutdown.

b. Securing Turbine Building Ventilation if operating.

c. Starting SBGT AND aligning it to the Turbine Building.

d. Starting Reactor Building Ventilation AND aligning it to the Turbine Building.

QUESTION: 045 (1.00)

Isolation of a primary system leak is required by DEOP 300-1 AND 300-2, in order to limit radioactive discharge.

Under these conditions, the term "Primary System" refers to any system....
(Select ONE of the following)

a. containing reactor coolant.

b. for which the ASME "N" stamp is issued.

c. connected to the RPV that contains radioactive water.

d. connected to the RPV that has a reduced leak rate if RPV pressure is lowered.

QUESTION: 046 (1.00)

During Unit 2 100% reactor power operation, the following events occur:

- DAN 923-1 C-1, "U2 RBCCW PP TRIP"
- DAN 902-3 A-13, "Drywell Pressure Hi"
- DAN 902-4 G-17, "Drywell Atmosphere Temp Hi"
- Standby RBCCW pump can NOT be started

Sixty seconds later:

- DAN 923-1 D-1, "U2 RBCCW Press Lo"
- Drywell temperature is 170 degree F AND slowly rising
- Drywell pressure is 2.1 psig AND slowly rising

What Immediate Operator Actions must you take?
(Select ONE of the following)

a. Verify reactor scram and/or manually scram the reactor, AND verify automatic isolation of RBCCW.

b. Verify reactor scram, trip Reactor Recirc pumps, AND isolate RBCCW to the Drywell.

c. Trip the Reactor Recirc pumps, verify core flow has dropped to less than 45 Mlbm/hr, AND enter DGP 2-1, "Normal Unit Shutdown" AND DEOP 200-2.

d. Reduce Reactor Recirc pump speeds, verify power/flow region, enter DEOP 100 AND 200-1, AND assess if any plant equipment damage is imminent before manually scramming the reactor.

QUESTION: 047 (1.00)

A fire has resulted in the evacuation of the Control Room. Implementation of DSSP 0100-CR "HOT SHUTDOWN PROCEDURE - CONTROL ROOM EVACUATION" is underway. Forty minutes have passed and control of all systems needed for safe shutdown has NOT been established.

Which ONE of the following emergency classification must be declared?

a. Unusual Event.

b. Alert.

c. Site Emergency.

d. General Emergency.

QUESTION: 048 (1.00)

A Group II Isolation signal exists AND the Containment Vent valves are required to be operated. The reactor Mode Switch is in "SHUTDOWN".

What condition(s) is(are) required to open valves 1601-61 (Torus 2 inch Vent) AND 1601-63 (Vent to SBGT)? (Select ONE of the following)

a. Vent Isol Signal Bypass switch in the "Torus" position.

b. Vent Isol Signal Bypass switch in the "Drywell" position.

c. Mode Switch in "RUN" AND Vent Isol Signal Bypass switch in the "Torus" position.

d. Mode Switch in "RUN" AND Vent Isol Signal Bypass switch in the "Drywell" position.

QUESTION: 049 (1.00)

Which ONE of the following is the MINIMUM Emergency Action Level that requires activation of the Emergency Response Data System (ERDS)?

a. Unusual Event

b. Alert

c. Site Emergency

d. General Emergency

QUESTION: 050 (1.00)

Which ONE of the following explains how ARM instruments are identified as "DEOP" instruments?

a. The labels have a black dot.

b. The instruments have purple labels.

c. The words "DEOP" are found on the label.

d. The instrument identifier is written in red.

QUESTION: 051 (1.00)

As the off-going Unit 2 NSO preparing for turnover, you have completed the Unit 2 Log and reviewed and initialed the surveillance performed during your shift.

Which ONE of the following is NOT required of you for proper shift turnover?

a. Perform a face to face turnover with the oncoming NSO.

b. Discuss applicable DEL items with the Unit Supervisor.

c. Complete appropriate portions of the NSO shift turnover checklist.

d. Remain on the unit until the oncoming NSO has received a satisfactory turnover AND is fully aware of existing conditions.

QUESTION: 052 (1.00)

According to DAP 12-04, the Radiation Protection Manager AND Operations Manager must make all efforts to eliminate power entries into Nitrogen-16 areas.

At what unit load does this first occur?
(Select ONE of the following)

a. 200 megawatts

b. 250 megawatts

c. 300 megawatts

d. 350 megawatts

QUESTION: 053 (1.00)

A licensed operator, as defined by 10CFR55, must meet several requirements to maintain his/her license.

Which ONE of the following is included in these requirements in order to maintain an ACTIVE status of his/her NRC license?

a. The licensee shall have an annual medical examination.

b. Pass a comprehensive requalification written examination AND an annual operating test.

c. Must apply for renewal of license at least 30 days prior to the five year expiration date (based on date of license issuance)

d. The licensee shall actively perform the functions of the licensed position on a minimum of seven 8-hour shifts or five 12-hour shifts per calendar year.

QUESTION: 054 (1.00)

While operating at 100% reactor power, feedwater regulating valve (FWRV) 2A is in service controlling reactor water level. The air line supplying FWRV 2A ruptures AND air is rapidly lost to the operator.

Which ONE of the statements below identifies the response of the FWRV?

a. The valve fails full open, but the speed is limited by the hydraulic damper.

b. The valve fails full closed, but the speed is limited by the hydraulic damper.

c. The valve fails full open immediately since it uses air to close, AND spring pressure to open.

d. The valve would "lock up" in its present position, due to the actuation of the air lock valve.

QUESTION: 055 (1.00)

Unit 3 is at 100% reactor power with feedwater in automatic control when the selected reactor water level transmitter (Narrow Range "A") fails off scale high. The Bailey System indicates "bad quality".

The Control System will ...
(Select ONE of the following)

a. automatically shift to Medium Range "A".

b. automatically shift to Narrow Range "B".

c. lock up at the last valid reactor water level signal sensed from Narrow Range "A".

d. reduce feedwater flow until an alternate reactor water level instrument has been manually selected.

QUESTION: 056 (1.00)

The automatic emergency start of the diesel generator bypasses some of the protective trips for the diesel and/or the diesel generator supply breaker.

Which ONE of the following statements is NOT correct concerning the diesel generator trip bypasses?

a. The ECCS auto start bypasses the diesel engine low lube oil pressure trip.

b. The undervoltage auto start bypasses the diesel generator output breaker overcurrent trip.

c. The ECCS auto start bypasses the diesel generator output breaker reverse power trip.

d. The undervoltage auto start bypasses the diesel generator high differential current trip.

QUESTION: 057 (1.00)

Unit 2 is operating at 75% reactor power with the 'A' Recirc Pump MG set scoop tube locked out for maintenance.

Which ONE of the following describes the preferred Recirc Flow Control System lineup under these conditions?

Controller Positions

	MASTER	'A' RECIRC	'B' RECIRC
a.	automatic	automatic	manual
b.	manual	manual	automatic
c.	manual	automatic	automatic
d.	manual	manual	manual

QUESTION: 058 (1.00)

During a normal control rod insertion, what prevents drive water from recirculating back into the cooling water header? (Select ONE of the following)

a. The system cooling water header pressure is greater than drive water pressure.

b. The cooling water supply header to the hydraulic control unit includes a check valve.

c. The control rod drive stabilizing valve closes on an "Insert" signal thus isolating the cooling water header.

d. The cooling water header isolation valve (104) to the hydraulic control unit closes when "Insert" is selected.

QUESTION: 059 (1.00)

A valid initiation signal for SBGT is received. Train "A" is in PRIMARY and Train "B" is in STANDBY. Under the existing circumstances, which ONE of the following will initiate Train "B" of the SBGT System?

a. Drywell radiation level of 100 R/hr.

b. Refueling Floor radiation level of 100 mr/hr.

c. Low flow condition on Train "A" for 20 seconds.

d. Reactor Building Ventilation radiation level of 4 mr/hr.

QUESTION: 060 (1.00)

In reference to the Standby Liquid Control System, a MINIMUM of 600 ppm boron in the reactor core is required to be injected within 100 minutes.

Which ONE of the following statements is correct concerning the boron injection?

a. A 600 ppm concentration will provide at least a 3% delta K Shutdown Margin during cold xenon free conditions.

b. The 100 minute maximum time requirement is necessary to provide adequate mixing, AND prevent "chugging" in the core.

c. The boron solution in the core results in changing the moderator temperature coefficient from positive to negative.

d. The required 600 ppm boron concentration, in the core, includes a 25% additional margin to accommodate improper mixing.

QUESTION: 061 (1.00)

Given the following Unit 2 plant conditions:

- Reactor Power is 1516 MWt
- "A" Recirculation Pump Trip
- Calculated Total Core Flow is 58 million lbs/hr
- Loop B Recirculation Flow is 55%

Which ONE of the following is the setpoint for the APRM Flow Biased Neutron Flux High Trip?

a. 92.8%

b. 93.5%

c. 95.9%

d. 97.8%

QUESTION: 062 (1.00)

While making a tour of the control room back panels, you notice an alarm light on a RBM channel on top of panel 902-37. The light is labeled, "REF APRM DOWNSCALE".

What RBM function is associated with this alarm? (Select ONE of the following)

a. Rod Block.

b. Automatically bypasses RBM.

c. Bypasses Rod Insert Blocks.

d. Indication that the reference APRM is at 40% reactor power.

QUESTION: 063 (1.00)

One of the refueling requirements at Dresden is to check the "one-rod-out" interlock for the Reactor Manual Control System (RMCS) prior to performing refueling operations.

Which ONE of the following statements correctly describes the proper operation of the "one-rod-out" interlock?

a. Selecting a rod which is fully inserted initiates a Rod Block.

b. A selected rod is fully withdrawn (notch 48); at which time a Rod Block is initiated.

c. A selected rod is withdrawn to any position (02 to 48); a second rod can be selected, but not withdrawn.

d. A selected rod can be fully withdrawn (notch 48); a second rod initiates a Rod Block if withdrawn to notch 02.

QUESTION: 064 (1.00)

Which ONE of the following statements correctly describes the operation of the INTERCEPT valves on a Main Turbine Overspeed?

a. A drop in pressure in the Moisture Separator to 105 psig will cause the Intercept Valves to go closed to prevent turbine overspeed.

b. Intercept Valves 2, 4 AND 6 fully close at 103% turbine speed, while valves 1, 3 AND 5 are not closed until 105% turbine speed.

c. As turbine speed is reduced the Intercept valves 1, 3 AND 5 will re-open, while valves 2, 4, AND 6 will ramp open after the first group is 90% open.

d. Once activated by turbine speed, all Intercept Valves remain closed until turbine speed drops to 50% at which time all intercept valves ramp open.

QUESTION: 065 (1.00)

Which ONE of the following is the MINIMUM water temperature allowed in the Unit 2 or 3 Spent Fuel Pool or reactor cavity when fuel is present?

a. 40 degrees F

b. 68 degrees F

c. 77 degrees F

d. 100 degrees F

QUESTION: 066 (1.00)

Operators have been attempting to reseat the Unit 3 Isolation Condenser Reactor INLET ISOLATION valve, 3-1301-3, which has lost power. The following conditions still exist:

- Reactor power is 100%
- HPCI system is Out of Service (OOS) for the next 72 hours
- Isolation Condenser shell side temperature is rapidly rising
- Isolation Condenser steam line temperature is 180 degrees F AND rapidly rising
- Isolation Condenser shell side water level is rapidly approaching 12 feet

The operators must . . . (Select ONE of the following)

a. immediately scram the reactor AND isolate the Isolation Condenser.

b. isolate the Isolation Condenser AND reduce Isolation Condenser shell side water level less than 8 feet within 12 hours.

c. isolate the Isolation Condenser AND reduce reactor pressure to less than or equal to 150 psig within 36 hours.

d. isolate the Isolation Condenser AND continue plant operations indefinitely as long as ADS remains operable.

QUESTION: 067 (1.00)

Dresden Unit 2 has just experienced a loss of offsite power concurrent with +2.5 psig Drywell pressure. The diesels auto start as designed.

Which ONE of the following states the timed starting sequence for the emergency bus equipment?

a. The diesel generator breaker closes within 10 seconds, then the first LPCI pump starts followed by the core spray pump 5 seconds later followed by the second LPCI pump 5 seconds later.

b. The diesel generator breaker closes within 10 seconds, then the first LPCI pump starts followed by the second LPCI pump 5 seconds later followed by the core spray pump 5 seconds later.

c. When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes, then the first LPCI pump starts followed by the second LPCI pump 5 seconds later followed by the core spray pump 5 seconds later.

d. When reactor pressure reaches 350 psig AND 8.5 minutes have elapsed from initiation, then the diesel generator breaker closes, then the first low pressure coolant injection (LPCI) pump starts followed by the core spray pump 5 seconds later followed by the second LPCI pump 5 seconds later.

QUESTION: 068 (1.00)

Which ONE of the following will NOT cause a trip of the RPS EPA breakers?

a. Over frequency

b. Under frequency

c. Over voltage

d. Under voltage

QUESTION: 069 (1.00)

The reactor was operating at approximately 5% reactor power following a refueling outage when a scram occurred due to a spurious main steamline isolation.

Which ONE of the following pressure control methods will minimize the inventory loss from the reactor vessel?

a. ADS valves

b. IC vent valve venting.

c. RWCU in Recirculation Mode

d. HPCI in the pressure control mode.

QUESTION: 070 (1.00)

The Intermediate Range Monitors are reading "10" on Range 9.

Which ONE of the following is the percent reactor power correlating to this IRM reading?

a. 0.04%

b. 0.40%

c. 4.0%

d. 40.0%

QUESTION: 071 (1.00)

Which ONE of the following conditions will result in the closure of the Chimney Isolation Valve?

a. ONE main steam line radiation monitor is alarming high AND ONE offgas radiation monitor has been alarming high for 15 minutes.

b. ONE offgas radiation monitor is alarming downscale AND ONE offgas radiation monitor has been alarming high-high for 15 minutes.

c. ONE main steam line radiation monitor is alarming high-high AND ONE offgas radiation monitor has been alarming high-high for 15 minutes.

d. ONE main steam line radiation monitor has been alarming high for 15 minutes AND ONE main steam line radiation monitor is alarming downscale.

QUESTION: 072 (1.00)

The Cram Arrays for this fuel cycle are as follows:

Array 1 - four rods at position 16
Array 2 - four rods at position 20
Array 3 - four rods at position 24
Array 4 - four rods at position 28

Unit 2 is operating at 95% reactor power. The 2D2 Heater Drains trip with subsequent drop in the feedwater temperature. The NSO reduces recirculation flow. APRM High alarms are received for Channel 1 AND 5.

Which ONE of the following actions can the NSO take in regards to control rod movement?

a. All rods in Cram Array 1 are continuously inserted to position 8.

b. One rod in Cram Array 1 is continuously inserted to position 0.

c. All rods in Cram Arrays are continuously inserted to position 16.

d. Two rods in Cram Array 2 are continuously inserted to position 0.

QUESTION: 073 (1.00)

DEOP 400-1, "RPV Flooding", has been entered during an ATWS condition, and reactor pressure is 450 psig.

What is the MINIMUM number of ADS valves required to be OPEN to ensure ADEQUATE CORE COOLING by submergence or steam cooling?
(Select ONE of the following)

a. 2

b. 3

c. 4

d. 5

QUESTION: 074 (1.00)

Select the ONE reason that DEOP 400-5, Failure to Scram, directs operators to terminate boron injection when there is a reduction in SBLC tank level to 27 percent.

a. Cavitation of the SBLC pumps will be prevented.

b. The hot shutdown boron weight has been injected.

c. The cold shutdown boron weight has been injected.

d. A positive temperature coefficient will be prevented.

QUESTION: 075 (1.00)

A leak on the Unit 2 Instrument Air header has resulted in lowering header pressure. The "U2 INST AIR PRESS LO" annunciator alarm is received.

Which ONE of the following actions will occur?

a. The Service Air to Instrument Air Crosstie valve will open at 85 psig AND remain open until manually reset.

b. The bypass valve around Instrument Air Dryer 2B will automatically open at 60 psig to restore header pressure.

c. The bypass valve around Instrument Air Dryer 2A will automatically open at 70 psig to restore header pressure.

d. The Service Air to Instrument Air Crosstie valve will open AND close as necessary to maintain Instrument Air header pressure at 100 psig.

QUESTION: 076 (1.00)

The "AREA TEMP HI" annunciator has alarmed. Investigation reveals that the temperature in the HPCI Pump Room is 150 degrees F AND rising slowly, with the HPCI room cooler fan running.

Which ONE of the following actions will be used to mitigate this problem?

a. Startup the X-Area cooler.

b. Maximize RBCCW cooling to the HPCI room cooler.

c. Reverse service water flow through the HPCI room coolers.

d. Line up the containment cooling service water system to the area room cooler.

QUESTION: 077 (1.00)

Which ONE of the following is the MINIMUM emergency action level which REQUIRES the performance of EPIP 0400-1, Plant Assembly and Accountability"?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

QUESTION: 078 (1.00)

The Unit 2 Isolation Condenser automatically initiated during a reactor transient. The Unit Supervisor determines that the Isolation Condenser is NOT required even though an initiation signal is still present.

Which ONE of the following must be performed to assure that the RX INLET ISOL valve (MO 1301-3) will remain CLOSED?

a. Place the valve control switch in PULL-TO-LOCK.

b. Cycle the ISOL COND RESET switch in BOTH directions.

c. Place the RX INLET ISOL VLV HAND/RESET switch in HAND.

d. Hold the valve control switch closed at least 5 seconds after the full close indication has been received.

QUESTION: 079 (1.00)

With Recirculation Pump 2B running, an attempt has been made to start the 2A Recirculation pump. As the attempt is made, the 2A Recirc M-G Set Drive Motor breaker trips and the alarm is received. Further investigation reveals that the Scoop Tube Brake has actuated.

Which ONE of the following has caused the failure of the pump to start?

a. Low feedwater flow.

b. High lube oil temperature.

c. Reactor water level at +5 inches.

d. Pump speed mismatch greater than 10%.

QUESTION: 080 (1.00)

Reactor startup is in progress on Unit 2 with reactor power at 22%. Prior to startup of the Hydrogen Addition system, which ONE of the following actions is required to prevent an inadvertent reactor scram?

a. Lower the H2/FW RATIO SET setpoint to zero.

b. Place the BYP MSL HI HI RAD SIGNAL ON/OFF switches in the ON position.

c. Place the BYP MSL HI HI RAD SIGNAL ON/OFF switches in the OFF position.

d. Isolate hydrogen injection to all but one operating condensate booster pump.

QUESTION: 081 (1.00)

Which ONE of the following will cause Fuel Storage Pool Pump 2A to trip?

a. Undervoltage on Bus 29.

b. Pump suction pressure is 8 psig.

c. Filter inlet pressure is 152 psig.

d. Skimmer Surge Tank water level is 20 inches.

QUESTION: 082 (1.00)

Unit 2 was maintaining reactor water level at +30 inches prior to a loss of coolant accident. 5 seconds following the accident, reactor water level is -2 inches on all level indications and drywell pressure is 2.4 psig.

Which ONE of the following is the reactor water level setpoint signal from the Feedwater Level Control system?

a. 0 inches

b. +5 inches

c. +15 inches

d. +30 inches

QUESTION: 083 (1.00)

An ATWS is in progress on Unit 3.

Which ONE of the following systems is the LEAST desirable for maintaining reactor water level?

a. CRD

b. Core Spray

c. Low Pressure Coolant Injection

d. High Pressure Coolant Injection

QUESTION: 084 (1.00)

Unit 2 has just shutdown after a 12 month operating period. Shutdown cooling has been placed in service with the 2A AND 2B Shutdown Cooling (SDC) pumps running aligned to the reactor. The reactor water temperature is 214 deg. F. Which ONE of the following actions would be taken if the reactor coolant cooldown rate needed to be raised?

a. Start the 2C SDC pump.

b. Throttle the RBCCW Outlet Valve, MO 2-3704.

c. Open the SDC pump discharge valves on the operating SDC pumps to the desired position.

d. From fully closed, open the SDC pump suction valve to the specified position within 20 to 25 seconds.

QUESTION: 085 (1.00)

On the full core display, during normal full reactor power operations, all 177 rod drift lights come on AND the scram valves indicate closed. You also recognize four (4) scram relay white lights are off.

Which ONE of the following Immediate Operator Actions must be taken?

a. Investigate a power loss from the Instrument Bus.

b. Press the scram buttons AND place the mode switch to SHUTDOWN.

c. Verify control rods have inserted (< 02) using the Rod Worth Minimizer CRT or OD-7.

d. Direct the High Voltage Operator to Auxiliary Electrical Equipment Room to investigate cause of the power loss.

During the exam, the candidates were told that the "four scram relay lights that are off are in the same channel."

QUESTION: 086 (1.00)

Unit 2 is operating at 60% reactor power. Plant indications show that condenser vacuum is slowly dropping.

Which ONE of the following can NOT be used to maintain condenser vacuum?

a. Reduce reactor power.

b. Start the Mechanical Vacuum Pump.

c. Start the Standby Circulating Water Pump.

d. Isolate the Hydrogen AND Oxygen injection systems.

QUESTION: 087 (1.00)

Given the following plant conditions on Unit 3:

- Reactor power is 80%
- All ECCS systems are in Standby Readiness
- CRD Pump 3A is in service
- No surveillance testing is in progress at this time

Unit 3 experiences a partial loss of 125VDC power. Computer point C181, "125VDC Reserve Bus 3B-1", indicates Bus failure.

Which ONE of the following is immediately available for emergency core cooling initiated from the control room?

a. HPCI

b. CRD Pump 3B

c. LPCI Pump 3C

d. Core Spray 3B

QUESTION: 088 (1.00)

Which ONE of the following plant conditions requires entry into DEOP 300-2, Radioactive Release Control?

a. Reactor building vent radiation is 6 mr/hr.

b. Offsite release rate has been 20 x ODCM for 30 minutes.

c. Failure of the MSIVs to close on high radiation signal.

d. Two reactor building area radiation levels are above max safe levels.

QUESTION: 089 (1.00)

Unit 3 is operating at 60% rated power with CRD pump 3A tagged out
of service for maintenance. A trip of CRD pump 3B results in two control rod "ACCUMULATOR TROUBLE" alarms. The two control rods are NOT at notch 00.

Which ONE of the following is the reason that DOA 0300-01, Control Rod Drive System Failure, directs the operator to immediately scram the reactor?

a. To prevent damage to the control rod drive mechanisms due to overheating.

b. To prevent criticality in a control cell if the two rods, which are both in a three by three rod array, fail to scram.

c. To prevent reactor coolant back leakage into the CRD hydraulic accumulators AND the subsequent high area radiation conditions.

d. To ensure sufficient control rods can be fully inserted to shutdown the reactor before additional CRD problems can occur.

QUESTION: 090 (1.00)

A high torus water level condition exists on Unit 2 and the NSO is aligning LPCI to remove water from the torus.

Which ONE of the following describes where this excess torus water may be discharged?

a. Waste Surge Tank

b. Waste Sample Tanks

c. Unit 3 Hotwell

d. Unit 2 Fuel Pool

QUESTION: 091 (1.00)

During a Normal Shutdown per DGP 02-01, a HPCI inadvertent injection occurs and the following conditions develop:

- Total core flow is 27%
- Reactor power is 27%
- Reactor pressure is 783 psig

Which ONE of the following actions is required?

a. Notify the NRC within one hour.

b. Notify the state within one hour.

c. Raise reactor pressure AND be in hot shutdown within 12 hours.

d. Raise Total Core Flow as necessary to limit the probability of reactor power oscillations occurring.

QUESTION: 092 (1.00)

Unit 2 is operating at 95% reactor power. The reactor operator has withdrawn control rod G-06 to notch 24. A few seconds later a Control Rod Drift alarm comes in on rod G-06. Position indication shows the rod drifting out.

Which ONE of the following could be the cause of the control rod drift?

a. Stuck collet piston.

b. Worn drive piston seals.

c. Leaking outlet scram valve.

d. High cooling water pressure.

QUESTION: 093 (1.00)

An ERV has inadvertently opened while at operating at full reactor power.

Prompt actions by the NSO in accordance with DOA 0250-01, Relief Valve Failure, were successful in closing the open ERV.

Which ONE of the following methods can be used as a POSITIVE method to verify that the ERV has reclosed?

a. Main Generator MWe going up AND steam flow indication going down.

b. Total recirculation flow goes down AND Main Generator MWe goes up.

c. Acoustic monitoring green AND amber lights lit AND steam flow indication goes up.

d. Closed light indication at the 902-3 panel AND steam flow indication goes down.

QUESTION: 094 (1.00)

The control rod EMERGENCY IN switch is used to . . .
(Select ONE of the following)

a. override an RWM insert block.

b. individually scram a CRD under emergency conditions.

c. stop control rod insertion during planned control rod movements.

d. latch a control rod that is continuously drifting out of the core.

QUESTION: 095 (1.00)

A reactor scram has occurred AND all control rods are NOT inserted.

According to DEOP 500-05, Alternate Insertion of Control Rods, which ONE of the following methods for inserting control rods REQUIRES the Scram Discharge Volume vent and drain valves be closed?

a. Venting the overpiston area

b. Venting the scram air header

c. Pulling fuses for scram solenoids

d. Use of individual scram test switches

QUESTION: 096 (1.00)

During fuel handling activities a fuel assembly is placed in a high density storage rack. You observe that the assembly (upper tie plate) is stuck approximately eight (8) inches above the top of the rack.

Which ONE of the following would apply to this condition?

a. Verify that the hoist load condition has cleared AND continue with the fuel handling sequence.

b. With the Shift Manager's permission use the force necessary to correctly position the assembly.

c. Verify that a hoist load condition exists AND is maintained until the assembly is repositioned.

d. With permission from the Fuel Handling Supervisor, perform an interlock check per DOS 0800-01.

QUESTION: 097 (1.00)

Dresden Unit 3 is at 65% reactor power with the recirculation pumps operating at a 6% speed difference from each other.

Which ONE of the following conditions is a symptom of a jet pump failure?

a. Unexplained reduction in indicated core flow.

b. Unexplained rise in reactor power.

c. An unexplained rise in recirculation drive flow to loop flow.

d. An unexplained rise in indicated delta pressure on the jet pump riser Associated with a failed jet pump.

QUESTION: 098 (1.00)

Both units are operating at 100% reactor power when an electrical fault results in a complete loss of Division II 125 VDC power on Unit 2.

Which ONE of the following describes how this failure will affect plant operations?

a. The 2B Recirc Pump will trip.

b. Unit 2 ADS logic will not operate.

c. Unit 2 Isolation Condenser will initiate.

d. Diesel Generator 2 will not be able to start.

QUESTION: 099 (1.00)

During a reactor accident on Unit 3, the following conditions exist:

- Reactor pressure is 430 psig
- Drywell temperature is 345 degrees F
- Reactor building temperature is 215 degrees F

Which ONE of the following will give accurate reactor water level indication?

a. -16 inches on Wide Range

b. -78 inches on Fuel Zone Range

c. -54 inches on Medium Range "A"

d. -48 inches on Medium Range "B"

QUESTION: 100 (1.00)

While at 90% power condenser vacuum is observed to be decreasing.

Which ONE of the following states the expected plant response that will occur if vacuum further decreases without operator action?

a. The turbine will trip after the reactor scrams at 21" Hg vacuum.

b. The turbine will trip at 20" Hg vacuum which will cause a reactor scram.

c. The turbine will trip at 21" Hg vacuum resulting in a generator load reject.

d. The turbine will trip AND the reactor will scram from a turbine control valve fast closure at 20" Hg vacuum

(********** END OF EXAMINATION **********)

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 001 (1.00)

REFERENCE:

1. DAP 7-27, Rev. 13, Independent Verification, page 9 of 12.

294001K101

QUESTION: 002 (1.00)

REFERENCE:

1. DLP 206L-S1, Rev. 03, HPCI.

206000K605

QUESTION: 003 (1.00)

REFERENCE:

1. DLP 201L-S6, Rev. 04, Rod Worth Minimizer.
2. Technical Specification 3.4, page 3/4.3-18, Amend. No. 3/16/88.

201006K105

QUESTION: 004 (1.00)

REFERENCE:

1. DLP 232L-S1, Rev. 03, Nuclear Fuel, page 16 of 30.

234000K505

QUESTION: 005 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev. 03, Main Turbine.

295005K304

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 006 (1.00)

REFERENCE:

1. DLP 295L-S2, Rev. 1, Primary Containment Control and Primary Containment Hydrogen Control, DEOP's 200-1 and 200-2.

   295024K101

QUESTION: 007 (1.00)

REFERENCE:

1. DOP 6400-8, Rev. 7, Figure 7.

   294001A108

QUESTION: 008 (1.00)

REFERENCE:

1. DAP 3-05, Rev. 39, Out-of-Service Program, page 26 of 34.

   294001K102

QUESTION: 009 (1.00)

REFERENCE:

1. DAP 7-05, Rev. 16, Operating Charts, Logs and Records.

   294001A106

QUESTION: 010 (1.00)

REFERENCE:

1. DOA 0040-01, Rev. 15, Slow Leak.

   295027G010

QUESTION: 011 (1.00)

REFERENCE:

1. DOA 6600-01, Diesel Generator Failure, Rev. 09.
2. DLP 262L-S1, Auxiliary Power.

   262001K406

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 012 (1.00)

REFERENCE:

1. Technical Specifications 3/4.2.A, 3/4.6.K, and 3/4.6.L.

290002A204

QUESTION: 013 (1.00)

REFERENCE:

1. Technical Specification 3/4.6.1 and Safety Limit 2.1.B.

259002G005

QUESTION: 014 (1.00)

REFERENCE:

1. DAP 18-08, Rev. 02, Guideline for the Performance of Online Maintenance. The answer is based on 14 day LCO. 14 days = 336 hours. 50% of 336 = 168 hours. 168 hours is the total limit based on DAP 18-08 MPLD rules. Since Operations needs 8 hours to fill, vent and retest the system. The maintenance window is further reduced by 8 hours, thus 160 hours.
2. Technical Specification 3.5.A, page 3/4.5-3.
3. K/A: 3.6/4.2.

294001A110

QUESTION: 015 (1.00)

REFERENCE:

1. DOA 4700-01, Rev. 17, Instrument Air System Failure, page 10 of 29.
2. DOA 3900-01, Rev. 09, Loss of Cooling by the Service Water System, page 4.
3. DLP 261L-S1, Rev. 5, Standby Gas Treatment (SBGT).

261000K405

QUESTION: 016 (1.00)

REFERENCE:

1. Technical Specification 3/4.6.M and Bases.
2. DLP 239L-S1, Rev. 18, Main Steam System.

212000K307

QUESTION: 017

REFERENCE:

1. DLP 215L-S5, Rev. 05, Average Power Range Monitoring, p. 16A.

202001A412

QUESTION: 018 (1.00)

REFERENCE:

1. DEOP 100, Tables 100-E, 100-F, and 100-G.

295030A201

QUESTION: 019 (1.00)

REFERENCE:

1. DGP 02-03, Rev. 29, Reactor Scram, Immediate Operator Actions.

295006G011

QUESTION: 020 (1.00)

REFERENCE:

1. DOA 0600-01, Rev. 17, Transient Level Control, page 4 of 14.

295009A102

QUESTION: 021 (1.00)

REFERENCE:

1. DEOP 0100-00, Rev. 04, Reactor Control.
2. DEOP 0400-01, Rev. 04, RPV Flooding.

295031K101

QUESTION: 022 (1.00)

REFERENCE:

1. DEOP 0100, Rev. 04, Reactor Control.
2. DEOP 0200-00, Rev. 03, Primary Containment Control.

295010G011

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 023 (1.00)

REFERENCE:

1. DLP 271L-S1, Rev. 2, Off-Gas System, page 16A of 31A.

271000K508

QUESTION: 024 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 04, Low Pressure Coolant Injection (LPCI), page 21.

219000K409

QUESTION: 025 (1.00)

REFERENCE:

1. DLP 295L-S2, Rev. 1, DEOPs 200-1 & 200-2, Primary Containment Control and Primary Containment Hydrogen Control.

294001K115

QUESTION: 026 (1.00)

REFERENCE:

1. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

212000K202

QUESTION: 027 (1.00)

REFERENCE:

1. DLP 216L-S2, Rev. 0, Analog Trip System.
2. DLP 212L-S1, Rev. 3, Reactor Protection System.
3. DAN 903-5, C-13, Rev. 04, CHANNEL A/B RPV PRESS HI-HI.
4. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

212000K502

QUESTION: 028 (1.00)

REFERENCE:

1. DLP 212L-S1, Rev. 3, Reactor Protection System, page 20.

212000K106

QUESTION: 029 (1.00)

REFERENCE:

1. DLP 216L-S1, Rev. 03, Nuclear Boiler Instrumentation.

216000K507

QUESTION: 030 (1.00)

REFERENCE:

1. DLP 223L-S1, Rev. 2, Containment Systems.

290001A202

QUESTION: 031 (1.00)

REFERENCE:

1. DLP 202L-S1, Rev. 4, Recirculation System.

202001A109

QUESTION: 032 (1.00)

REFERENCE:

1. DAN 902(3)-5 G-3, "RPIS SYS INOP"
2. DLP 201L-S2.

214000K303

QUESTION: 033 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 04, LPCI, page 16.

203000K411

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 034 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, DEOP 100, Reactor Control, p. 35.

295007G007

QUESTION: 035 (1.00)

REFERENCE:

1. DLP 295L-S2, Rev. 1, Pri. Containment Hydrogen & Oxygen Control, page 35A.

295010A107

QUESTION: 036 (1.00)

REFERENCE:

1. DAP 21-07, Rev. 2, Control/Maintenance of Fuses & the Fuse List, page 6.

294001K107

QUESTION: 037 (1.00)

REFERENCE:

1. DFPP 4100-03, Rev. 8, E.2 p. 3.
2. DAP 03-16, Rev. 04, F.3.a, p. 9.

294001K116

QUESTION: 038 (1.00)

REFERENCE:

1. DAP 05-08, Rev. 5, Control of Temporary System Alterations, E.2.1, page 6.

294001K102

QUESTION: 039 (1.00)

REFERENCE:

1. DFP 0800-01, Rev. 28, Refueling Procedure, G.7, page 7 of 14.

294001A112

QUESTION: 040 (1.00)

REFERENCE:

1. DLP 295L-S8, Rev. 1, DEOP 400-5.

295037G012

QUESTION: 041 (1.00)

REFERENCE:

1. DOA 3300-02, Rev. 13, Loss of Condenser Vacuum, page 3 of 8, Step C.7, Note: Question delineates differences in actions between Units.

295002G007

QUESTION: 042 (1.00)

REFERENCE:

1. DOA 6900-01, Rev. 8, and Modification M12-3-95-003, changed ATS panel trouble alarm from 24/48 VDC to 125 VDC.

295004K303

QUESTION: 043 (1.00)

REFERENCE:

1. DOA 6800-01, Rev 13, Loss of Power to Essential Service System Bus or Instrument Bus, page 2 of 14.
2. DLP 259L-S1, Rev. 02, Condensate/Feedwater.
3. DLP 223L-S1, Rev. 1, PCIS.
4. DLP 209L-S1, Rev. 5, Core Spray System.

295004K303

QUESTION: 044 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, DEOP 300-2.

295017K302

QUESTION: 045 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, DEOP 300-1 and 300-2.
2. DEOP 0100, Rev. 6.

295017G012

QUESTION: 046 (1.00)

REFERENCE:

1. DOA 3700-01, Rev. 14, Loss of Cooling By Reactor Building Closed Cooling Water (RBCCW) System, Immediate Operator Actions.
2. DLP 208L-S1, Rev. 02, Reactor Building Closed Cooling Water (RBCCW), page 10A, "potential breach of primary containment".

295018G011

QUESTION: 047 (1.00)

REFERENCE:

1. EPIP 0200-T1, Rev. 07, page 1 of 89.

295016G002

QUESTION: 048 (1.00)

REFERENCE:

1. DLP 223L-S1, Containment Systems, Rev. 2.

295020K203

QUESTION: 049 (1.00)

REFERENCE:

1. EPIP 0100-01, Rev. 07, Acting Station Director Implementing Procedure, Section F.1.5, page 7.

294001A115

QUESTION: 050 (1.00)

REFERENCE:

1. DAN 902(3)-3 A-1, Rev. 08, page 1 of 3.

294001A113

QUESTION: 051 (1.00)

REFERENCE:

1. DAP 07-02, Rev. 31, page 19 of 26.

   294001A102

QUESTION: 052 (1.00)

REFERENCE:

1. DAP 12-04, Control of Access to High Radiation Areas, Rev. 27, p. 5.

   294001K114

QUESTION: 053 (1.00)

REFERENCE:

1. DAP 07-47, Rev. 01, NRC License Active Status Maintenance and Reactivation.
2. 10CFR55, parts 51, 53, 55, 57 and 59.
3. IE Notice 94-14, Supplement 1, Failure to Implement Requirements for Biennial Medical Examinations and Changes in Licensed Operator Medical Conditions, dated April 14, 1997.

   294001A102

QUESTION: 054 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, pages 7 and 21 of 31.

   259002K601

QUESTION: 055 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, page 27 of 31.

   259002K605

QUESTION: 056 (1.00)

REFERENCE:

1. DLP 264L-S1, Rev. 3, Emergency Diesel Generators Control and Operation, pages 17 and 18 of 37.

   264000K402

QUESTION: 057 (1.00)

REFERENCE:

1. DOP 0202-12, Rev. 10, Recirculation Pump Motor Generator Set Scoop Tube Operation, Section G.1.d and G.1.e, pages 4 and 5 of 8.
2. DLP 202L-S2, Rev. 04, Recirculation Flow Control.

   202002A408

QUESTION: 058 (1.00)

REFERENCE:

1. DLP 201L-S1, Control Rod Drive Hydraulic System.

   201001G007

QUESTION: 059 (1.00)

REFERENCE:

1. DLP 261L-S1, Rev. 5, SBGT, page 11A.

   261000A301

QUESTION: 060 (1.00)

REFERENCE:

1. DLP 211L-S1, Rev. 02, Standby Liquid Control (SBLC), page 5A.

   211000K301

QUESTION: 061 (1.00)

REFERENCE:

1. Technical Specification Table 2.2.A-1.

   215005A104

QUESTION: 062 (1.00)

REFERENCE:

1. DLP 215L-S2, Rev. 03, RBM System, p. 5A.

215002A305

QUESTION: 063 (1.00)

REFERENCE:

1. DOS 800-1, Rev. 18, Refueling Interlock Checks.
2. DLP 201L-S2, Reactor Manual Control System (RMCS) and Rod Position Indication System (RPIS).
3. Technical Specification 3.10.I.

234000A302

QUESTION: 064 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev. 03, Main Turbine, page 13A of 55A.

245000A312

QUESTION: 065 (1.00)

REFERENCE:

1. DOP 1900-01, Rev. 04, Fuel Pool Cooling and Cleanup System Startup.

233000A405

QUESTION: 066 (1.00)

REFERENCE:

1. DLP 207L-S1, Rev. 02, Isolation Condenser.
2. Technical Specification 3.5.D.

207000K607

QUESTION: 067 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 4, Low Pressure Coolant Injection, page 23.

262001K602

QUESTION: 068 (1.00)

REFERENCE:

1. DLP 262L-S5, Rev. 2, Low Voltage AC Distribution, Attachment B, page 2.
2. DOP 0500-03, Rev. 06, Reactor Protection System Power Supply Operation, Section F.3.d, page 4 of 14.

212000K601

QUESTION: 069 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, Reactor Control (DEOP 100), page 37.

295025A206

QUESTION: 070 (1.00)

REFERENCE:

1. DLP 215L-S3, Rev. 2, Intermediate Range Monitoring, page 15A.

215003A401

QUESTION: 071 (1.00)

REFERENCE:

1. DGA-16, Rev. 07, Coolant High Activity/Fuel Element Failure, page 4.

295017K301

QUESTION: 072 (1.00)

REFERENCE:

1. DGP 03-04, Rev. 24, Control Rod Movements.

295014A103

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 073 (1.00)

REFERENCE:

1. DLP 295L-S1, Rev. 1, RPV Flooding, DEOP 400-1 Lesson Plan, page 11A & 12A.
2. DEOP 0400-01, Rev. 04, RPV Flooding.

295031K101

QUESTION: 074 (1.00)

REFERENCE:

1. DLP 295L-S8, Rev. 1, Failure to Scram (DEOP 400-5), page 44A of 45A.

295037A203

QUESTION: 075 (1.00)

REFERENCE:

1. DOA 4700-01, Rev. 14, Instrument Air System Failure, page 2.

295019A101

QUESTION: 076 (1.00)

REFERENCE:

1. DAN 902(3)-3 G-2, Rev. 03, page 1.

295032K201

QUESTION: 077 (1.00)

REFERENCE:

1. EPIP 0100-1, Rev. 05, Acting Station Director Implementing Procedure, page 6.
2. EPIP 0400-01, Plant Assembly and Accountability, Rev. 08, page 1.

294001A116

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 078 (1.00)

REFERENCE:

1. DOP 1300-2, Rev. 10, Automatic Operation of the Isolation Condenser, E.2, page 4.

   207000A405

QUESTION: 079 (1.00)

REFERENCE:

1. DLP 202L-S3, Rev. 1, Recirc MG Set and Auxiliaries, page 26 and Table 1.

   202002A303

QUESTION: 080 (1.00)

REFERENCE:

1. DGP 01-01, Rev. 73, Unit Startup, Section G.117, p 52.

   272000K501

QUESTION: 081 (1.00)

REFERENCE:

1. DLP 233L-S1, Rev. 12, Fuel Pool Cooling, p. 15.

   233000K601

QUESTION: 082 (1.00)

REFERENCE:

1. DLP 259L-S2, Rev. 04, Feedwater Level Control System, page 25.

   295006K202

QUESTION: 083 (1.00)

REFERENCE:

1. DLP 295L-S8, Rev. 1, Failure to Scram, p. 25A.

   295031K203

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 084 (1.00)

REFERENCE:

1. DOP 1000-03, Rev. 21, Shutdown Cooling Mode of Operation, pages 10, 11 and 12.

   295021K103

QUESTION: 085 (1.00)

REFERENCE:

1. DOA 6800-01, Rev. 13, Loss of Power to the ESS Bus or Instrument Bus, page 3 of 14.
2. DAN 902(3)-8 F-8, Rev. 03, ESS UPS TROUBLE.

   295003K103

QUESTION: 086 (1.00)

REFERENCE:

1. DOA 3300-02, Rev. 13, Loss of Condenser Vacuum, page 3.

   295002K207

QUESTION: 087 (1.00)

REFERENCE:

1. DOA 6900-T1, Rev. 7. Page 23, Bus 34-1 supplies breaker control power to LPCI Pump 3C and Core Spray Pump 3B. Page 24, Bus 34 supplies control power to CRD Pump 3B. Page 25, line item 20, HPCI loses normal power feed and transfers to alternate feed.

   295004A102

QUESTION: 088 (1.00)

REFERENCE:

1. DLP 295L-S3, Rev. 1, Radiation Release Control (DEOP 300-02), page 25A.

   295038G011

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 089 (1.00)

REFERENCE:

1. DOA 0300-01.
2. Technical Specification 3.3.G.

295022K301

QUESTION: 090 (1.00)

REFERENCE:

1. DLP 203L-S1, Rev. 4, Low Pressure Coolant Injection (LPCI), page 8.

295029K201

QUESTION: 091 (1.00)

REFERENCE:

1. Technical Specification 6.7.A, p. 6-8 & Technical Specification 2.1.A, page 2-1.

294001A105

QUESTION: 092 (1.00)

REFERENCE:

1. DLP 201L-S3, Rev. 02, CRD Blade and Mechanism, page 27A.

201003A203

QUESTION: 093 (1.00)

REFERENCE:

1. DOA 0250-01, Rev. 15.
2. DLP 239L-S1, Rev. 17, Main Steam System.

295013A102

QUESTION: 094 (1.00)

REFERENCE:

1. DOA 0300-05, Rev. 11, Inoperable or Failed Control Rod Drives.

295014K208

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 095 (1.00)

REFERENCE:

1. DEOP 0500-05, Rev. 06, Alternate Insertion of Control Rods, page 4 of 14, "Failure to perform the following steps in order may result in an open flowpath from the RPV to the Reactor Building via the SDV Vents and Drains".

295015A101

QUESTION: 096 (1.00)

REFERENCE:

1. DFP 0800-32, Rev. 08, Fuel Movements Within the Spent Fuel Pool, page 3, item E.2.b.

295023K201

QUESTION: 097 (1.00)

REFERENCE:

1. DOA 0201-01, Rev. 06, Jet Pump Failure, page 2 of 4.
2. DLP ILTS026, Rev. 1, Reactor Related DOA Training.

295001A205

QUESTION: 098 (1.00)

REFERENCE:

1. DOA 6900-02, Rev. 05, Failure of Unit 2 125 VDC Power Supply, page 4.

295004A202

QUESTION: 099 (1.00)

REFERENCE:

1. DEOP 0100, Rev. 04, Reactor Control, Table 100-C.

295028K101

SENIOR REACTOR OPERATOR WRITTEN EXAM

QUESTION: 100 (1.00)

REFERENCE:

1. DLP 245L-S1, Rev 03, Main Turbine.
2. DLP 212L-S1, Rev 03, Reactor Protection System.

295002K103

Written Examination Summary - Applicant Handouts

Reactor Operator

1. DOP 6400-8, Rev. 7, Figure 7, "Regulator: Out of Service"
2. Steam Tables and calculator
3. Technical Specification Table 2.2.A-1
4. DEOP 0400-01, Rev. 4, RPV Flooding (entry condition setpoints removed)

RO Total: Steam tables, calculator, and three pages of handouts

Senior Reactor Operator

1. DOP 6400-8, Rev. 7, Figure 7, "Regulator: Out of Service"
2. Steam Tables and calculator
3. Tech Spec 3.5.A, page 3/4.5-3
4. DEOP 0100, Tables 0100-C, 0100-E, 0100-F and 0100-G
5. EPIP 0200-T11, Rev. 7, page 1 of 89
6. Technical Specification Table 2.2.A-1 (entry condition setpoints removed)

SRO Total: Steam Tables, calculator, and eight pages of handouts