March 14, 2000

Mr. D. R. Gipson
Senior Vice President
Nuclear Generation
The Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: OPERATOR LICENSING EXAMINATION REPORT 50-341/2000301(DRS)

Dear Mr. Gipson:

The Nuclear Regulatory Commission examiners completed initial operator licensing examinations at your Fermi 2 Nuclear Station on February 4, 2000. The license applicants' performance evaluations were finalized on February 28, 2000. During the examination preparation, validation, and administration, the examiners reviewed several administrative and operating procedures. The enclosed report presents the results of the examination and concurrent operation's inspection.

The examiners administered operating and written examinations to one reactor operator and five senior reactor operator license applicants. All six applicants passed all sections of the examination and were issued operating licenses. The applicants were well prepared for the examination. The licensed shift operators involved in the examination validation provided good insight for improving examination quality. Your training staff provided satisfactory support during the examination process.

It is our expectation that the Fermi training department will use the examination and applicant deficiencies outlined in the accompanying report as feedback to improve the operator license training program in accordance with your Systematic Approach to Training program.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. Specifically, a lack of understanding and sensitivity on the part of some plant staff regarding examination security measures resulted in three instances in which examination integrity was adversely affected. However, this violation is being treated as a Non-Cited Violation (NCV), consistent with Appendix C of the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III, and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington DC 20555-0001.

IE42

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter and the enclosures to this letter will be placed in the NRC Public Document Room.

We will gladly discuss any questions you have concerning this examination.

Sincerely,

David E. Hills

David E. Hills, Chief
Operations Branch
Division of Reactor Safety

Docket No. 50-341
License No. NPF-43

Enclosures:
1. Inspection Report 50-341/2000301(DRS)
2. Simulation Facility Report
3. Written Examination and Answer Keys (RO and SRO)

cc w/encls 1 & 2:
N. Peterson, Director, Nuclear Licensing
P. Marquardt, Corporate Legal Department
Compliance Supervisor
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Monroe County, Emergency Management Division
Emergency Management Division
MI Department of State Police

cc w/encls 1, 2 & 3: L. D. Sanders, Training Department

DOCUMENT NAME: G:DRS\FER2000301DRS.WPD
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IE-42

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-341
License No:	NPF-43
Report No:	50-341/2000301(DRS)
Licensee:	The Detroit Edison Company
Facility:	Fermi 2
Location:	6400 North Dixie Highway Newport, MI 48166
Dates:	January 31 - February 4, 2000
Examiners:	M. Bielby, Chief Examiner, RIII H. Peterson, Examiner, RIII R. Vogt-Lowell, Resident Inspector - Perry Station
Approved by:	David E. Hills, Chief, Operations Branch Division of Reactor Safety

EXECUTIVE SUMMARY

Fermi 2 Nuclear Power Plant
NRC Inspection Report 50-341/2000301(DRS)

A licensee developed and NRC approved initial operator licensing examination was administered by the NRC to six operator license applicants. One applicant applied for a Reactor Operator (RO), and five applied for Senior Reactor Operator (SRO) licenses. Two of the SRO applicants were previously licensed at the facility as ROs. The examination process included development, validation, and administration of a written and operating examination to each applicant.

Examination Results:

All applicants passed all portions of the examination and were issued operating licenses.

Examination Preparation and Administration:

The licensee's shift operators and license applicants were able to use the provided administrative and operating procedures efficiently and correctly during the operating examination validation and administration (Section O3.1).

The examiners considered the licensee's submitted outline and proposed examination to be satisfactory. The licensed shift operators involved in the examination validation provided good insight for improving examination quality (Sections O4.1 and O5.2).

The licensee's training staff satisfactorily administered the written examination and provided satisfactory support during the operating examination. The licensee's submittal of post examination documentation was satisfactory. There were no post examination comments (Sections O5.3 and O5.6).

The applicants were confident and well prepared for the operating and written examination based on overall examination results. However, the examiners identified some knowledge and performance deficiencies. In particular, at least half of the applicants incorrectly answered the same six written examination questions, and several applicants failed to verify three phase voltage values during an Emergency Diesel Generator paralleling job performance measure (Section O5.4).

The licensee had appropriate procedures established to control examination security. However, the examiners identified a non-cited violation involving a lack of understanding and sensitivity on the part of some plant staff regarding examination security measures that resulted in three instances in which examination integrity was adversely affected. These instances involved leaving an examination material locker unlocked, leaving a computer disk containing the examination scenarios unattended in a non-secure location, and an unauthorized individual entering the examination area contrary to posted signs (Section O5.5).

Report Details

I. Operations

O3 Operations Procedures and Documentation

O3.1 General Comments

a. Scope (71707)

The examiners reviewed portions of selected administrative and operating procedures during review, validation, and administration of the initial license examination using Inspection Procedure 71707. See the end of this report for a partial list of procedures reviewed.

b. Observations and Findings

The NRC examiners observed that most procedures were well organized and normally used correctly by the shift operators and by the license applicants during the operating examination validation and administration.

At least half of the applicants had difficulty properly executing one System Operating Procedure (SOP 23.307) during their system Job Performance Measure (JPM) examination (Section O5.4).

c. Conclusions

The licensee's shift operators and license applicants were able to use the provided administrative and operating procedures efficiently and correctly during the operating examination validation and administration.

O4 Operator Knowledge and Performance

O4.1 General Comments-Licensed Shift Operators

The examiners observed performance of three shift operators who validated the operating examination. The operators provided good insights for improving examination quality on nine administrative and ten system JPMs, and three dynamic simulator scenarios. The operators demonstrated satisfactory performance and knowledge of their responsibilities during the validation.

O5 Operator Training and Qualification

O5.1 General Comments-Initial Operator License Examination

The licensee submitted the proposed written and operating examination ahead of schedule. The licensee examination quantitatively contained the required written questions, JPMs and scenarios to adequately evaluate the applicants. The examiners reviewed and discussed examination comments with the licensee in the regional office during the week of January 3, 2000, and validated the examination at the site during the week of January 10, 2000.

O5.2 Pre-Examination Activities

a. Scope

The licensee's training staff prepared and submitted an outline and proposed initial operator license examination to the NRC for review and comment.

b. Observations and Findings

The licensee's training staff used the guidance prescribed in NUREG 1021, Operator Licensing Examination Standards for Power Reactors (ES), Revision 8, dated April 1999, to prepare the outline, operating and written examinations. The licensee submitted the proposed outline and examination to the NRC ahead of schedule. The NRC examiners reviewed and commented on the licensee's proposed examination submittal, and the licensee effected changes agreed upon between the NRC and the facility licensee in accordance with NUREG 1021 prior to the validation week. Subsequent to the incorporation of NRC examiner comments and final review, a potential examination compromise (Section O5.5) necessitated the replacement, review, and re-validation of three new examination scenarios.

c. Conclusions

The examiners considered the licensee's submitted outline and proposed examination to be satisfactory.

O5.3 Examination Administration

a. Scope

The NRC examiners administered the examination using the guidance prescribed in Sections ES-302 and ES-402 of NUREG 1021. The examiners administered the operating examination, consisting of JPMs and dynamic scenarios, February 1 - 3, 2000. The licensee's training staff administered the written examination on February 4, 2000.

b. Observations and Findings

Operating Examination

The training staff's support of the examination process was satisfactory. Turnover sheets, surveillances, procedures, and other paperwork required to support the administration of the operating examination were generally correct and detailed. Shift turnovers provided by the instructors during the dynamic scenarios were typical of those used during the training process and varied slightly from those used in the control room. The licensee's daily setup and execution of the operating portion of the examination during the validation and administration weeks was timely and generally accurate. However, during administration of the dynamic scenarios, the examiners observed one instance of procedures not being replaced after being marked by applicants during a previous run of the same scenario.

Generally, the scenarios performed satisfactorily. However, during the initial run of Scenario 1 major transient (recirc loop rupture with loss of high pressure feedwater), Residual Heat Removal (RHR) was not required to restore reactor water level (RWL) as originally validated. The event was twice validated by an operating shift crew to ensure an adequate line break size after emergency depressurization that would require the operator to re-align and manually initiate the non-selected Low Pressure Coolant Injection (LPCI) RHR loop to restore the decreasing RWL. During the next administration of the scenario, the lead examiner directed the simulator operator to increase the break size to require the same effect. The examiners determined that the scenario remained discriminating, but not as challenging as intended.

The examiners also noted that on one occasion during performance of the administrative JPM for performing a short term relief, the Reactor Recirculation Sample Valves, B3100-F019 and -F020, were inadvertently left open during the simulator setup.

Written Examination

The licensee administered the written examination in their training center classroom. The room contained a satisfactory arrangement of tables and spacing between the applicants. An instructor read the examination requirements to the applicants prior to the examination and verified that applicants had all of the required examination references and materials. The examination was completed within the allowable five hours.

c. Conclusions

The licensee's training staff satisfactorily administered the written examination and provided satisfactory support during the operating examination.

O5.4 License Applicant Performance

a. Scope

The NRC examiners administered the operating examination, and the licensee administered the written examination to one RO and five SRO applicants. The examiners evaluated the applicants' performance using dynamic scenarios, JPMs, and written examinations in accordance with ES-303, "Documenting and Grading Initial Operating Tests," and ES-403, "Grading Initial Site-Specific Written Examinations," contained in NUREG 1021, Revision 8.

b. Observations and Findings

Operating Examination

The NRC examiners determined that the overall performance of all six applicants during the dynamic scenario examination was satisfactory. The examiners identified individual discrepancies, but did not consider any generic. The examiners identified that applicants quickly found and correctly executed appropriate procedures. Applicants generally communicated clearly and accurately using three-way communications. The examiners noted two instances when applicants were speaking over one another, but no mis-communications resulted. Applicants in control board positions performed self-checking during normal and transient conditions. While in the shift supervisor position, the applicants generally maintained their position of oversight, performed peer checks during normal evolutions, and conducted informative briefings at appropriate times.

Although the examiners identified individual discrepancies, the overall performance of all six applicants during the administrative and system JPMs was satisfactory. Examiners identified one generic deficiency during performance of the system JPM for paralleling Emergency Diesel Generator (EDG) 14 from the Control Room to Bus 65F. Applicants were expected to use SOP 23.307 to start and parallel EDG 14 to Bus 65F. A procedural caution following Step 6 reminded applicants of the potential consequence of failure to ensure EDG output voltage was greater than the bus voltage on each of the respective three phases before synchronizing and closing the EDG output breaker. At least three of the applicants failed to have the local operator verify the voltage values on all three phases; however, the action was not considered critical in that instance.

Written Examination

All six of the applicants passed the written examination with scores ranging from 87 percent to 96 percent. The licensee performed a preliminary assessment of the common questions that were incorrectly answered on the written examination. The licensee considered questions as potential generic knowledge deficiencies based on incorrect answers by at least half of the applicants. The licensee trained the applicants on the potential knowledge deficiencies during their post-exam review. Furthermore, as part of their corrective action program, the licensee wrote Corrective Action Resolution Document (CARD) 99-16945, to have an analysis done to determine if the content of the initial license operator program would need to be changed to address the weaknesses.

Question #	(% incorrect) Potential Knowledge Weakness
#7(RO/SRO)	(50%) Misconception that the non-Automatic Depressurization System (ADS) Safety Relief Valves do not transfer to the alternate power supply upon failure of the primary power supply.
#15(RO/SRO)	(83%) Knowledge of the start interlock between the Reactor Recirc Motor Generator and Drive Motor Breaker position.

#28(RO/SRO)	(50%) Knowledge of whether the main turbine generator mechanical overspeed trip testing on-load device was prevented electrically or mechanically.
#33(RO/SRO)	(50%) Knowledge of why control rods did not insert based on analysis of Anticipated Transient Without Scram (ATWS) conditions.
#37(RO/SRO)	(50%) Knowledge of the Digital Load Sequencer operation after closure of the EDG output breaker.
#88(SRO)	(60%) Knowledge of the bases for limiting Drywell to Torus differential pressure.

c. Conclusions

The applicants were confident and well prepared for the operating and written examination based on overall examination results. However, the examiners identified some knowledge and performance deficiencies. In particular, at least half of the applicants incorrectly answered the same six written examination questions, and several applicants failed to verify three phase voltage values during an Emergency Diesel Generator paralleling job performance measure.

O5.5 Examination Security

a. Scope

The examiners reviewed and observed the licensee's implementation and controls of examination security during the examination preparation and administration. The examiners reviewed the licensee's procedure for maintaining examination security, Operations Training Policy (OTP) OTP-020, "Examination Safeguards and Controls," Revision 0, dated September 9, 1999.

b. Observations and Findings

The examiners determined that the licensee had established appropriate procedures to control examination security. The licensee training staff coordinated the arrival times of the applicants and provided escorts to maintain examination security during administration of the operating examination. However, during the initial license examination process, the examiners noted three events adversely affecting the integrity of the NRC examination.

The first event occurred during examination preparation by the licensee. Licensee personnel discovered the initial license examination room examination material locker (filing cabinet) was unlocked and unattended at approximately 1:30 p.m. on December 28, 1999. The filing cabinet and combination lock device were in good condition and not physically disturbed. The following actions were taken:

- Review of OTP-020 examination security procedure to verify requirements;
- Inventory of material in the locker revealed none were missing or the placement disturbed;
- Security was contacted and verified no door alarms during the time when the room was unattended (December 22 - 28, 1999); and
- Notification of NRC Region III Chief Examiner.

The licensee took short term corrective action to counsel responsible personnel and generated CARD 99-19040 to track the event for subsequent evaluation and long term corrective actions.

The second event occurred after the examination had been prepared by the licensee, but prior to the examination administration week. On January 28, 2000, a licensee training instructor discovered that a floppy disk containing NRC initial license examination dynamic scenarios had been inadvertently left inserted in the computer at the simulator console. The instructor turned the disk over to the Operations Training Supervisor who secured the disk. The licensee conducted initial interviews and believed that none of the examination material was viewed by any applicants, trainees or others that had not signed the security agreement. The NRC Region III Chief Examiner was notified at 10:00 a.m., January 28, 2000. Additional discussions between the licensee and Chief Examiner noted the following approximate time-line:

January 27, 2000	
1:30 p.m.	Examination preparer completed validation of the scenarios, cleared the simulator and computer memory of scenario traces, and turned the simulator over to a License Operator Requalification Training (LORT) instructor that was preparing training events for the Initial License Operator applicants.
3:30 p.m.	LORT instructor left for the day, but allowed the applicants to continue training on their own recognizance since at least two of them had the required knowledge to operate the simulator.
4:00 p.m.	Another staff instructor returned, shutdown, and cleared the simulator computer memory for the day.
January 28, 2000	
7:30-10:00 a.m.	Computer restarted, floppy disk discovered, initial interviews conducted, Chief Examiner notified of the potential scenario examination compromise.

Due to concerns about examination integrity, the licensee took short term corrective action to prepare three new initial license examination scenarios prior to the scheduled examination week utilizing their full, normal Quality Assurance process that included management review. The NRC examiners arrived at the site one day prior to the scheduled examination week to perform a preliminary review of the new scenarios. On January 31, 2000, the NRC examiners completed their review and validation in accordance with NUREG 1021, Section ES-301 and Appendix D. The examiners utilized a shift operating crew during the validation. The licensee incorporated all comments and changes prior to the scheduled start of the operating examination on

February 1, 2000. The licensee generated CARD 99-9040 to track the event for subsequent evaluation and long term corrective actions.

The third event occurred during the examination administration on Tuesday, February 1, 2000. A person not signed on the security agreement, violated a sign posting on the simulator door during the NRC administration of one dynamic scenario. The individual was immediately apprehended by the simulator operator and escorted out of the simulator. At the conclusion of the scenario, the Chief Examiner discussed the incident with the Operations Training Supervisor and simulator operator, and determined the scenario had not been compromised based on the short duration of the incident and tall panels that blocked the sight of vision of the individual. However, the examination integrity had been adversely impacted. During the remainder of the operating examination, a member of the training staff was continuously posted at the simulator door entrance to create a positive control of personnel entering the simulator.

These three examination security events are examples of a violation of 10 CFR 55.49, "Integrity of Examinations and Tests," which requires that facility licensees shall not engage in any activity that compromises the integrity of any examination required by 10 CFR Part 55 (50-341/2000301-01(DRS)). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is entered in the licensee's corrective action program as CARD 99-16945 and CARD 99-19040.

c. Conclusions

The licensee had appropriate procedures established to control examination security. However, the examiners identified a non-cited violation involving a lack of understanding and sensitivity on the part of some plant staff regarding examination security measures that resulted in three instances in which examination integrity was adversely affected. These instances involved leaving an examination material locker unlocked, leaving a computer disk containing the examination scenarios unattended in a non-secure location, and an unauthorized individual entering the examination area contrary to posted signs.

O5.6 Post Examination Activities

a. Examination Scope

The NRC examiners independently graded the written examinations and compared their results to the licensee's in accordance with form ES-403-1, "Written Examination Grading Quality Assurance Checklist." The examiners evaluated individual applicant performance and reviewed the licensee's post examination documentation in accordance with Sections ES-303, ES-403, and ES-501, of NUREG 1021.

b. Observations and Findings

The examiners' evaluations and documentation captured the individual applicant performance deficiencies. The licensee's post examination submittal included the necessary documentation in accordance with ES-501, "Initial Post Examination Activities." The submittal included an analysis of the written examination results and matrix of incorrectly answered questions by more than 50 percent of the applicants. The licensee's analysis identified and incorporated feedback of potential knowledge deficiencies to their training program (Section O5.4). The licensee did not submit any post examination comments.

c. Conclusions

The licensee's submittal of post examination documentation was satisfactory. There were no post examination comments.

O5.7 Simulator Fidelity

a. Examination Scope

The examiners observed operation and fidelity of the licensee's plant specific simulator during the operating examination.

b. Observations and Findings

The simulator performed satisfactorily throughout the NRC license examination.

c. Conclusions

The simulator performed satisfactorily throughout the examination with no noted deficiencies (Enclosure 2, Simulation Facility Report).

V. Management Meetings

X1 Exit Meeting Summary

The chief examiner presented the examination team's observations and findings to members of the licensee's management on February 4, 2000. The licensee acknowledged the findings presented and indicated that no proprietary information had been identified during the examination or at the exit meeting.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. O'Connor, Assistant Vice President
J. Davis, Director, Nuclear Training
L. Sanders, General Supervisor, Operations Training
K. Snyder, Supervisor, Operations Training
S. Stasek, Supervisor, Independent Safety Engineering Group

NRC

S. Campbell, Senior Resident Inspector, Fermi

INSPECTION PROCEDURES USED

IP 71707: Plant Operations

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-341/2000301-01	NCV	Three examples of failure to maintain integrity of the NRC license examination due to inadequate security practices which is a violation of 10 CFR 55.49 (Section O5.5).

Closed

50-341/2000301-01	NCV	Three examples of failure to maintain integrity of the NRC license examination due to inadequate security practices which is a violation of 10 CFR 55.49 (Section O5.5).

Discussed

None

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without Scram
CARD	Corrective Action Resolution Document
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ES	Examiner Standards (NUREG 1021, Operator Licensing Examination Standards for Power Reactors, Revision 8, April 1999)
HVAC	Heating, Ventilation, Air Condition
JPM	Job Performance Measure
LORT	License Operator Requalification Training
LPCI	Low Pressure Coolant Injection
NRC	Nuclear Regulatory Commission
OTP	Operations Training Policy
RHR	Residual Heat Removal
RO	Reactor Operator
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWL	Reactor Water Level
SLC	Standby Liquid Control System
SOP	System Operating Procedure
SRO	Senior Reactor Operator

PARTIAL LIST OF PROCEDURES REVIEWED

Abnormal Operating Procedure (AOP):
AOP 20.106.07, "Control Rod Drift," Revision 16; Immediate Actions, Subsequent Actions, and Symptoms - Multiple Control Rod Drift
AOP 20.000.19, "Shutdown from Outside the Control Room," Revision 28; Steps 10 - 17

Emergency Operating Procedure (EOP):
EOP 29.100.01, Sheet 1, "RPV Control," Revision 7
EOP 29.100.01, Sheet 1A, "RPV Control - ATWS," Revision 7
EOP 29.100.01, Sheet 2, "Primary Containment Control," Revision 6
EOP 29.100.01, Sheet 3, "RPV Flooding, Emergency Depressurization, & Steam Cooling," Revision 5
EOP 29.100.01, Sheet 3A, "RPV Flooding & Emergency Depressurization - ATWS," Revision 7
EOP 29.100.01, Sheet 5, "Secondary Containment and Radiation Release," Revision 6
EOP 29.100.01, Sheet 6, "Curves, Cautions, and Tables," Revision 7

System Operating Procedure (SOP):
SOP 23.307, "Emergency Diesel Generator System," Revision 63; Section 6.1, Paralleling From the Control Room
SOP 23.404, "Standby Gas Treatment System," Revision 35; Section 6.1, SGTS (Standby Gas Treatment System) Manual Startup
SOP 23.316, "RPS (Reactor Protection System) 120V (volt) AC (Alternating Current) And RPS MG (Motor Generator) Sets," Revision 39; Section 5.0, Powering RPS Bus A(B) from RPS Alternate Transformer A(B)
SOP 23.205, "Residual Heat Removal," Revision 67; Section 7.5, Forced LPCI Loop Select Logic Operation
SOP 23.707, "Reactor Water Cleanup," Revision 95; Section 8.2, Blowdown Operation
SOP 23.127, "Reactor Building Closed Cooling Water/Emergency Equipment Cooling Water System," Revision 69; Section 7.2, RBCCW (Reactor Building Closed Cooling Water) Restoration Following EECW (Emergency Equipment Cooling Water) SYSTEM Auto/Manual Initiation
SOP 23.413, "Control Center HVAC (Heating, Ventilation, Air Conditioning)," Revision 54; Section 7.10, Control Center HVAC Manual Mode Shift To Chlorine

General Administration Conduct Manual, MGA04, "Temporary Change Notices," Revision 7
Operations Conduct Manual MOP07, "Shift Turnover," Revision 2; Section 2.3, Short Term Relief
Operations Training Policy, OTP-020, "Examination Safeguards and Controls," Revision 0
Radiation Protection Conduct Manual, MRP12, "Requesting Dose Extensions," Revision 3

Enclosure 2

SIMULATION FACILITY REPORT

Facility Licensee: Fermi 2

Facility Licensee Docket No: 50-341

Operating Examinations Administered: February 1 - 3, 2000

The following documents observations made by the NRC examination team during the initial license examination. 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 examinations, the following items were observed:

ITEM	DESCRIPTION

1. None

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information	
Name: MASTER EXAMINATION	Region III
Date: FEBRUARY 4, 2000	Facility/Unit: Fermi 2
License Level: RO	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

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

______________________________________ Applicant's Signature

Results

Examination Value	100	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

Question 1.

Reactor pressure is 950 psig and the SLC system has been initiated. The operator has the following indications:

- Run indication for the 'A' pump.
- The SLC tank level is decreasing.
- Pump discharge pressure is 1000 psig.
- The squib valve continuity indicating light is LIT.

The SLC system is:

______ A. NOT injecting. The squib valves have not fired.

______ B. NOT injecting. There is a rupture in the injection line.

______ C. injecting. The continuity indication is not always lost.

______ D. injecting. The continuity indication illuminates on loss of continuity.

Point Value: 1

Question 2.

The plant was operating at 70% rated power when an increase in Main Generator megawatts was observed. The positions of the control rods and recirculation flow rate have not changed.

Which ONE of the following is the cause of the INCREASE in Main Generator megawatts?

______ A. An SRV has inadvertently lifted.

______ B. A turbine bypass valve is partially open.

______ C. Xenon is building toward a peak in the core.

______ D. There has been a loss of feedwater heating.

Point Value: 1

Question 3.

The reactor is in Mode 2, with reactor pressure at 800 psig when the operating CRD pump trips. Before any action can be taken annunciator 3D10, CRD ACCUMULATOR TROUBLE, alarms.

What action is required in accordance with procedure 20.106.01, CRD Hydraulic System Failure?

______ A. Immediately start the standby CRD pump.

______ B. Immediately place the mode switch in SHUTDOWN.

______ C. Within 20 minutes, close C11-F034, Charging Water Header Isolation Valve.

______ D. Within 20 minutes restart at least one CRD pump or place the mode switch in SHUTDOWN.

Point Value: 1

Question 4.

The plant is at 80% power when the turbine trips. The reactor will scram in anticipation of the rapid:

______ A. INCREASE in thermal power

______ B. INCREASE in reactor water level

______ C. DECREASE in reactor water level

______ D. DECREASE in main steam line pressure

Point Value: 1

Question 5.

A reactor startup is in progress with the mode switch in START-UP. Due to a malfunction, the 24/48 V Division 1 bus is de-energized. What is the plant response to this event?

______ A. Half scram, rod block

______ B. Full scram, rod block

______ C. Full scram, no rod block

______ D. Rod block, no half or full scram

Point Value: 1

Question 6.

The following plant conditions exist:
- ADS has received a valid initiation signal.
- The ADS timer is at 86 seconds and lowering.

Which ONE of the following will reset the ADS timer.

_____ A. 'A' and 'B' core spray pumps trip off.

_____ B. Drywell pressure decreases to 1.5 psig.

_____ C. Reactor water level increases to 55 inches.

_____ D. RHR pump discharge pressure decreases to 122 psig.

Point Value: 1

Question 7.

The 130 VDC Distribution Cabinet 2PA2-5 has been de-energized due to a fire. Which SRVs can be opened using the control room pushbuttons?

_____ A. All the SRVs

_____ B. Only the ADS SRVs

_____ C. Only the non-ADS SRVs

_____ D. All the ADS and some of the non-ADS SRVs

Point Value: 1

Question 8.

The following plant conditions exist:

- Reactor power 28% and stable.
- Reactor recirculation Pump 'A' is shutdown, the loop is NOT isolated.
- Reactor Recirculation Loop 'B' flow is 51% and stable.

Under which of the following conditions is an increase in core flow prohibited?
(Consider each condition separately.)

_____ A. The temperature difference between the two loops is 45° F.

_____ B. Isolation of the recirculation loop 'A' from the reactor pressure vessel.

_____ C. The temperature difference between the reactor pressure vessel coolant and the coolant in loop A is 60° F.

_____ D. The temperature difference between the reactor vessel steam space coolant and the reactor pressure vessel bottom head drain line coolant is 135° F.

Point Value: 1

Question 9.

When the mode switch is placed in SHUTDOWN a scram signal is initiated. After a short time this signal is automatically bypassed.

Which ONE of the following describes why this automatic bypass occurs?

_____ A. allows the scram to be reset.

_____ B. prevents damage to the CRDM seals.

_____ C. prevents the scram discharge volume from over filling.

_____ D. prevents the scram solenoid pilot valves from overheating.

Point Value: 1

Question 10.

The Emergency Operating Procedures require emergency depressurization if torus water level cannot be maintained above -38 inches.

What is the basis for requiring emergency depressurization at this point?

_____ A. Prevent exceeding LPCI NPSH requirements.

_____ B. Prevent exceeding SRV tailpipe back pressure limits.

_____ C. Steam discharged from RCIC will not be suppressed.

_____ D. Suppression of steam discharged from the RPV cannot be assured.

Point Value: 1

Question 11.

The alternate boron injection flowpath is from the precoat tank to a portable pump that discharges directly to the:

______ A. Heater Feed Pump suction.

______ B. Heater Feed Pump discharge.

______ C. Standby Feedwater Pump suction.

______ D. Standby Feedwater Pump discharge.

Point Value: 1

Question 12.

A loss of 120 KV offsite power has occurred.
EDG 11 has started and loaded.
EDG 12 did NOT start.

Which one of the following prevented EDG 12 from starting?

______ A. Exciter Trip

______ B. Bus lockout on bus 64C

______ C. Fuel oil day tank level of 12"

______ D. Lube oil temperature of 165° F

Point Value: 1

Question 13.

Select the two materials used as neutron absorbers in the Control Rod Blades.

_____ A. Cadmium and Boron Carbide

_____ B. Hafnium and Stainless Steel

_____ C. Hafnium and Boron Carbide

_____ D. Cadmium and Stainless Steel

Point Value: 1

Question 14.

Which ONE of the following is the power supply to Division 1 RHR LPCI Injection Mode Initiation Logic?

_____ A. MPU 1

_____ B. Division 1 130VDC

_____ C. Division 1 24/48 VDC

_____ D. Bus 72CF (swing bus)

Point Value: 1

Question 15.

Which ONE of the following indicates the required positions for the listed components to satisfy the Reactor Recirculation Pump and Motor Generator Set (RRMG) starting interlocks.

	Field Breaker	Drive Motor Breaker	Discharge Valve
______ A.	Closed	Open	Closed
______ B.	Open	Open	Closed
______ C.	Open	Closed	Closed
______ D.	Closed	Closed	Closed

Point Value: 1

Question 16.

During a plant transient the following sequence of events occur.
- Drywell Pressure rises and stabilizes at 5 psig .
- RPV decreases to 105 inches.
- HPCI automatically starts and injects.
- HPCI subsequently trips on high water level.

Which ONE of the following is required to automatically restart the HPCI System?

______ A. Reactor water level drops below Level 2.

______ B. Reactor water level drops below Level 3.

______ C. Reactor water level drops below Level 8.

______ D. The HPCI Initiation Signal Reset pushbutton is depressed.

Point Value: 1

Question 17.

Following an automatic actuation of the Core Spray System, an electrical fault closes E2150-F036A, Div 1 CS Pmps Torus Suct Vlv.

Which ONE of the following completes the statement concerning the expected response of the Core Spray system to this event?

The 'A' Core Spray Pump will:

______ A. trip on low Suction Pressure.

______ B. continuously operate until damage occurs.

______ C. continuously operate with suction from the Condensate Storage Tank.

______ D. continuously operate with suction from the Div 2 suction cross-connect.

Point Value: 1

Question 18.

While retracting SRM B during a startup, it is observed that the SRM B RETRACT PERMIT light goes out.

Which ONE of the following describes why the SRM RETRACT PERMIT light went out?

______ A. SRM B failed upscale.

______ B. SRM B failed downscale.

______ C. The mode Switch was placed in run.

______ D. IRM E was down-ranged to range 2.

Point Value: 1

Question 19.

Following a LOCA from 95% power the following conditions exist:

- Reactor power (APRMs) is 0% and stable.
- RPV Level has decreased to 144" and is steady
- Reactor pressure is 750 psig and stable.
- Drywell pressure is 2.5 psig and stable.
- NO area temperature or radiation alarms have actuated
- NO operator actions have been performed since the time of the automatic scram

Based on these conditions, which of the following PCIS isolation signals are present?

- Group 1 Main Steam
- Group 2 Reactor Water Sample
- Group 3 RHR
- Group 8 RCIC
- Group 11 RWCU Outboard
- Group 13 Drywell Sumps
- Group 16 Nitrogen Injection
- Group 18 Primary Containment Pneumatics

______ A. Groups 1, 3, 8, 13, & 18

______ B. Groups 1, 2, 11, 13, & 16

______ C. Groups 2, 3, 13, 16, & 18

______ D. Groups 1, 2, 3, 13, 16, & 18

Point Value: 1

Question 20.

Late in the operating cycle with all control rods withdrawn, RPS Drywell Pressure instrument C71-MYE-N650B, fails upscale. A short time later the fuse (C71-F18C) supplying power to the Scram Trip System A Group 3 Scram Pilot Valve Solenoids blows.
No operator actions have been taken.

In response to these events, __________ control rods will scram into the core.

______ A. 0

______ B. 22

______ C. 47

______ D. 185

Point Value: 1

Question 21.

The following conditions exist 10 minutes after a LOCA event:

- Drywell pressure: 4 psig and stable.
- Reactor Water Level (Wide Range): 25 inches and stable.

Select the response that contains the MINIMUM actions necessary to open the Drywell Spray Inboard Isolation Valve, E1150-F021A.

_____ A. Depress the OPEN pushbutton.

_____ B. Place Containment Spray Mode Select Switch to MANUAL and depress the OPEN pushbutton.

_____ C. Place Containment Spray Mode Select Switch to MANUAL, place 2/3 Core Height Override Switch to MANUAL OVERRIDE, and depress the OPEN pushbutton.

_____ D. Shut E1150-F016A, Drywell Spray Outboard Isolation Valve, place Containment Spray Mode Select Switch to MANUAL, place 2/3 Core Height Override Switch to MANUAL OVERRIDE, and depress the OPEN pushbutton.

Point Value: 1

Question 22.

The P603 operator has rotated the SLC injection control switch to the Pump A position. No other isolation signals have been present for the RWCU system. How does the RWCU system respond?

______ A. G3352-F001, RWCU SUPPLY INBD ISO VLV, Closes
G3352-F004, RWCU SUPPLY OTBD ISO VLV, Closes

______ B. G3352-F004, RWCU SUPPLY OTBD ISO VLV, Closes
G3352-F220, RWCU TO FW OTBD CTMT ISO VLV, Closes

______ C. G3352-F001, RWCU SUPPLY INBD ISO VLV, Closes
G3352-F119, RWCU SUPPLY SUCT ISO VLV, Closes

______ D. G3352-F220, RWCU TO FW OTBD CTMT ISO VLV, Closes
G3352-F119, RWCU SUPPLY SUCT ISO VLV, Closes

Point Value: 1

Question 23.

The plant is in cold shutdown with RHR Pump A operating in Shutdown Cooling when the operator observes that reactor coolant temperature has increased. The operator verifies that RHR Pump A is still running and that no annunciators have been received. Which ONE of the following valves, if inadvertently closed, would have caused the increase in temperature.

______ A. E11-F009, RHR SDC Inboard Isolation Valve

______ B. E11-F008, RHR SDC Outboard Isolation Valve

______ C. E11-F003A, RHR Heat Exchanger A Outlet Valve

______ D. E11-F048A, RHR Heat Exchanger A Bypass Valve

Point Value: 1

Question 24.

Placing the B Main Steam Line Radiation Monitor Mode Selector Switch in any position other than OPERATE will actuate a ________.

______ A. half scram and de-energize one half of the MSIV solenoids

______ B. full scram and de-energize one half of the MSIV solenoids

______ C. half scram and de-energize only the inboard AC MSIV solenoids

______ D. full scram and de-energize only the outboard DC MSIV solenoids

Point Value: 1

Question 25.

Which one of the following would prevent continued plant operation during a reactor startup if the mode switch were placed in RUN?

______ A. 1 IRM is failed downscale

______ B. 1 APRM is failed upscale

______ C. 2 IRMs are indicating upscale

______ D. 2 APRMs are indicating downscale

Point Value: 1

Question 26.

A plant transient has occurred. Plant conditions are:

Mode 3.
RPV pressure is 950 psig and stable.
RPV water level is 120 inches and stable.
Recirculation Pumps A & B have tripped.

Which RPV level instrument would provide the most accurate level reading at this time?

______ A. Floodup

______ B. Core Level

______ C. Wide Range

______ D. Narrow Range

Point Value: 1

Question 27.

Plant conditions are as follows:

- The plant is operating at 20% power.
- The turbine generator is paralleled to the grid.
- The Generator is providing an output of 200MW.
- The TURBINE FLOW LIMIT is set at 25%.
- The REACTOR FLOW LIMIT is set at 115%.
- The PRESSURE REGULATOR is set at 944 psig.
- The Turbine Bypass Valves are CLOSED.
- The SPEED/LOAD is set at 300MW.

How will the plant respond to an INCREASE to 45% power without any further operator action?

______ A. Generator output will remain constant, Bypass Valves will OPEN.

______ B. Generator output will remain constant, Bypass Valves will remain CLOSED.

______ C. Generator output will increase to approximately 250MW, Bypass Valves will OPEN.

______ D. Generator output will increase to approximately 250MW, Bypass Valves will remain CLOSED.

Point Value: 1

Question 28.

Which ONE of the following completes the statement concerning the main turbine generator mechanical overspeed trip testing on-load?

Oil is ported to 1 of 2 trip rings, causing it to operate, however, since it takes:

______ A. both trip rings to cause a trip, the turbine will not trip.

______ B. only 1 trip ring to cause a trip, the turbine will trip and must be restarted.

______ C. only 1 trip ring to cause a trip, the trip is electrically prevented during testing.

______ D. only 1 trip ring to cause a trip, the trip linkage is prevented from operating by a mechanical stop.

Point Value: 1

Question 29.

The plant is operating at 100% power with both feed pumps in service. Narrow range level instrument C32-N004A is failed high and feedwater level control is selected to the "B" level instrument. Narrow range instrument C32-N004C suddenly fails high. Which of the following describes the response of the RFPTs?

______ A. The RFPTs will immediately trip due to satisfying the Level 8 trip logic.

______ B. The RFPTs speed will increase, and the RFPs will steady out and control RPV level 11 inches above setpoint.

______ C. The RFPTs speed will decrease, and the RFPs will steady out and control RPV level 11 inches below setpoint.

______ D. The RFPs will continue to control level since the level instrument C32-N004B is selected for input to Feedwater Level Control.

Point Value: 1

Question 30.

The Post Scram Feedwater Logic circuit:

______ A. prevents inadvertent tripping of the RFPTs on high reactor water level.

______ B. prevents inadvertent tripping of the RFPTs due to low suction pressure.

______ C. ensures that a low level scram condition cannot be reset for a minimum of 10 seconds.

______ D. ensures that adequate subcooling is maintained to prevent recirculation pump cavitation.

Point Value: 1

Question 31.

A LOCA has occurred outside of the primary containment. Plant conditions are as follows:

- Reactor Building Temperature (near all instrument runs) = 220° F and stable.
- Reactor Pressure = 250 psig and stable.
- Drywell Temperature = 155° F and stable.

Assuming the indicated level on each of the below instruments is 163 inches, which level instrument may be used for trending indication?

______ A. B21-N027

______ B. B21-N080A

______ C. B21-N091B

______ D. B21-N095C

Point Value: 1

Question 32.

With the plant at 100% power and no testing in progress, what is the maximum Suppression Chamber water temperature allowed without entry into a Limiting Condition for Operation?

_____ A. 95° F

_____ B. 105° F

_____ C. 110° F

_____ D. 120° F

Point Value: 1

Question 33.

Plant conditions are as follows:

- 3 turbine stop valves have drifted to 80% open.
- 50% of the control rods fully inserted.
- The remaining are scattered throughout the core at positions 02 to 48.
- The individual blue lights for each control rod on the full-core display are illuminated.
- The eight scram solenoid group indicating lights are extinguished.

Based on these indications, which of the following is the cause for the failure of the rods to insert?

_____ A. Failure of scram inlet and outlet valves to open.

_____ B. Failure of the back-up scram valves to de-energize.

_____ C. Failure of scram pilot valve solenoids to de-energize.

_____ D. Failure of RPS to detect water in the scram discharge volume.

Point Value: 1

Question 34.

The Rod Block Monitor setpoint circuitry:

______ A. provides alarms when power level approaches to within 2% of the trip setpoint.

______ B. generates three flow biased trip reference signals using a flow signal from the APRM.

______ C. allows the operator to manually select the proper power level setpoint during operation.

______ D. selects the appropriate setpoint when a rod is selected based on current reactor power.

Point Value: 1

Question 35.

The reactor is operating in mode 2 when an IRM mode switch is placed in STANDBY. In addition to a half scram occurring, the IRM will read:

______ A. downscale on the meter and indicate an Inop trip

______ B. downscale on the meter and indicate a Downscale trip

______ C. current power level on the meter and indicate an Inop trip

______ D. current power level on the meter and indicate a Downscale trip

Point Value: 1

Question 36.

The following plant conditions exist:

- Drywell pressure is 3 psig and stable.
- Reactor water level is 180 inches and stable.

Which ONE of the following lists the only valves that require the CONTAINMENT SPRAY MODE SELECT switch to be in the MANUAL position in order to open the valves?

______ A. E11-F024A, RHR Loop A Suppression Pool Cooling/Test Isolation Valve
E11-F026B, RHR Warmup Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ B. E11-F016A, Drywell Spray Outboard Isolation Valve
E11-F024B, RHR Loop B Suppression Pool Cooling/Test Isolation Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ C. E11-F015A, LPCI Loop A Inboard Isolation Valve
E11-F016A, Drywell Spray Outboard Isolation Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ D. E11-F016B, Drywell Spray Outboard Isolation Valve
E11-F027A, RHR Loop A Suppression Chamber Spray Inboard Isolation Valve
E11-F048A, RHR Heat Exchanger A Bypass Valve

Point Value: 1

Question 37.

Following a loss of power to Bus 64B, the bus was subsequently restored to the normal lineup EXCEPT the operators neglected to reset the digital load sequencer. Following the restoration, all power is again lost to Bus 64B. How will the EDG and electrical distribution system respond to this event?

______ A. The EDG will require a manual start. The loads on bus 64B will sequence after the output breaker is closed.

______ B. The EDG will require a manual start. The loads on bus 64B will NOT sequence after the output breaker is closed.

______ C. The EDG will automatically start. The loads on bus 64B will sequence after the output breaker is closed.

______ D. The EDG will automatically start. The loads on bus 64B will NOT sequence after the output breaker is closed.

Point Value: 1

Question 38.

Procedure 23.206, Reactor Core Isolation Cooling System, contains a precaution concerning extended operation of the RCIC turbine below 2100 RPM. The purpose of this precaution is to prevent:

______ A. bearing damage.

______ B. oil cooler damage.

______ C. the stop valve from closing spuriously.

______ D. turbine damage due to inadequate steam cooling.

Point Value: 1

Question 39.

Which ONE of the following will trip the 345kV Switchyard source breaker to System Service Transformer 65:

______ A. Low bus voltage

______ B. Low transformer oil level

______ C. Primary phase overcurrent

______ D. Transformer high temperature

Point Value: 1

Question 40.

The plant is being controlled using the Remote Shutdown System when a loss of Division 2 DC Power occurs. Which ONE of the following describes how this loss effects operations from the Remote Shutdown Panel.

______ A. RCIC flow indication and drywell pressure indication is lost.

______ B. The ability to operate Mechanical Draft Cooling Tower Fans in fast speed is lost.

______ C. The ability to operate RHR SDC Suction Inboard Isolation Valve, E1150-F009, is lost.

______ D. The ability to operate RHR SDC Suction Outboard Isolation Valve, E1150-F008, is lost.

Point Value: 1

Question 41.

Main condenser vacuum is 1 psia and a loss of vacuum is in progress. What is the MINIMUM condenser pressure when the main condenser will be COMPLETELY lost as a heat sink for any decay heat removal?

_____ A. 2.7 psia

_____ B. 6.8 psia

_____ C. 6.85 psia

_____ D. 12.2 psia

Point Value: 1

Question 42.

Due to fire in the control room the plant is being controlled from the Dedicated Shutdown Panel. Plant conditions are as follows:

- Reactor pressure is being controlled using SRVs.
- Reactor water level is 218 inches and increasing.

Which ONE of the following describes the reason for the increasing water level?

_____ A. The SBFW System does not have a level 8 isolation.

_____ B. The RPV Level Instrument is malfunctioning otherwise SBFW Pump would have tripped.

_____ C. The level 8 isolation of SBFW is not operable when operating from the Dedicated Shutdown Panel.

_____ D. When operating from the Dedicated Shutdown Panel, the setpoint for the RPV Level 8 isolation is raised 10 inches to provide a wider level control band.

Point Value: 1

Question 43.

Procedures 22.000.02, Plant Startup to 25% Power, and 23.137, Nuclear Boiler System, have cautions to prevent opening MSIVs with excessive differential pressure across their seats. One reason for this precaution is due to the potential damage to MSIVs and/or other Main Steam Line components. Another reason for this caution is to prevent:

______ A. Over speeding the turbine.

______ B. Loss of condenser vacuum.

______ C. Transients in indicated RPV level.

______ D. Damage to the MSL flow restrictors.

Point Value: 1

Question 44.

The Control Rod Drive Hydraulic system is operating normally when C11-F412, CRD Pump Suction Pressure Control Valve, fails closed. How does the system respond?

______ A. The running CRD pump trips on low suction pressure.

______ B. The Demin Water Pumps auto start to supply CRD Pump.

______ C. A check valve opens to supply the CRD system from the Torus.

______ D. A check valve opens to supply the CRD system from the Condensate Storage Tank.

Point Value: 1

Question 45.

The basis for limiting the Reactor Pressure Vessel (RPV) heatup and cooldown rates to 100° F/hr is to satisfy the stress limits for:

______ A. RPV cyclic operation.

______ B. RPV steady state operation.

______ C. feedwater nozzles.

______ D. jet pump expansion/contraction

Point Value: 1

Question 46.

There has been a transient involving the failure of fuel and the breach of a nuclear system process barrier. One hour and 30 minutes after the accident an off-site survey determines that the dose rate at the site boundary is 20 Rem per hour whole body and 180 Rem per hour thyroid and steady. Given this data, what can be determined about the Primary Containment Isolation System (PCIS)?

______ A. The system is meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR20.

______ B. The system is NOT meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR20.

______ C. The system is meeting its design bases of maintaining the dose at the site boundary within the limits of 10CFR100.

______ D. The system is NOT meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR100.

Point Value: 1

Question 47.

Which ONE of the following completes the following statement.

No more than ____(1)____ fuel bundles shall be allowed in or around the fuel prep machines. The fuel shall be separated from the main body of other fuel by at least _____(2)_____ inches.

		(1)	(2)
_____	A.	2	6
_____	B.	2	12
_____	C.	3	6
_____	D.	3	12

Point Value: 1

Question 48.

The main turbine trips at 45% power. RPS B scram relays fail to reposition when deenergized causing a failure to scram.

ASSUMING NO OPERATOR ACTIONS, Which ONE of the following statements describes the expected response to this situation?

______ A. The backup scram valves associated with RPS A will energize when the RPS A scram relays trip.

______ B. The backup scram valves associated with RPS A will deenergize when the RPS A scram relays trip.

______ C. The alternate rod insertion valves deenergize when the high reactor pressure setpoint is exceeded on one trip unit in each trip channel.

______ D. The alternate rod insertion valves energize when the high reactor pressure setpoint is exceeded on both trip units in one trip channel.

Point Value: 1

Question 49.

Per Abnormal Operating Procedure (AOP) 20.000.19, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM, which ONE of the following is the preferred method of scramming the reactor if you are UNABLE to do so prior to leaving the control room?

______ A. Depress the Main Turbine Trip buttons on H11-P632, Turbine Protection Cabinet.

______ B. Open RPS Motor-Generator Output Breakers on RPS MG Set A and B Control Panels AB3-H11.

______ C. Insert a trip signal for RPS A and B channels on H11-P609/611, Relay Room RPS Logic Cabinets.

______ D. Scram Reactor by taking two operable APRM Mode Switches out of OPER at H11-P608, Power Range Neutron Monitoring Cabinet.

Point Value: 1

Question 50.

Which ONE of the following statements describe the relationship between the Boron Injection Initiation Temperature (BIIT) curve and the Heat Capacity Limit (HCL) curve?

______ A. The HCL curve, based on reactor pressure, determines how long the operator has until the limits of the BIIT curve are reached requiring boron injection.

______ B. Operating by the BIIT curve ensures that enough boron is in the core to shutdown the reactor before the suppression pool reaches the limits of the HCL curve.

______ C. Operating by either the BIIT curve or the HCL curve, whichever is most restrictive, ensures the limits of the Primary Containment Pressure Limit curve will not be reached.

______ D. The BIIT curve and the HCL curve combined, monitor the energy additions to the suppression pool to prevent exceeding the limits of the Pressure Suppression Pressure curve.

Point Value: 1

Question 51.

The Dedicated Shutdown System transfer switches are designed to:

______ A. Bypass all interlocks and breaker protective functions.

______ B. Transfer equipment control power to a source powered directly from the control room or relay room.

______ C. Transfer indicating instrument power to sources powered directly from the control room or relay room.

______ D. Provide electrical separation between the Main Control Room and the Local Panel controls.

Point Value: 1

Question 52.

The following plant conditions exist following a LOCA event:

- Torus Water Temperature is 180° F and stable.
- Torus Water level is -70 inches and stable.
- Drywell Pressure is 0 psig and stable.
- Reactor Pressure 0 psig and stable.
- Core Spray A is injecting into the Division 1 loop.
- Core Spray B is injecting into the Division 2 loop.

Assuming no change in the above conditions, what is the maximum flow allowed for the Division 1 Core Spray Loop?

______ A. 3500 gpm

______ B. 3800 gpm

______ C. 7000 gpm

______ D. 7600 gpm

Point Value: 1

Question 53.

A plant transient has occurred.
Reactor pressure is 350 psig and steady.
Which ONE of the following pumps are capable of injecting at this time?

______ A. Core Spray Pumps

______ B. Condenser Pumps

______ C. Heater Feed Pumps

______ D. RHR Service Water Pumps

Point Value: 1

Question 54.

A reactor scram has occurred and all control rods are NOT inserted.

Which one of the following methods for inserting control rods REQUIRES the scram to be reset?

______ A. Vent CRD Overpiston Volumes

______ B. Open the Scram Test Switches

______ C. Manual Control Rod Insertion

______ D. Increase the CRD Cooling Water Differential Pressure

Point Value: 1

Question 55.

If hydrogen concentration can not be determined to be <6% AND oxygen concentration can not be determined to be <5%, the drywell fans must be shutdown.

Which ONE of the following describes the reason for this requirement?

_____ A. Allow drywell sprays to be initiated.

_____ B. Eliminate a potential ignition source.

_____ C. Prevent the hydrogen and oxygen from combining into a flammable mixture.

_____ D. Allow the hydrogen to accumulate in the top of the drywell where the oxygen concentration is the least.

Point Value: 1

Question 56.

Which ONE of the following is the reason for initiating drywell sprays before drywell temperature exceeds 340° F per 29.100.01, Sheet 2, Primary Containment Control?

_____ A. To prevent exceeding the design temperature of the drywell structure.

_____ B. To minimize damage to non-environmentally qualified drywell components.

_____ C. To minimize the damage to the drywell coolers and the reactor recirculation pumps.

_____ D. To prevent exceeding the design capacity of the torus-to-drywell vacuum breakers upon spray initiation.

Point Value: 1

Question 57.

The plant has experienced a small break LOCA. The following plant conditions exist:

- Reactor Water level is 175 inches and stable.
- Drywell Temperature is 187°F and stable.
- Drywell Pressure is 1.9 psig and stable.
- Suppression Pool water level is -1 inch and stable.
- Suppression Pool temperature is 94°F and stable.
- No operator actions have been taken

Given these conditions, Which ONE of the following describes the status of the EECW/RBCCW System.?

______ A. RBCCW will be supplying all loads. No components are being supplied by EECW.

______ B. EECW will be supplying Control Air Compressor Space Coolers, ESF Battery Rooms AC unit, Core Spray Pumps Seal Coolers. RBCCW will be isolated from essential components.

______ C. ECCW will be supplying Control Air Compressor Space Coolers, CCHVAC Condenser, Core Spray Space Coolers, and Thermal Recombiner Space Coolers. RBCCW will be isolated from essential components.

______ D. EECW will be supplying the Standby Gas Treatment System Space Coolers, EECW Space Cooler, Drywell Penetration Coolers, and the Post Accident Sampling System. RBCCW will be isolated from essential components.

Point Value: 1

Question 58.

Due to a large leak in the system, the Station Air Compressors are unable to maintain Station Air Header pressure above 72 psig.
Which ONE of the following indicates the current status of the air system.

______ A. P50-F440/441 Control Air Isolation valves have OPENED, Station Air compressors have TRIPPED.

______ B. P50-F440/441 Control Air Isolation valves have CLOSED, P50-F403 Div 2 NIAS Crosstie to IAS valve has OPENED

______ C. Div 1 and Div 2 Control Air Compressors are RUNNING, P50-F440/441 Control Air Isolation valves have CLOSED.

______ D. P50-F473 Station Air to IAS Isolation Valve has CLOSED, Div 1 and Div 2 Control Air Compressors are NOT RUNNING.

Point Value: 1

Question 59.

The plant is in extended maintenance shutdown in Mode 4 with the following conditions:
- A loss of shutdown cooling occurs.
- No RHR pumps can be recovered and placed into shutdown cooling.
- The North Reactor Recirc pump is running.

Which ONE of the following is an allowable option for alternate shutdown cooling under these conditions in accordance with 20.205.01, Loss of Shutdown Cooling?

______ A. RWCU Blowdown to the Main Condenser, makeup with CRD pumps.

______ B. Maximizing FPCCU flow and RBCCW Flow to the FPCCU Heat Exchangers.

______ C. Bleed Steam via SRVs when pressure reaches 100 psig. Makeup with Core Spray.

______ D. Bleed Steam Via Bypass Valves when pressure reaches 50 psig. Makeup with SBFW.

Point Value: 1

Question 60.

Which HPCI valve DIRECTLY receives a high suppression pool water level signal to reposition?

______ A. HPCI Minimum Flow Valve (E41-F012)

______ B. CST to HPCI Suction Isolation Valve (E41-F004)

______ C. HPCI test return to CST Isolation Valve (E41-F011)

______ D. Suppresion Pool to HPCI Suction Isolation Valve (E41-F042)

Point Value: 1

Question 61.

The Reactor Building Ventilation Exhaust Radiation readings have been showing a steady increase. Which of the following is the MINIMUM Reactor Building Ventilation Exhaust Radiation Level above which the SGTS will automatically start?

_____ A. 10,000 CPM

_____ B. 12,000 CPM

_____ C. 16,000 CPM

_____ D. 20,000 CPM

Point Value: 1

Question 62.

Which signal will cause a DIRECT trip of the Torus Water Management Pumps?

_____ A. High drywell pressure

_____ B. RPV water level (Level 2)

_____ C. Hi-Hi DW Equipment Drain Sump Level

_____ D. Hi-Hi Torus Room Floor Drain Sump Level

Point Value: 1

Question 63.

You are the CRNSO in the control room.
Which of the following is an Immediate Operator Action for a confirmed fire in accordance with 20.000.22, Plant Fires?

______ A. Identify the type (class) of fire.

______ B. Announce the fire alarm over the Hi-Com system.

______ C. Send an operator to verify the magnitude and location of the fire.

______ D. Establish communications between Control Room and Fire Brigade.

Point Value: 1

Question 64.

Which of the following configurations will produce a RFPT speed change in response to a change in total steam flow?

______ A. Master Level Controller in Auto, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in three element

______ B. Master Level Controller in Manual, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in three element

______ C. Master Level Controller in Auto, S RFPT Flow Control M/A Station in Manual, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in single element

______ D. Master Level Controller in Manual, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Auto, Level Control Mode Select switch in single element

Point Value: 1

Question 65.

The recirculation pump speed is limited if feedwater flow is less than 20%. The purpose of this recirculation pump speed limiter is to ensure:

______ A. reactor power will not exceed the feedwater pump capability.

______ B. the recirculation pumps have adequate net positive suction head.

______ C. the feedwater heaters are on-line prior to higher power operations.

______ D. reactor feed pump turbines are operating on low pressure steam supply prior to increasing reactor power.

Point Value: 1

Question 66.

A fault in the AC power supply has caused the Main Generator Automatic Voltage Regulator (AVR) master channel to fail during normal power operations. How is main generator operation affected?

______ A. The Main Turbine Generator trips.

______ B. The Main Generator Field Breaker trips open.

______ C. The AVR master channel shifts to MANUAL control.

______ D. The AVR master channel shifts to the AVR slave channel.

Point Value: 1

Question 67.

The plant is shutown in mode 5.
A fuel bundle has been dropped in the spent fuel pool.
Which ONE of the following would be the cause of increase in offsite radiation release rates?

______ A. Both Rail Airlock door seals are INOPERABLE.

______ B. Pressure in the Secondary Containment is -0.132 inches wc.

______ C. T41-F011, Reactor Building Inboard Supply Damper, is INOPERABLE and closed.

______ D. CCHVAC is operating in recirc mode.

Point Value: 1

Question 68.

When in Recirculation mode, how is the positive differential pressure in the Control Center Envelope maintained?

______ A. Modulating vanes on the CCHVAC Exhaust fans.

______ B. Modulating the Emergency Makeup and Recirculation dampers.

______ C. Modulating the Emergency Exhaust and Inlet Pressure control dampers.

______ D. Modulating the Emergency Makeup and Inlet Pressure control dampers.

Point Value: 1

Question 69.

The plant is in a refueling outage with the mode switch in REFUEL and all rods inserted. The refuel crew has used the grapple to pick up a fuel bundle. They start to move towards the core when the control room operator withdraws a rod. When the bridge reaches the core area the bridge will:

______ A. stop and a hoist block will be generated

______ B. stop and the hoist will remain operable

______ C. continue moving and a hoist block will be generated

______ D. continue moving and the hoist will remain operable

Point Value: 1

Question 70.

What is the technical specification limit for water level above the top of the RPV flange when the plant is in Mode 5 and fuel movement is in progress?

______ A. 19.0 feet

______ B. 19.5 feet

______ C. 20 feet

______ D. 20.5 feet

Point Value: 1

Question 71.

Select the conditions that are necessary for the plant to be in COLD SHUTDOWN mode.

______ A. Mode switch in Shutdown, Coolant temperature less than 200° F and all head closure bolts fully tensioned.

______ B. Mode switch in Shutdown or Refuel, Coolant temperature less than 212° F and all head closure bolts fully tensioned.

______ C. Mode switch in Shutdown or Refuel, Coolant temperture less than 200° F and all head closures bolts fully tensioned.

______ D. Mode switch in Shutdown, Coolant temperature less than 212° F and no more than one head closure bolt detensioned.

Point Value: 1

Question 72.

I & C technicians are preparing to perform a surveillance procedure on an instrument. They observe that a prerequisite to the procedure cannot be met under present plant conditions. The prerequisite will not affect the performance of the procedure and the surveillance will be past it's critical date the next day. Which ONE of the following actions must be taken in order to continue with the surviellance?

______ A. The NASS must waive the prerequisite.

______ B. The I&C supervisor must waive the prerequisite.

______ C. The I&C tech must mark the step as N/A (Not Applicable).

______ D. The I&C supervisor must mark the step as N/A (Not Applicable).

Point Value: 1

Question 73.

The plant is at 40% power conducting the quarterly surveillance test on the turbine bypass valves when the following occurs:
- A turbine bypass valve fails to 100% open and cannot be closed.
- 3D91 Turbine Stop/Cont Val Channel Trip Bypassed alarms.

Which ONE of the following contains a complete list of ALL the equipment that must be considered INOPERABLE in response to this event.

______ A. Turbine bypass valves

______ B. RPS Turbine control valve trip function

______ C. Turbine bypass valves and RPS turbine control valve trip function

______ D. Turbine bypass valves, RPS turbine control valve and turbine stop valve trip functions

Point Value: 1

Question 74.

You have donned PCs to hang a tag in a contaminated area where mechanics are disassembling a RWCU valve. The area is also posted as a High Radiation Area with a survey reading of 250 mr/hr in the work area. While attempting to reach the valve requiring a tag, you catch your rubber overshoe on an obstruction and it tears and comes off. You should:

______ A. leave the area and check yourself for contamination.

______ B. stand fast and request one of the mechanics to call Radiation Protection.

______ C. continue hanging the tag to reduce the additional exposure resulting from exiting and re-entry into the area, but attempt to prevent tearing the shoe covers.

______ D. put the overshoe back on attempting to prevent damage to the shoe cover. When leaving the area perform a thorough frisk and inform Radiation Protection of the occurrence.

Point Value: 1

Question 75.

In accordance with 23.406, Standby Gas Treatment System procedure; when performing a containment purge, the Reactor Building HVAC system is the preferred purge path UNLESS:

______ A. SGTS system has both trains operable.

______ B. Effluent release monitoring is required.

______ C. The containment is NOT completely inerted.

______ D. SGTS is needed to comply with the Effluent Release Limit.

Point Value: 1

Question 76.

Which ONE of the following set of conditions MUST be present to satisfy RBCCW pump starting interlocks?

______ A. RBCCW Make-Up Tank Isolation Valve OPEN and Control Switch in MANUAL.

______ B. RBCCW Pump Suction Pressure 10 psig and RBCCW Make-Up Tank Isolation Valve OPEN.

______ C. RBCCW Make-Up Tank Level 5 inches from the bottom and suction and discharge valves to the RBCCW Pump OPEN.

______ D. RBCCW Pump Suction Pressure 8 psig and RBCCW Make-Up Tank Level 14 inches from the bottom.

Point Value: 1

Question 77.

The plant was performing a power ascension when the 'B' Recirculation Pump tripped. Plant conditions are as follows:

- Reactor Power is 48%
- 'A' loop flow indicates 20,500 gpm

In what region of the Power/Flow map is the reactor operating?

______ A. Scram region

______ B. Exit region

______ C. Stability Awareness region

______ D. Normal operating region

Point Value: 1

Question 78.

A valid secondary containment isolation has occurred. SELECT the item that contains ONLY those actions that will occur as a result of this isolation.

______ A. SGTS for both divisions start
Control Center HVAC shifts to Recirculation Mode
T4100-F600 and T4100-F601, RB Main Supply Header Isolation Valves close

______ B. SGTS for both divisions start
Control Center HVAC shifts to Recirculation Mode
T4100-F010 and T4100-F011, RBHVAC Supply Outboard and Inboard Isolation dampers close

______ C. SGTS for both divisions start
T4100-F600 and T4100-F601, RB Main Supply Header Isolation Valves close
T4100-F010 and T4100-F011, RBHVAC Supply Outboard and Inboard Isolation dampers close

______ D. Control Center HVAC shifts to Recirculation Mode
T4100-F600 and T4100-F601, RB Main Supply Header Isolation Valves close
T4100-F010 and T4100-F011, RBHVAC Supply Outboard and Inboard Isolation dampers close

Point Value: 1

Question 79.

Which ONE of the following describes the arrangement of the valves in the Reactor Building-to-Suppression Chamber Vacuum Relief System?

_____ A. A single line with two self-actuated check valves in series.

_____ B. A single line with a self-actuated check valve and an air operated butterfly valve in series.

_____ C. Two lines, each with a self-actuated check valve and an air operated butterfly valve in series.

_____ D. Two lines, each with a self-actuated check valve and an air operated butterfly valve in parallel.

Point Value: 1

Question 80.

Carbon Dioxide (CO_2) has been released into the Standby Gas Treatment system Charcoal Adsorber due to an automatic initiation on high temperature. At the end of a 10-minute period, the admission valve is automatically closed. At the time the admission valves closes, the high temperature condition still exists. Select the item that describes how CO_2 would be re-admitted to the charcoal adsorber.

_____ A. the CO_2 admission valve will IMMEDIATELY re-open and re-admit CO_2 to the charcoal.

_____ B. the manual release station pushbutton MUST be depressed.

_____ C. the manual actuation lever MUST be used to actuate the system.

_____ D. if the high temperature exists ONLY after an additional 5-minute period, the CO_2 admission valve will re-open.

Point Value: 1

Question 81.

The CRD system is in operation with the reactor operating at 50% power. The CRD flow controller is in automatic. The operator depresses the Drive Water Pressure Control Throttle Valve (C11-F003) open pushbutton for two seconds.

Which ONE of the following describes how the parameters will stabilize when the transient is over?

______ A. Drive water pressure will increase. Cooling water flow will remain the same.

______ B. Drive water pressure will increase. Cooling water flow will decrease.

______ C. Drive water pressure will decrease. Cooling water flow will remain the same.

______ D. Drive water pressure will decrease. Cooling water flow will decrease.

Point Value: 1

Question 82.

During insertion of a control rod the Reactor Operator notices that the Settle Light did not illuminate when rod motion was stopped.

Which of the following caused this indication?

______ A. The operator selected another rod during rod movement.

______ B. Control Rod Drive Hydraulic Pressure decreased during rod movement.

______ C. The operator used the Emergency Rod In position of the RONOR switch.

______ D. The Settle Function is automatically bypassed whenever inserting control rods.

Point Value: 1

Question 83.

A LPCI Loop Selection initiation signal has occurred. Operation of the logic has progressed to the actual point of loop selection (2 second time delay has expired).

Select the item below which correctly completes the following statement:

The logic determines if ___(1)___ recirc riser pressure is ___(2)___ than ___(3)___ recirc riser pressure by 0.626 psid. If so, then loop ___(4)___ is selected for injection.

	(1)	(2)	(3)	(4)
_____ A.	A	greater	B	A
_____ B.	B	greater	A	B
_____ C.	A	less	B	A
_____ D.	B	less	A	B

Point Value: 1

Question 84.

Complete the following statement with respect to the Rod Worth Minimizer (RWM).

When power is __________ the Low Power Setpoint, the RWM provides rod block signals to the RMCS to __________.

______ A. above, prevent a rod drop accident from occurring.

______ B. below, prevent a rod drop accident from occurring.

______ C. above, minimize the effects of a rod drop accident.

______ D. below, minimize the effects of a rod drop accident.

Point Value: 1

Question 85.

EDG 14 is loaded to 1800 KW for monthly testing. Off-site power is subsequently lost. Which ONE of the following describes how EDG 14 will respond?

______ A. The EDG output breaker will open, and EDG 14 will shut down.

______ B. The EDG output breaker will open, Load Shed will occur, and the EDG output breaker will re-close.

______ C. The EDG output breaker will remain closed, and the EDG will shut down then restart in isochronous mode.

______ D. The EDG breaker will remain closed, and the EDG will remain running. The governor will shift to isochronous mode.

Point Value: 1

Question 86.

The SLC Injection Sparger is connected to a Differential Pressure switch that is used for line break detection. Which one of the following components is this switch designed to detect a line break in?

_____ A. LPCI Injection Line

_____ B. RCIC Injection Line

_____ C. Core Spray System Injection Line

_____ D. High Pressure Coolant Injection System Injection Line

Point Value: 1

Question 87.

The plant is operating at 100% power when annunciator 3D164, FEEDWATER CONTROL SIGNAL FAILURE, is received. The South RFPT Control Signal Failure white light on H11-P603 is also illuminated. Which ONE of the following conditions describes how the Feedwater systems will initially respond to these plant conditions?

_____ A. Both RFPT's speed will decrease.

_____ B. Both RFPT's speed will increase.

_____ C. North RFPT speed will increase, South RFPT speed will decrease.

_____ D. North RFPT speed will decrease, South RFPT speed will increase.

Point Value: 1

Question 88.

Which of the following is the reason for initiating drywell sprays within the safe region of the Drywell Spray Initiation Limit Curve?

To preclude:

______ A. Torus-to-Drywell pressure differential from directly causing failure of the primary containment.

______ B. Reactor Building-to-Torus differential pressure from directly causing failure of the primary containment.

______ C. the occurence of an uncontrolled rise in drywell pressure due to flashing to steam of the drywell water.

______ D. the occurence of a saturated drywell atmosphere which could lead to containment failure due to a convective cooling pressure drop.

Point Value: 1

Question 89.

29.100.01, Sheet 5, 'Secondary Containment and RAD Release', requires a rapid depressurization of the RPV if, while a primary system is discharging into the secondary containment, Maximum Safe Operating Temperature is exceeded for more than one area.

The criteria of "more than one area" is based on:

______ A. rejecting the energy from the RPV to the suppression pool.

______ B. the potential for fire damage to equipment, instrumentation and controls.

______ C. the potential for damage to equipment and loss of secondary containment integrity.

______ D. protecting personnel from high temperature environments while operating equipment.

Point Value: 1

Question 90.

During a reactor startup with reactor pressure at 300 psig, the excess flow check valve for a narrow range reactor level indicator reference leg inadvertently shuts and remains undetected. Which ONE of the following will be the response of INDICATED level as reactor pressure increases to 350 psig?

______ A. Level indication will fail low as the pressure equalizes on both legs.

______ B. Level indication will fail high as the pressure equalizes on both legs.

______ C. Indicated level will increase as the pressure increases on the variable leg.

______ D. Indicated level will decrease as the pressure increases on the variable leg.

Point Value: 1

Question 91.

SELECT the answer which correctly completes the following:

The Rod Position Indication system provides the process computer with Control Rod ____________ and ____________ information.

______ A. speed, alarm

______ B. position, alarm

______ C. speed, temperature

______ D. position, identification

Point Value: 1

Question 92.

SELECT the item that describes what happens to the Off Gas system if the Off Gas Ring Water Pumps are lost?

_____ A. Cooling flow to the Off Gas Condenser will be lost and Off Gas will have an increase in Off Gas flow.

_____ B. Negative pressure in the Off Gas System is lost and radioactive gases leak out of Off Gas and into the Turbine Building.

_____ C. Nothing would occur in the Off Gas System since the Ring Water Pumps are only installed as a backup to the SJAEs.

_____ D. Flow from the Main Condenser to the Off Gas System is lost resulting in the loss of Main Condenser Vacuum.

Point Value: 1

Question 93.

20 minutes after a LOCA the following conditions exist.

•Reactor Water Level	17 inches
•Reactor Pressure	46 psig
•Main Steam Line Pressure	38 psig
•Control Air Pressure	85 psig

Which of the following conditions must be met for the first component of Div 1 MSIV leakage Control System (MSIVLCS) to actuate?

Initiate keylock switch placed in INITIATE and_____________.

_____ A. reactor pressure less than 44 psig

_____ B. control air pressure greater than 95 psig

_____ C. 3rd MSIV and outboard MSIV d/p less than 138.4 psid

_____ D. inboard and outboard MSIV d/p less than 30 psid

Point Value: 1

Question 94.

A startup is in progress with the mode switch in STARTUP.

- The crew is conducting RPV pressurization in accordance with procedure 22.000.02, PLANT STARTUP TO 25% POWER.
- The NASS directed the Main Turbine Bypass Valves Low Vacuum trip to be reset at 5.7 psia
- Reactor Pressure is 375 psig and stable.
- The reactor Pressure Regulator setpoint is 275 psig.

Which ONE of the following describes the plant response to these conditions?

______ A. The RPV will rapidly depressurize.

______ B. The plant will scram on High RPV Pressure.

______ C. The condenser will be damaged due to water impingement on the tubes.

______ D. The plant will scram on APRM Neutron Flux-Upscale (Setdown) or IRM High Flux.

Point Value: 1

Question 95.

The plant is in mode 5 with refueling activities in progress. You have been notified that a new fuel bundle is being placed in the core. Shortly thereafter you observe the SRM counts increasing with a steady positive period. In accordance with procedure MOP13, Refueling Operations, you IMMEDIATLEY:

______ A. place the mode switch in shutdown.

______ B. check to see if the SRM is functioning properly.

______ C. direct the refuel floor to stop fuel movement immediately.

______ D. inform the refuel floor to remove the fuel bundle and try again.

Point Value: 1

Question 96.

An operator is conducting a normal day to day rounds inspection of equipment which is located in a high radiation area. In accordance with MOP04, Shift Operations, and MRP05 , ALARA/RWPs, the operator must:

_____ A. preplan the inpection during turnover.

_____ B. conduct the inspection from the door.

_____ C. obtain Radiation Protection supervisor approval.

_____ D. enter the area with a hand held monitoring device.

Point Value: 1

Question 97.

The Reactor Pressure Controller Pressure Regulator Signal has failed high. Which ONE of the following immediate operator actions is required in accordance with 20.109.02, Reactor Pressure Controller Failure?

_____ A. Trip the Main Turbine.

_____ B. Place the Reactor Mode Switch in SHUTDOWN.

_____ C. Arm and trip the MS East & West Bypass Valves

_____ D. Verify the Backup Pressure Regulator takes control.

Point Value: 1

Question 98.

Plant conditions are as follows:

- Reactor Water level is 154 inches and stable.
- Drywell Pressure is 1.88 psig and stable.
- Torus Water Level is +1 inch and stable.
- Drywell Temperature is 140° F and stable.
- MSIVs are closed.
- Containment Hydrogen is 0.5% and stable.

From the list below, SELECT the EOPs that are required to be in use. (Assume no other information other than the information stated)

1. RPV Control
2. Containment Control
3. Primary Containment H2/O2 Control
4. Secondary containment and Rad Release

CHOOSE from the following selections:

______ A. 1 and 2

______ B. 1 and 4

______ C. 2 and 3

______ D. 3 and 4

Point Value: 1

Question 99.

Operator Caution 2 of the EOPs provides guidance concerning operating HPCI and RCIC with torus temperature above 140° F. What is the reason for this caution?

______ A. At this temperature HPCI and RCIC may have inadequate NPSH which would lead to pump damage.

______ B. Water at this temperature may not provide adequate cooling for RCIC and HPCI lubricating oil.

______ C. Water at this temperature may not provide adequate cooling for the barometric condensers and therefore the turbines will not have adequate gland sealing.

______ D. Above this temperature, the exhaust from the RCIC and HPCI turbines may not be sufficiently condensed hence adding steam directly to the containment.

Point Value: 1

Question 100.

The Fermi 2 Fire Brigade is responsible for responding to fires inside the protected area, general service water pump house, circ water pump house and Fermi 1. Nuclear Security is responsible for responding to all other fires. When Nuclear Security responds to a fire, a Nuclear Supervising Operator (NSO) is dispatched also. The duties of this NSO include:

______ A. Coordinating the activities of all involved groups.

______ B. Providing respirators and miscellaneous fire fighting equipment.

______ C. Acting as fire brigade leader for the Nuclear Security fire responders.

______ D. Initially contacting the off-site (Frenchtown) fire department.

Point Value: 1

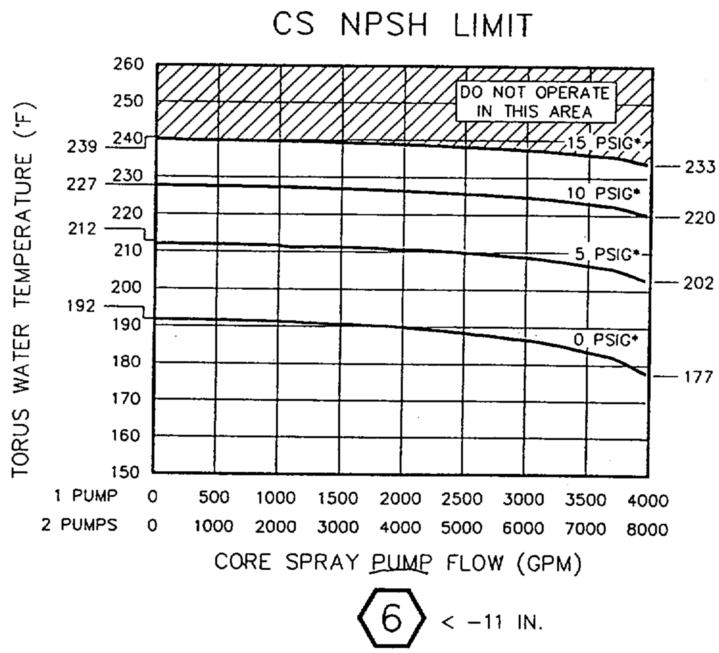
Emergency Action Level Definitions

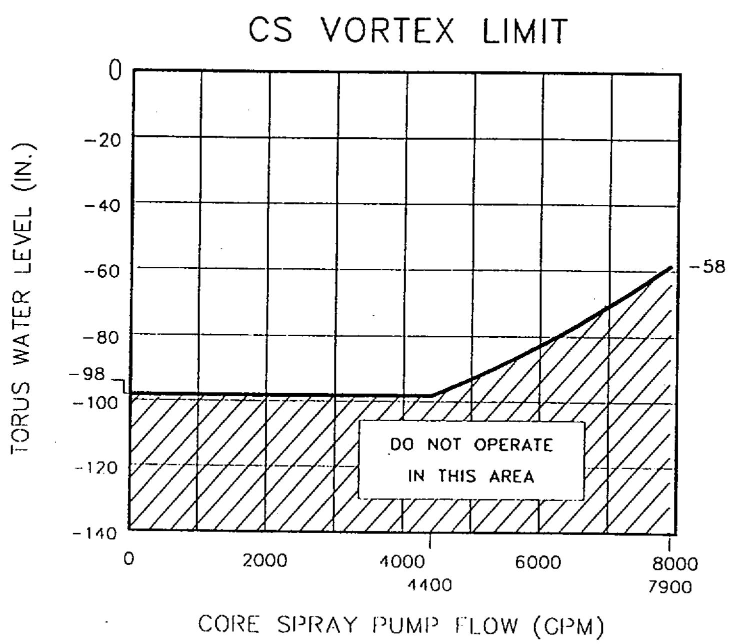
Unusual Event - Events are in process or have occurred that indicate a potential degradation of the level of safety of the plant. No release of radioactive material requiring offsite response or monitoring is expected.

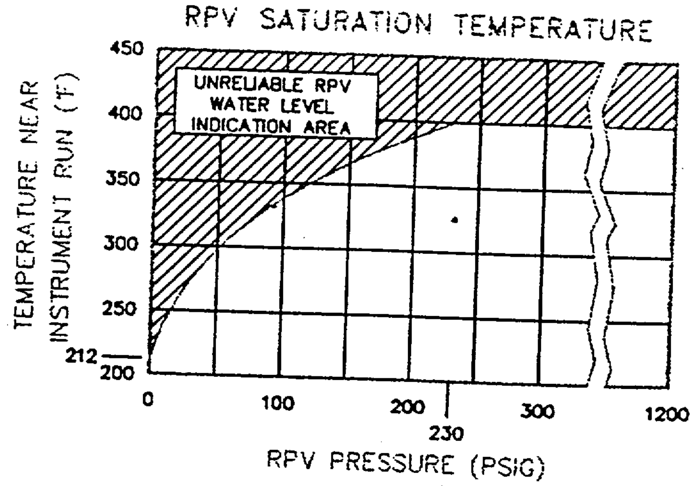
Alert - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels.

Site Area Emergency - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels except at or near the site boundary.

General Emergency - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels offsite for more than the immediate site area.


CS NPSH LIMIT
DO NOT OPERATE IN THIS AREA
15 PSIG*
10 PSIG*
5 PSIG*
0 PSIG*
239
227
212
192
233
220
202
177
260
250
240
230
220
210
200
190
180
170
160
150
TORUS WATER TEMPERATURE (°F)
1 PUMP 0 500 1000 1500 2000 2500 3000 3500 4000
2 PUMPS 0 1000 2000 3000 4000 5000 6000 7000 8000
CORE SPRAY PUMP FLOW (GPM)
6 < -11 IN.


CS VORTEX LIMIT
0
-20
-40
-60
-80
-98
-100
-120
-140
-58
TORUS WATER LEVEL (IN.)
DO NOT OPERATE IN THIS AREA
0 2000 4000 6000 8000
4400
7900
CORE SPRAY PUMP FLOW (GPM)



RPV SATURATION TEMPERATURE
UNRELIABLE RPV
WATER LEVEL
INDICATION AREA
TEMPERATURE NEAR
INSTRUMENT RUN (°F)
450
400
350
300
250
212
200
0
100
200
230
300
1200
RPV PRESSURE (PSIG)

3. For each of the following instruments, the instrument reads above the minimum indicated level associated with the highest temperature near an instrument reference leg vertical run:

A. Wide range level detectors (10 to 220 IN.)
(B21-N081A,B)
(B21-N091A,C)

HIGHEST REACTOR BUILDING RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
-	80	10
80	150	15
150	250	26
250	350	40

B. Wide range level detectors (10 to 220 IN.)
(B21-N081C,D)
(B21-N091B,D)

HIGHEST REACTOR BUILDING RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
-	167	10
167	250	20
250	350	36

C. Flood up level detector (160 to 560 IN.)
(B21-N027)

HIGHEST DRYWELL RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
-	150	175
150	250	190
250	350	210
350	450	237
450	550	274

CAUTIONS

1

An RPV water level instrument may be used to determine RPV water level only when all the following conditions are satisfied for that instrument:

1. The temperatures near all the instrument runs are below the RPV saturation temperature.
2. For each of the instruments in the following table, the instrument reads above the minimum indicated level or the temperatures near all the instrument reference leg vertical runs are below the maximum run temperature.

INSTRUMENT	RANGE(IN.)	MAXIMUM REACTOR BUILDING RUN TEMPERATURE (°F)	MINIMUM INDICATED LEVEL(IN.)
Core Level Detector (B21-N085A)	-150 to 50	327	-142
Core Level Detector (B21-N085B)	-150 to 50	309	-134
Narrow Range Level Detectors (B21-N080A,B) (B21-N095A,C) (C32-N004A,C)	160 to 220	103	169
Narrow Range Level Detectors (B21-N080C,D) (B21-N095B,D) (C32-N004B,D)	160 to 220	273	165

PART 1

CODE I.D. NUMBER AT LEFT BY FILLING IN THE APPROPRIATE BOXES. IF A NUMBER IS LARGER THAN 5, FILL IN 5 PLUS THE DIGIT NEEDED TO ADD UP TO THE DESIRED NUMBER.

WRITE I.D. NUMBER HERE

EXAMPLE

IMPORTANT

- MAKE *DARK* MARKS (USE NO. 2 PENCIL ONLY)
- EXAMPLE: A B C D E
- ERASE *COMPLETELY* TO CHANGE

NAME Fermi 2 Key

SUBJECT RO ILO NRC Exam

DATE 2/4/2000 HOUR

SCANTRON® FORM 2052

TEST RECORD

PART 1

PART 2

TOTAL

FEED THIS DIRECTION

PART 2

(T) (F) KEY

	Key		Key
1	C	51	D
2	D	52	B
3	B	53	C
4	A	54	B
5	A	55	B
6	C	56	A
7	A	57	C
8	C	58	C
9	A	59	A
10	D	60	D
11	C	61	C
12	B	62	D
13	C	63	B
14	B	64	A
15	C	65	B
16	A	66	D
17	B	67	A
18	B	68	B
19	D	69	A
20	C	70	D
21	B	71	A
22	B	72	A
23	C	73	D
24	A	74	A
25	D	75	D
26	C	76	D
27	C	77	B
28	C	78	B
29	D	79	C
30	A	80	A
31	C	81	C
32	A	82	C
33	D	83	A
34	D	84	D
35	C	85	B
36	B	86	C
37	C	87	C
38	A	88	A
39	C	89	C
40	D	90	C
41	C	91	D
42	C	92	B
43	C	93	A
44	D	94	A
45	A	95	C
46	D	96	B
47	B	97	B
48	D	98	A
49	D	99	B
50	B	100	A

4567-CP 999-12 11 10 9 8 7 6 5 4 3 2 1

Exam Answer Key

Reactor Operator

EC-99-11-04
9801 ILO

Q#	Ans.	Pts.	Question Number	References
1.	C	1	EQ-OP-315-0114-000-0008-006	23.139 Section 5.3
2.	D	1	EQ-OP-315-0131-000-0005-005	ST-OP-315-0031-001
3.	B	1	EQ-OP-315-0109-000-0006-005	20.106.01, CRD HYDRAULIC SYSTEM FAILURE
4.	A	1	EQ-OP-315-0105-000-0004-008	ST-OP-315-0028-001 ST-OP-315-0027-001
5.	A	1	EQ-OP-315-0164-000-0005-005	AOP 20.300.11, Loss of 24/48 VDC Battery Busses
6.	C	1	EQ-OP-315-0142-000-0004-006	ST-OP-315-0142-001
7.	A	1	EQ-OP-315-0105-000-0026-008	ST-OP-315-0005-001
8.	C	1	EQ-OP-315-0104-000-0011-006	23.138.01, REACTOR RECIRCULATION SYSTEM
9.	A	1	EQ-OP-315-0127-000-0001-004	ST-OP-315-0027-001
10.	D	1	EQ-OP-802-3004-000-0007-001	ST-OP-802-3004-001
11.	C	1	EQ-OP-802-3006-000-0027-003	29.ESP.02, Alternate Boron Injection, ST-OP-802-3006-001
12.	B	1	EQ-OP-315-0165-000-0019-005	ST-OP- 315-0065-001; 20.300.04, Loss of 4160V ESS Busses

Exam Answer Key

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9801 ILO

Q#	Ans.	Pts.	Question Number	References
13.	C	1	EQ-OP-315-0109-000-0002-002	ST-OP-315-0009-001
14.	B	1	EQ-OP-315-0141-000-0004-006	ST-OP-315-0041-001
15.	C	1	EQ-OP-315-0104-000-0006-006	ST-OP-315-0004-001
16.	A	1	EQ-OP-315-0139-000-0004-013	ST-OP-315-0039-001
17.	B	1	EQ-OP-315-0140-000-0003-001	ST-OP-315-0040-001
18.	B	1	EQ-OP-315-0122-000-0003-010	ST-OP-315-0022-001
19.	D	1	EQ-OP-315-0148-000-0004-008	ST-OP-315-0048
20.	C	1	EQ-OP-315-0127-000-0005-005	I-2155-10
21.	B	1	EQ-OP-315-0141-000-0012-004	ST-OP-315-0041-001
22.	B	1	EQ-OP-315-0108-000-0005-004	ST-OP-315-0008 ST-OP-315-0014
23.	C	1	EQ-OP-315-0141-000-0003-030	ST-OP-315-0141-001
24.	A	1	EQ-OP-315-0150-000-0003-005	ST-OP-315-0050-001, ST-OP-315-0027-001, ST-OP-315-0048-001

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Q#	Ans.	Pts.	Question Number	References
25.	D	1	EQ-OP-802-1001-000-0008-014	22.000.02, Plant Startup to 25% Power, ST-OP-315-0027-001
26.	C	1	EQ-OP-315-0121-000-0003-008	ST-OP-315-0021-001
27.	C	1	EQ-OP-315-0145-000-0010-006	ST-OP-0045-001
28.	C	1	EQ-OP-315-0128-000-0004-003	ST-OP-315-0028-001
29.	D	1	EQ-OP-315-0146-000-0004-017	ST-OP-315-0046-001
30.	A	1	EQ-OP-315-0146-000-0003-007	ST-OP-315-0046-001
31.	C	1	EQ-OP-802-3002-000-0005-004	29-100-01 SH 6
32.	A	1	EQ-OP-315-0116-000-0008-007	Technical Specifications 3.6.2.1
33.	D	1	EQ-OP-802-2002-000-0002-007	ST-OP-315-0010-001
34.	D	1	EQ-OP-315-0126-000-0002-003	ST-OP-315-0025-001
35.	C	1	EQ-OP-315-0123-000-0005-006	ST-OP-315-0123-001
36.	B	1	EQ-OP-315-0141-000-0012-003	ST-OP-315-0041-001

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Q#	Ans.	Pts.	Question Number	References
37.	C	1	EQ-OP-315-0165-000-0015-003	ST-OP-315-0058-001, ST-OP-315-0065-001
38.	A	1	EQ-OP-315-0143-000-0007-004	ST-OP-315-0043-001, 23,206, Reactor Core Isolation Cooling System
39.	C	1	EQ-OP-315-0157-000-0005-002	ST-OP-315-0057-001
40.	D	1	EQ-OP-315-0144-000-0002-006	ST-OP-315-0044-001, RSD DBD
41.	C	1	EQ-OP-315-0032-000-0108-001	20.125.01, Loss of Condenser Vacuum
42.	C	1	EQ-OP-315-0199-000-0004-004	ST-OP-315-0018-001
43.	C	1	EQ-OP-315-0029-000-0108-001	22.000.02, Plant Startup to 25% Power 23.137
44.	D	1	EQ-OP-315-0110-000-0003-003	ST-OP-315-0010-001
45.	A	1	EQ-OP-802-1001-000-0021-005	Technical Specification Basis 3.4.10
46.	D	1	EQ-OP-315-0048-000-0104-001	ST-OP-0048-001, 10CFR100 100.11.(1)
47.	B	1	EQ-OP-315-0010-000-0105-001	ST-OP-315-0090-001
48.	D	1	EQ-OP-315-0127-000-0004-012	ST-OP-315-0021-001 ST-OP-315-0010-001

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Q#	Ans.	Pts.	Question Number	References
49.	D	1	EQ-OP-802-2017-000-0007-003	20.000.19, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM
50.	B	1	EQ-OP-802-3002-000-0107-001	ST-OP-802-3002-001
51.	D	1	EQ-OP-802-2017-000-0002-004	ST-OP-315-0099-001
52.	B	1	EQ-OP-802-3002-000-0106-001	23.203, Low Pressure Core Spray EOP CS NPSH Limit Curve EOP CS Vortex Limit Curve
53.	C	1	EQ-OP-802-3003-000-0009-003	ST-OP-802-3003-001
54.	B	1	EQ-OP-802-3003-000-0122-001	20.ESP.03, Alternate Rod Insertion Methods
55.	B	1	EQ-OP-802-3004-000-0011-004	29.100.01, ST-OP-802-3004-001
56.	A	1	EQ-OP-802-3004-000-0110-001	ST-OP-802-3004-001
57.	C	1	EQ-OP-315-0167-000-0003-002	ST-OP-315-0067-001
58.	C	1	EQ-OP-315-0171-000-0004-007	ST-OP-315-0071-001
59.	A	1	EQ-OP-802-2011-000-0004-001	20.205.01, LOSS OF SHUTDOWN COOLING
60.	D	1	EQ-OP-315-0139-000-0003-023	ST-OP-315-0039-001

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Q#	Ans.	Pts.	Question Number	References
61.	C	1	EQ-OP-315-0150-000-0003-006	ST-OP-315-0050
62.	D	1	EQ-OP-315-0169-000-0004-008	ST-OP-315-0069-001
63.	B	1	EQ-OP-802-2016-000-0011-003	20.000.22, PLANT FIRES
64.	A	1	EQ-OP-315-0146-000-0003-008	ST-OP-315-0046-001
65.	B	1	EQ-OP-315-0104-000-0006-007	EQ-OP-315-0004-001
66.	D	1	EQ-OP-315-0155-000-0003-014	ST-OP-315-0055-001
67.	A	1	EQ-OP-802-2007-000-0001-007	ST-OP-802-2007-001
68.	B	1	EQ-OP-315-0173-000-0005-003	ST-OP-315-0073-001
69.	A	1	EQ-OP-315-0190-000-0005-003	Technical Specification Basis 3.9.1-1 ST-OP-315-0190-001
70.	D	1	EQ-OP-315-0190-000-0012-014	Technical Specification 3.9.9
71.	A	1	EQ-OP-804-0001-000-0003-021	Technical Specifications
72.	A	1	EQ-OP-802-4101-000-0014-003	MGA03

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Q#	Ans.	Pts.	Question Number	References
73.	D	1	EQ-OP-315-0128-000-0009-003	Technical Specifications and Technical Specification bases.
74.	A	1	EQ-OP-508-0001-000-0001-001	ST-GN-508-0000-001
75.	D	1	EQ-OP-315-0120-000-0005-004	23.406, PRIMARY CONTAINMENT NITROGEN INERTING AND PURGE SYSTEM
76.	D	1	EQ-OP-315-0167-000-0004-003	ST-OP-315-0067-001
77.	B	1	EQ-OP-802-2003-000-0014-006	20.138.01, Recirculation Pump Trip
78.	B	1	EQ-OP-315-0166-000-0004-005	SOP 23.426, REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING, SECT 7.0
79.	C	1	EQ-OP-315-0116-000-0002-003	ST-OP-315-0016-001
80.	A	1	EQ-OP-315-0120-000-0006-003	ST-OP-315-0020-001
81.	C	1	EQ-OP-315-0110-000-0004-016	ST-OP-315-0010-001, Drawing ST-OP-315-0010-001
82.	C	1	EQ-OP-315-0111-000-0009-003	ST-OP-315-0111-001
83.	A	1	EQ-OP-315-0141-000-0006-013	ST-OP-315-0041-001
84.	D	1	EQ-OP-315-0113-000-0001-003	ST-OP-315-0013-001

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Q#	Ans.	Pts.	Question Number	References
85.	B	1	EQ-OP-315-0165-000-0018-011	ST-OP-315-0165-0
86.	C	1	EQ-OP-315-0114-000-0007-003	ST-OP-315-0040-001
87.	C	1	EQ-OP-315-0146-000-0004-019	ST-OP-315-0146-001
88.	A	1	EQ-OP-802-3002-000-0006-014	ST-OP-802-3002-001
89.	C	1	EQ-OP-802-3005-000-0014-003	ST-OP-802-3005-001
90.	C	1	EQ-OP-315-0121-000-0003-009	ST-OP-315-0021-001
91.	D	1	EQ-OP-315-0111-000-0002-003	ST-OP-315-0011-001
92.	B	1	EQ-OP-315-0135-000-0008-003	ST-OP-315-0035-001
93.	A	1	EQ-OP-315-0105-000-0032-004	23.137.01 Sect 4.2
94.	A	1	EQ-OP-315-0145-000-0006-008	22.000.02, PLANT STARTUP TO 25% POWER
95.	C	1	EQ-OP-802-2006-000-0016-002	MOP13, Conduct of Refueling and Core Alterations
96.	B	1	EQ-OP-802-4101-000-0033-002	MRP05, ALARA/RWPS

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Q#	Ans.	Pts.	Question Number	References
97.	B	1	EQ-OP-802-2002-000-0015-004	20.109.02
98.	A	1	EQ-OP-802-3002-000-0002-004	29.100.01, Emergency Operating Procedure Flow Charts
99.	B	1	EQ-OP-802-3002-000-0004-010	ST-OP-802-3002-001
100.	A	1	EQ-OP-802-2016-000-0011-004	20.000.22, Plant Fires

U.S. Nuclear Regulatory Commission
Site-Specific
Written Examination

Applicant Information

Name: MASTER EXAMINATION	Region III
Date: FEBRUARY 4, 2000	Facility/Unit: Fermi 2
License Level: SRO	Reactor Type: GE
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

Applicant Certification

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

____
Applicant's Signature

Results

Examination Value	100	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

Question 1.

Reactor pressure is 950 psig and the SLC system has been initiated. The operator has the following indications:

- Run indication for the 'A' pump.
- The SLC tank level is decreasing.
- Pump discharge pressure is 1000 psig.
- The squib valve continuity indicating light is LIT.

The SLC system is:

______ A. NOT injecting. The squib valves have not fired.

______ B. NOT injecting. There is a rupture in the injection line.

______ C. injecting. The continuity indication is not always lost.

______ D. injecting. The continuity indication illuminates on loss of continuity.

Point Value: 1

Question 2.

The plant was operating at 70% rated power when an increase in Main Generator megawatts was observed. The positions of the control rods and recirculation flow rate have not changed.

Which ONE of the following is the cause of the INCREASE in Main Generator megawatts?

______ A. An SRV has inadvertently lifted.

______ B. A turbine bypass valve is partially open.

______ C. Xenon is building toward a peak in the core.

______ D. There has been a loss of feedwater heating.

Point Value: 1

Question 3.

The reactor is in Mode 2, with reactor pressure at 800 psig when the operating CRD pump trips. Before any action can be taken annunciator 3D10, CRD ACCUMULATOR TROUBLE, alarms.

What action is required in accordance with procedure 20.106.01, CRD Hydraulic System Failure?

______ A. Immediately start the standby CRD pump.

______ B. Immediately place the mode switch in SHUTDOWN.

______ C. Within 20 minutes, close C11-F034, Charging Water Header Isolation Valve.

______ D. Within 20 minutes restart at least one CRD pump or place the mode switch in SHUTDOWN.

Point Value: 1

Question 4.

The plant is at 80% power when the turbine trips. The reactor will scram in anticipation of the rapid:

______ A. INCREASE in thermal power

______ B. INCREASE in reactor water level

______ C. DECREASE in reactor water level

______ D. DECREASE in main steam line pressure

Point Value: 1

Question 5.

The plant is operating in mode 1 with all systems aligned for normal operation.
Due to a tagging error, the DC control power to bus 72C is deenergized.
How is bus 72CF affected?

______ A. The normal and alternate source breakers remain in their current positions.

______ B. The normal source breaker opens and the alternate source breaker closes.

______ C. The normal source breaker opens and the alternate source breaker remains open.

______ D. The normal source breaker remains closed and the alternate source breaker closes.

Point Value: 1

Question 6.

The following plant conditions exist:
- ADS has received a valid initiation signal.
- The ADS timer is at 86 seconds and lowering.

Which ONE of the following will reset the ADS timer.

______ A. 'A' and 'B' core spray pumps trip off.

______ B. Drywell pressure decreases to 1.5 psig.

______ C. Reactor water level increases to 55 inches.

______ D. RHR pump discharge pressure decreases to 122 psig.

Point Value: 1

Question 7.

The 130 VDC Distribution Cabinet 2PA2-5 has been de-energized due to a fire. Which SRVs can be opened using the control room pushbuttons?

______ A. All the SRVs

______ B. Only the ADS SRVs

______ C. Only the non-ADS SRVs

______ D. All the ADS and some of the non-ADS SRVs

Point Value: 1

Question 8.

The following plant conditions exist:

- Reactor power 28% and stable.
- Reactor recirculation Pump 'A' is shutdown, the loop is NOT isolated.
- Reactor Recirculation Loop 'B' flow is 51% and stable.

Under which of the following conditions is an increase in core flow prohibited?
(Consider each condition separately.)

______ A. The temperature difference between the two loops is 45° F.

______ B. Isolation of the recirculation loop 'A' from the reactor pressure vessel.

______ C. The temperature difference between the reactor pressure vessel coolant and the coolant in loop A is 60° F.

______ D. The temperature difference between the reactor vessel steam space coolant and the reactor pressure vessel bottom head drain line coolant is 135° F.

Point Value: 1

Question 9.

When the mode switch is placed in SHUTDOWN a scram signal is initiated. After a short time this signal is automatically bypassed.

Which ONE of the following describes why this automatic bypass occurs?

______ A. allows the scram to be reset.

______ B. prevents damage to the CRDM seals.

______ C. prevents the scram discharge volume from over filling.

______ D. prevents the scram solenoid pilot valves from overheating.

Point Value: 1

Question 10.

The Emergency Operating Procedures require emergency depressurization if torus water level cannot be maintained above -38 inches.

What is the basis for requiring emergency depressurization at this point?

______ A. Prevent exceeding LPCI NPSH requirements.

______ B. Prevent exceeding SRV tailpipe back pressure limits.

______ C. Steam discharged from RCIC will not be suppressed.

______ D. Suppression of steam discharged from the RPV cannot be assured.

Point Value: 1

Question 11.

The alternate boron injection flowpath is from the precoat tank to a portable pump that discharges directly to the:

______ A. Heater Feed Pump suction.

______ B. Heater Feed Pump discharge.

______ C. Standby Feedwater Pump suction.

______ D. Standby Feedwater Pump discharge.

Point Value: 1

Question 12.

A loss of 120 KV offsite power has occurred.
EDG 11 has started and loaded.
EDG 12 did NOT start.

Which one of the following prevented EDG 12 from starting?

______ A. Exciter Trip

______ B. Bus lockout on bus 64C

______ C. Fuel oil day tank level of 12"

______ D. Lube oil temperature of 165° F

Point Value: 1

Question 13.

Select the two materials used as neutron absorbers in the Control Rod Blades.

______ A. Cadmium and Boron Carbide

______ B. Hafnium and Stainless Steel

______ C. Hafnium and Boron Carbide

______ D. Cadmium and Stainless Steel

Point Value: 1

Question 14.

Which ONE of the following is the power supply to Division 1 RHR LPCI Injection Mode Initiation Logic?

______ A. MPU 1

______ B. Division 1 130VDC

______ C. Division 1 24/48 VDC

______ D. Bus 72CF (swing bus)

Point Value: 1

Question 15.

Which ONE of the following indicates the required positions for the listed components to satisfy the Reactor Recirculation Pump and Motor Generator Set (RRMG) starting interlocks.

	Field Breaker	Drive Motor Breaker	Discharge Valve
______ A.	Closed	Open	Closed
______ B.	Open	Open	Closed
______ C.	Open	Closed	Closed
______ D.	Closed	Closed	Closed

Point Value: 1

Question 16.

During a plant transient the following sequence of events occur.

- Drywell Pressure rises and stabilizes at 5 psig .
- RPV decreases to 105 inches.
- HPCI automatically starts and injects.
- HPCI subsequently trips on high water level.

Which ONE of the following is required to automatically restart the HPCI System?

______ A. Reactor water level drops below Level 2.

______ B. Reactor water level drops below Level 3.

______ C. Reactor water level drops below Level 8.

______ D. The HPCI Initiation Signal Reset pushbutton is depressed.

Point Value: 1

Question 17.

Following an automatic actuation of the Core Spray System, an electrical fault closes E2150-F036A, Div 1 CS Pmps Torus Suct Vlv.

Which ONE of the following completes the statement concerning the expected response of the Core Spray system to this event?

The 'A' Core Spray Pump will:

______ A. trip on low Suction Pressure.

______ B. continuously operate until damage occurs.

______ C. continuously operate with suction from the Condensate Storage Tank.

______ D. continuously operate with suction from the Div 2 suction cross-connect.

Point Value: 1

Question 18.

While retracting SRM B during a startup, it is observed that the SRM B RETRACT PERMIT light goes out.

Which ONE of the following describes why the SRM RETRACT PERMIT light went out?

_____ A. SRM B failed upscale.

_____ B. SRM B failed downscale.

_____ C. The mode Switch was placed in run.

_____ D. IRM E was down-ranged to range 2.

Point Value: 1

Question 19.

Following a LOCA from 95% power the following conditions exist:

- Reactor power (APRMs) is 0% and stable.
- RPV Level has decreased to 144" and is steady
- Reactor pressure is 750 psig and stable.
- Drywell pressure is 2.5 psig and stable.
- NO area temperature or radiation alarms have actuated
- NO operator actions have been performed since the time of the automatic scram

Based on these conditions, which of the following PCIS isolation signals are present?

- Group 1 Main Steam
- Group 2 Reactor Water Sample
- Group 3 RHR
- Group 8 RCIC
- Group 11 RWCU Outboard
- Group 13 Drywell Sumps
- Group 16 Nitrogen Injection
- Group 18 Primary Containment Pneumatics

______ A. Groups 1, 3, 8, 13, & 18

______ B. Groups 1, 2, 11, 13, & 16

______ C. Groups 2, 3, 13, 16, & 18

______ D. Groups 1, 2, 3, 13, 16, & 18

Point Value: 1

Question 20.

Late in the operating cycle with all control rods withdrawn, RPS Drywell Pressure instrument C71-MYE-N650B, fails upscale. A short time later the fuse (C71-F18C) supplying power to the Scram Trip System A Group 3 Scram Pilot Valve Solenoids blows.
No operator actions have been taken.

In response to these events, __________ control rods will scram into the core.

______ A. 0

______ B. 22

______ C. 47

______ D. 185

Point Value: 1

Question 21.

The following conditions exist 10 minutes after a LOCA event:

- Drywell pressure: 4 psig and stable.
- Reactor Water Level (Wide Range): 25 inches and stable.

Select the response that contains the MINIMUM actions necessary to open the Drywell Spray Inboard Isolation Valve, E1150-F021A.

______ A. Depress the OPEN pushbutton.

______ B. Place Containment Spray Mode Select Switch to MANUAL and depress the OPEN pushbutton.

______ C. Place Containment Spray Mode Select Switch to MANUAL, place 2/3 Core Height Override Switch to MANUAL OVERRIDE, and depress the OPEN pushbutton.

______ D. Shut E1150-F016A, Drywell Spray Outboard Isolation Valve, place Containment Spray Mode Select Switch to MANUAL, place 2/3 Core Height Override Switch to MANUAL OVERRIDE, and depress the OPEN pushbutton.

Point Value: 1

Question 22.

The P603 operator has rotated the SLC injection control switch to the Pump A position. No other isolation signals have been present for the RWCU system. How does the RWCU system respond?

______ A. G3352-F001, RWCU SUPPLY INBD ISO VLV, Closes
G3352-F004, RWCU SUPPLY OTBD ISO VLV, Closes

______ B. G3352-F004, RWCU SUPPLY OTBD ISO VLV, Closes
G3352-F220, RWCU TO FW OTBD CTMT ISO VLV, Closes

______ C. G3352-F001, RWCU SUPPLY INBD ISO VLV, Closes
G3352-F119, RWCU SUPPLY SUCT ISO VLV, Closes

______ D. G3352-F220, RWCU TO FW OTBD CTMT ISO VLV, Closes
G3352-F119, RWCU SUPPLY SUCT ISO VLV, Closes

Point Value: 1

Question 23.

The plant is in cold shutdown with RHR Pump A operating in Shutdown Cooling when the operator observes that reactor coolant temperature has increased. The operator verifies that RHR Pump A is still running and that no annunciators have been received. Which ONE of the following valves, if inadvertently closed, would have caused the increase in temperature.

______ A. E11-F009, RHR SDC Inboard Isolation Valve

______ B. E11-F008, RHR SDC Outboard Isolation Valve

______ C. E11-F003A, RHR Heat Exchanger A Outlet Valve

______ D. E11-F048A, RHR Heat Exchanger A Bypass Valve

Point Value: 1

Question 24.

Placing the B Main Steam Line Radiation Monitor Mode Selector Switch in any position other than OPERATE will actuate a ________.

______ A. half scram and de-energize one half of the MSIV solenoids

______ B. full scram and de-energize one half of the MSIV solenoids

______ C. half scram and de-energize only the inboard AC MSIV solenoids

______ D. full scram and de-energize only the outboard DC MSIV solenoids

Point Value: 1

Question 25.

Which one of the following would prevent continued plant operation during a reactor startup if the mode switch were placed in RUN?

______ A. 1 IRM is failed downscale

______ B. 1 APRM is failed upscale

______ C. 2 IRMs are indicating upscale

______ D. 2 APRMs are indicating downscale

Point Value: 1

Question 26.

A plant transient has occurred. Plant conditions are:

Mode 3.
RPV pressure is 950 psig and stable.
RPV water level is 120 inches and stable.
Recirculation Pumps A & B have tripped.

Which RPV level instrument would provide the most accurate level reading at this time?

_____ A. Floodup

_____ B. Core Level

_____ C. Wide Range

_____ D. Narrow Range

Point Value: 1

Question 27.

Plant conditions are as follows:

- The plant is operating at 20% power.
- The turbine generator is paralleled to the grid.
- The Generator is providing an output of 200MW.
- The TURBINE FLOW LIMIT is set at 25%.
- The REACTOR FLOW LIMIT is set at 115%.
- The PRESSURE REGULATOR is set at 944 psig.
- The Turbine Bypass Valves are CLOSED.
- The SPEED/LOAD is set at 300MW.

How will the plant respond to an INCREASE to 45% power without any further operator action?

______ A. Generator output will remain constant, Bypass Valves will OPEN.

______ B. Generator output will remain constant, Bypass Valves will remain CLOSED.

______ C. Generator output will increase to approximately 250MW, Bypass Valves will OPEN.

______ D. Generator output will increase to approximately 250MW, Bypass Valves will remain CLOSED.

Point Value: 1

Question 28.

Which ONE of the following completes the statement concerning the main turbine generator mechanical overspeed trip testing on-load?

Oil is ported to 1 of 2 trip rings, causing it to operate, however, since it takes:

______ A. both trip rings to cause a trip, the turbine will not trip.

______ B. only 1 trip ring to cause a trip, the turbine will trip and must be restarted.

______ C. only 1 trip ring to cause a trip, the trip is electrically prevented during testing.

______ D. only 1 trip ring to cause a trip, the trip linkage is prevented from operating by a mechanical stop.

Point Value: 1

Question 29.

The plant is operating at 100% power with both feed pumps in service. Narrow range level instrument C32-N004A is failed high and feedwater level control is selected to the "B" level instrument. Narrow range instrument C32-N004C suddenly fails high. Which of the following describes the response of the RFPTs?

______ A. The RFPTs will immediately trip due to satisfying the Level 8 trip logic.

______ B. The RFPTs speed will increase, and the RFPs will steady out and control RPV level 11 inches above setpoint.

______ C. The RFPTs speed will decrease, and the RFPs will steady out and control RPV level 11 inches below setpoint.

______ D. The RFPs will continue to control level since the level instrument C32-N004B is selected for input to Feedwater Level Control.

Point Value: 1

Question 30.

The Post Scram Feedwater Logic circuit:

______ A. prevents inadvertent tripping of the RFPTs on high reactor water level.

______ B. prevents inadvertent tripping of the RFPTs due to low suction pressure.

______ C. ensures that a low level scram condition cannot be reset for a minimum of 10 seconds.

______ D. ensures that adequate subcooling is maintained to prevent recirculation pump cavitation.

Point Value: 1

Question 31.

A LOCA has occurred outside of the primary containment. Plant conditions are as follows:

- Reactor Building Temperature (near all instrument runs) = 220° F and stable.
- Reactor Pressure = 250 psig and stable.
- Drywell Temperature = 155° F and stable.

Assuming the indicated level on each of the below instruments is 163 inches, which level instrument may be used for trending indication?

______ A. B21-N027

______ B. B21-N080A

______ C. B21-N091B

______ D. B21-N095C

Point Value: 1

Question 32.

With the plant at 100% power and no testing in progress, what is the maximum Suppression Chamber water temperature allowed without entry into a Limiting Condition for Operation?

_____ A. 95° F

_____ B. 105° F

_____ C. 110° F

_____ D. 120° F

Point Value: 1

Question 33.

Plant conditions are as follows:

- 3 turbine stop valves have drifted to 80% open.
- 50% of the control rods fully inserted.
- The remaining are scattered throughout the core at positions 02 to 48.
- The individual blue lights for each control rod on the full-core display are illuminated.
- The eight scram solenoid group indicating lights are extinguished.

Based on these indications, which of the following is the cause for the failure of the rods to insert?

_____ A. Failure of scram inlet and outlet valves to open.

_____ B. Failure of the back-up scram valves to de-energize.

_____ C. Failure of scram pilot valve solenoids to de-energize.

_____ D. Failure of RPS to detect water in the scram discharge volume.

Point Value: 1

Question 34.

The Rod Block Monitor setpoint circuitry:

______ A. provides alarms when power level approaches to within 2% of the trip setpoint.

______ B. generates three flow biased trip reference signals using a flow signal from the APRM.

______ C. allows the operator to manually select the proper power level setpoint during operation.

______ D. selects the appropriate setpoint when a rod is selected based on current reactor power.

Point Value: 1

Question 35.

The reactor is operating in mode 2 when an IRM mode switch is placed in STANDBY. In addition to a half scram occurring, the IRM will read:

______ A. downscale on the meter and indicate an Inop trip

______ B. downscale on the meter and indicate a Downscale trip

______ C. current power level on the meter and indicate an Inop trip

______ D. current power level on the meter and indicate a Downscale trip

Point Value: 1

Question 36.

The following plant conditions exist:

- Drywell pressure is 3 psig and stable.
- Reactor water level is 180 inches and stable.

Which ONE of the following lists the only valves that require the CONTAINMENT SPRAY MODE SELECT switch to be in the MANUAL position in order to open the valves?

______ A. E11-F024A, RHR Loop A Suppression Pool Cooling/Test Isolation Valve
E11-F026B, RHR Warmup Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ B. E11-F016A, Drywell Spray Outboard Isolation Valve
E11-F024B, RHR Loop B Suppression Pool Cooling/Test Isolation Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ C. E11-F015A, LPCI Loop A Inboard Isolation Valve
E11-F016A, Drywell Spray Outboard Isolation Valve
E11-F028A, RHR Loop A Containment Spray/Test Isolation Valve

______ D. E11-F016B, Drywell Spray Outboard Isolation Valve
E11-F027A, RHR Loop A Suppression Chamber Spray Inboard Isolation Valve
E11-F048A, RHR Heat Exchanger A Bypass Valve

Point Value: 1

Question 37.

Following a loss of power to Bus 64B, the bus was subsequently restored to the normal lineup EXCEPT the operators neglected to reset the digital load sequencer. Following the restoration, all power is again lost to Bus 64B. How will the EDG and electrical distribution system respond to this event?

______ A. The EDG will require a manual start. The loads on bus 64B will sequence after the output breaker is closed.

______ B. The EDG will require a manual start. The loads on bus 64B will NOT sequence after the output breaker is closed.

______ C. The EDG will automatically start. The loads on bus 64B will sequence after the output breaker is closed.

______ D. The EDG will automatically start. The loads on bus 64B will NOT sequence after the output breaker is closed.

Point Value: 1

Question 38.

Procedure 23.206, Reactor Core Isolation Cooling System, contains a precaution concerning extended operation of the RCIC turbine below 2100 RPM. The purpose of this precaution is to prevent:

______ A. bearing damage.

______ B. oil cooler damage.

______ C. the stop valve from closing spuriously.

______ D. turbine damage due to inadequate steam cooling.

Point Value: 1

Question 39.

Which ONE of the following will trip the 345kV Switchyard source breaker to System Service Transformer 65:

______ A. Low bus voltage

______ B. Low transformer oil level

______ C. Primary phase overcurrent

______ D. Transformer high temperature

Point Value: 1

Question 40.

The plant is being controlled using the Remote Shutdown System when a loss of Division 2 DC Power occurs. Which ONE of the following describes how this loss effects operations from the Remote Shutdown Panel.

______ A. RCIC flow indication and drywell pressure indication is lost.

______ B. The ability to operate Mechanical Draft Cooling Tower Fans in fast speed is lost.

______ C. The ability to operate RHR SDC Suction Inboard Isolation Valve, E1150-F009, is lost.

______ D. The ability to operate RHR SDC Suction Outboard Isolation Valve, E1150-F008, is lost.

Point Value: 1

Question 41.

Main condenser vacuum is 1 psia and a loss of vacuum is in progress. What is the MINIMUM condenser pressure when the main condenser will be COMPLETELY lost as a heat sink for any decay heat removal?

______ A. 2.7 psia

______ B. 6.8 psia

______ C. 6.85 psia

______ D. 12.2 psia

Point Value: 1

Question 42.

Due to fire in the control room the plant is being controlled from the Dedicated Shutdown Panel. Plant conditions are as follows:

- Reactor pressure is being controlled using SRVs.
- Reactor water level is 218 inches and increasing.

Which ONE of the following describes the reason for the increasing water level?

______ A. The SBFW System does not have a level 8 isolation.

______ B. The RPV Level Instrument is malfunctioning otherwise SBFW Pump would have tripped.

______ C. The level 8 isolation of SBFW is not operable when operating from the Dedicated Shutdown Panel.

______ D. When operating from the Dedicated Shutdown Panel, the setpoint for the RPV Level 8 isolation is raised 10 inches to provide a wider level control band.

Point Value: 1

Question 43.

Procedures 22.000.02, Plant Startup to 25% Power, and 23.137, Nuclear Boiler System, have cautions to prevent opening MSIVs with excessive differential pressure across their seats. One reason for this precaution is due to the potential damage to MSIVs and/or other Main Steam Line components. Another reason for this caution is to prevent:

______ A. Over speeding the turbine.

______ B. Loss of condenser vacuum.

______ C. Transients in indicated RPV level.

______ D. Damage to the MSL flow restrictors.

Point Value: 1

Question 44.

The Control Rod Drive Hydraulic system is operating normally when C11-F412, CRD Pump Suction Pressure Control Valve, fails closed. How does the system respond?

______ A. The running CRD pump trips on low suction pressure.

______ B. The Demin Water Pumps auto start to supply CRD Pump.

______ C. A check valve opens to supply the CRD system from the Torus.

______ D. A check valve opens to supply the CRD system from the Condensate Storage Tank.

Point Value: 1

Question 45.

The basis for limiting the Reactor Pressure Vessel (RPV) heatup and cooldown rates to 100° F/hr is to satisfy the stress limits for:

______ A. RPV cyclic operation.

______ B. RPV steady state operation.

______ C. feedwater nozzles.

______ D. jet pump expansion/contraction

Point Value: 1

Question 46.

There has been a transient involving the failure of fuel and the breach of a nuclear system process barrier. One hour and 30 minutes after the accident an off-site survey determines that the dose rate at the site boundary is 20 Rem per hour whole body and 180 Rem per hour thyroid and steady. Given this data, what can be determined about the Primary Containment Isolation System (PCIS)?

______ A. The system is meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR20.

______ B. The system is NOT meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR20.

______ C. The system is meeting its design bases of maintaining the dose at the site boundary within the limits of 10CFR100.

______ D. The system is NOT meeting its design bases of maintaining the dose at the boundary within the limits of 10CFR100.

Point Value: 1

Question 47.

Which ONE of the following completes the following statement.

No more than ____(1)____ fuel bundles shall be allowed in or around the fuel prep machines. The fuel shall be separated from the main body of other fuel by at least _____(2)_____ inches.

	(1)	(2)
______ A.	2	6
______ B.	2	12
______ C.	3	6
______ D.	3	12

Point Value: 1

Question 48.

The main turbine trips at 45% power. RPS B scram relays fail to reposition when deenergized causing a failure to scram.

ASSUMING NO OPERATOR ACTIONS, Which ONE of the following statements describes the expected response to this situation?

______ A. The backup scram valves associated with RPS A will energize when the RPS A scram relays trip.

______ B. The backup scram valves associated with RPS A will deenergize when the RPS A scram relays trip.

______ C. The alternate rod insertion valves deenergize when the high reactor pressure setpoint is exceeded on one trip unit in each trip channel.

______ D. The alternate rod insertion valves energize when the high reactor pressure setpoint is exceeded on both trip units in one trip channel.

Point Value: 1

Question 49.

Per Abnormal Operating Procedure (AOP) 20.000.19, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM, which ONE of the following is the preferred method of scramming the reactor if you are UNABLE to do so prior to leaving the control room?

______ A. Depress the Main Turbine Trip buttons on H11-P632, Turbine Protection Cabinet.

______ B. Open RPS Motor-Generator Output Breakers on RPS MG Set A and B Control Panels AB3-H11.

______ C. Insert a trip signal for RPS A and B channels on H11-P609/611, Relay Room RPS Logic Cabinets.

______ D. Scram Reactor by taking two operable APRM Mode Switches out of OPER at H11-P608, Power Range Neutron Monitoring Cabinet.

Point Value: 1

Question 50.

Which ONE of the following statements describe the relationship between the Boron Injection Initiation Temperature (BIIT) curve and the Heat Capacity Limit (HCL) curve?

______ A. The HCL curve, based on reactor pressure, determines how long the operator has until the limits of the BIIT curve are reached requiring boron injection.

______ B. Operating by the BIIT curve ensures that enough boron is in the core to shutdown the reactor before the suppression pool reaches the limits of the HCL curve.

______ C. Operating by either the BIIT curve or the HCL curve, whichever is most restrictive, ensures the limits of the Primary Containment Pressure Limit curve will not be reached.

______ D. The BIIT curve and the HCL curve combined, monitor the energy additions to the suppression pool to prevent exceeding the limits of the Pressure Suppression Pressure curve.

Point Value: 1

Question 51.

The Dedicated Shutdown System transfer switches are designed to:

______ A. Bypass all interlocks and breaker protective functions.

______ B. Transfer equipment control power to a source powered directly from the control room or relay room.

______ C. Transfer indicating instrument power to sources powered directly from the control room or relay room.

______ D. Provide electrical separation between the Main Control Room and the Local Panel controls.

Point Value: 1

Question 52.

The following plant conditions exist following a LOCA event:

- Torus Water Temperature is 180° F and stable.
- Torus Water level is -70 inches and stable.
- Drywell Pressure is 0 psig and stable.
- Reactor Pressure 0 psig and stable.
- Core Spray A is injecting into the Division 1 loop.
- Core Spray B is injecting into the Division 2 loop.

Assuming no change in the above conditions, what is the maximum flow allowed for the Division 1 Core Spray Loop?

______ A. 3500 gpm

______ B. 3800 gpm

______ C. 7000 gpm

______ D. 7600 gpm

Point Value: 1

Question 53.

A plant transient has occurred.
Reactor pressure is 350 psig and steady.
Which ONE of the following pumps are capable of injecting at this time?

______ A. Core Spray Pumps

______ B. Condenser Pumps

______ C. Heater Feed Pumps

______ D. RHR Service Water Pumps

Point Value: 1

Question 54.

A reactor scram has occurred and all control rods are NOT inserted.

Which one of the following methods for inserting control rods REQUIRES the scram to be reset?

______ A. Vent CRD Overpiston Volumes

______ B. Open the Scram Test Switches

______ C. Manual Control Rod Insertion

______ D. Increase the CRD Cooling Water Differential Pressure

Point Value: 1

Question 55.

If hydrogen concentration can not be determined to be <6% AND oxygen concentration can not be determined to be <5%, the drywell fans must be shutdown.

Which ONE of the following describes the reason for this requirement?

_____ A. Allow drywell sprays to be initiated.

_____ B. Eliminate a potential ignition source.

_____ C. Prevent the hydrogen and oxygen from combining into a flammable mixture.

_____ D. Allow the hydrogen to accumulate in the top of the drywell where the oxygen concentration is the least.

Point Value: 1

Question 56.

Which ONE of the following is the reason for initiating drywell sprays before drywell temperature exceeds 340° F per 29.100.01, Sheet 2, Primary Containment Control?

_____ A. To prevent exceeding the design temperature of the drywell structure.

_____ B. To minimize damage to non-environmentally qualified drywell components.

_____ C. To minimize the damage to the drywell coolers and the reactor recirculation pumps.

_____ D. To prevent exceeding the design capacity of the torus-to-drywell vacuum breakers upon spray initiation.

Point Value: 1

Question 57.

The plant has experienced a small break LOCA. The following plant conditions exist:

- Reactor Water level is 175 inches and stable.
- Drywell Temperature is 187°F and stable.
- Drywell Pressure is 1.9 psig and stable.
- Suppression Pool water level is -1 inch and stable.
- Suppression Pool temperature is 94°F and stable.
- No operator actions have been taken

Given these conditions, Which ONE of the following describes the status of the EECW/RBCCW System.?

______ A. RBCCW will be supplying all loads. No components are being supplied by EECW.

______ B. EECW will be supplying Control Air Compressor Space Coolers, ESF Battery Rooms AC unit, Core Spray Pumps Seal Coolers. RBCCW will be isolated from essential components.

______ C. ECCW will be supplying Control Air Compressor Space Coolers, CCHVAC Condenser, Core Spray Space Coolers, and Thermal Recombiner Space Coolers. RBCCW will be isolated from essential components.

______ D. EECW will be supplying the Standby Gas Treatment System Space Coolers, EECW Space Cooler, Drywell Penetration Coolers, and the Post Accident Sampling System. RBCCW will be isolated from essential components.

Point Value: 1

Question 58.

Due to a large leak in the system, the Station Air Compressors are unable to maintain Station Air Header pressure above 72 psig.
Which ONE of the following indicates the current status of the air system.

_____ A. P50-F440/441 Control Air Isolation valves have OPENED, Station Air compressors have TRIPPED.

_____ B. P50-F440/441 Control Air Isolation valves have CLOSED, P50-F403 Div 2 NIAS Crosstie to IAS valve has OPENED

_____ C. Div 1 and Div 2 Control Air Compressors are RUNNING, P50-F440/441 Control Air Isolation valves have CLOSED.

_____ D. P50-F473 Station Air to IAS Isolation Valve has CLOSED, Div 1 and Div 2 Control Air Compressors are NOT RUNNING.

Point Value: 1

Question 59.

The plant is in extended maintenance shutdown in Mode 4 with the following conditions:
- A loss of shutdown cooling occurs.
- No RHR pumps can be recovered and placed into shutdown cooling.
- The North Reactor Recirc pump is running.

Which ONE of the following is an allowable option for alternate shutdown cooling under these conditions in accordance with 20.205.01, Loss of Shutdown Cooling?

_____ A. RWCU Blowdown to the Main Condenser, makeup with CRD pumps.

_____ B. Maximizing FPCCU flow and RBCCW Flow to the FPCCU Heat Exchangers.

_____ C. Bleed Steam via SRVs when pressure reaches 100 psig. Makeup with Core Spray.

_____ D. Bleed Steam Via Bypass Valves when pressure reaches 50 psig. Makeup with SBFW.

Point Value: 1

Question 60.

Which HPCI valve DIRECTLY receives a high suppression pool water level signal to reposition?

_____ A. HPCI Minimum Flow Valve (E41-F012)

_____ B. CST to HPCI Suction Isolation Valve (E41-F004)

_____ C. HPCI test return to CST Isolation Valve (E41-F011)

_____ D. Suppresion Pool to HPCI Suction Isolation Valve (E41-F042)

Point Value: 1

Question 61.

The Reactor Building Ventilation Exhaust Radiation readings have been showing a steady increase. Which of the following is the MINIMUM Reactor Building Ventilation Exhaust Radiation Level above which the SGTS will automatically start?

______ A. 10,000 CPM

______ B. 12,000 CPM

______ C. 16,000 CPM

______ D. 20,000 CPM

Point Value: 1

Question 62.

Which signal will cause a DIRECT trip of the Torus Water Management Pumps?

______ A. High drywell pressure

______ B. RPV water level (Level 2)

______ C. Hi-Hi DW Equipment Drain Sump Level

______ D. Hi-Hi Torus Room Floor Drain Sump Level

Point Value: 1

Question 63.

You are the CRNSO in the control room.
Which of the following is an Immediate Operator Action for a confirmed fire in accordance with 20.000.22, Plant Fires?

_____ A. Identify the type (class) of fire.

_____ B. Announce the fire alarm over the Hi-Com system.

_____ C. Send an operator to verify the magnitude and location of the fire.

_____ D. Establish communications between Control Room and Fire Brigade.

Point Value: 1

Question 64.

Which of the following configurations will produce a RFPT speed change in response to a change in total steam flow?

_____ A. Master Level Controller in Auto, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in three element

_____ B. Master Level Controller in Manual, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in three element

_____ C. Master Level Controller in Auto, S RFPT Flow Control M/A Station in Manual, N RFPT Flow Control M/A Station in Manual, Level Control Mode Select switch in single element

_____ D. Master Level Controller in Manual, S RFPT Flow Control M/A Station in Auto, N RFPT Flow Control M/A Station in Auto, Level Control Mode Select switch in single element

Point Value: 1

Question 65.

The recirculation pump speed is limited if feedwater flow is less than 20%. The purpose of this recirculation pump speed limiter is to ensure:

______ A. reactor power will not exceed the feedwater pump capability.

______ B. the recirculation pumps have adequate net positive suction head.

______ C. the feedwater heaters are on-line prior to higher power operations.

______ D. reactor feed pump turbines are operating on low pressure steam supply prior to increasing reactor power.

Point Value: 1

Question 66.

A fault in the AC power supply has caused the Main Generator Automatic Voltage Regulator (AVR) master channel to fail during normal power operations. How is main generator operation affected?

______ A. The Main Turbine Generator trips.

______ B. The Main Generator Field Breaker trips open.

______ C. The AVR master channel shifts to MANUAL control.

______ D. The AVR master channel shifts to the AVR slave channel.

Point Value: 1

Question 67.

The plant is shutown in mode 5.
A fuel bundle has been dropped in the spent fuel pool.
Which ONE of the following would be the cause of increase in offsite radiation release rates?

______ A. Both Rail Airlock door seals are INOPERABLE.

______ B. Pressure in the Secondary Containment is -0.132 inches wc.

______ C. T41-F011, Reactor Building Inboard Supply Damper, is INOPERABLE and closed.

______ D. CCHVAC is operating in recirc mode.

Point Value: 1

Question 68.

When in Recirculation mode, how is the positive differential pressure in the Control Center Envelope maintained?

______ A. Modulating vanes on the CCHVAC Exhaust fans.

______ B. Modulating the Emergency Makeup and Recirculation dampers.

______ C. Modulating the Emergency Exhaust and Inlet Pressure control dampers.

______ D. Modulating the Emergency Makeup and Inlet Pressure control dampers.

Point Value: 1

Question 69.

The plant is in a refueling outage with the mode switch in REFUEL and all rods inserted. The refuel crew has used the grapple to pick up a fuel bundle. They start to move towards the core when the control room operator withdraws a rod. When the bridge reaches the core area the bridge will:

_____ A. stop and a hoist block will be generated

_____ B. stop and the hoist will remain operable

_____ C. continue moving and a hoist block will be generated

_____ D. continue moving and the hoist will remain operable

Point Value: 1

Question 70.

What is the technical specification limit for water level above the top of the RPV flange when the plant is in Mode 5 and fuel movement is in progress?

_____ A. 19.0 feet

_____ B. 19.5 feet

_____ C. 20 feet

_____ D. 20.5 feet

Point Value: 1

Question 71.

Select the conditions that are necessary for the plant to be in COLD SHUTDOWN mode.

______ A. Mode switch in Shutdown, Coolant temperature less than 200° F and all head closure bolts fully tensioned.

______ B. Mode switch in Shutdown or Refuel, Coolant temperature less than 212° F and all head closure bolts fully tensioned.

______ C. Mode switch in Shutdown or Refuel, Coolant temperture less than 200° F and all head closures bolts fully tensioned.

______ D. Mode switch in Shutdown, Coolant temperature less than 212° F and no more than one head closure bolt detensioned.

Point Value: 1

Question 72.

I & C technicians are preparing to perform a surveillance procedure on an instrument. They observe that a prerequisite to the procedure cannot be met under present plant conditions. The prerequisite will not affect the performance of the procedure and the surveillance will be past it's critical date the next day. Which ONE of the following actions must be taken in order to continue with the surviellance?

______ A. The NASS must waive the prerequisite.

______ B. The I&C supervisor must waive the prerequisite.

______ C. The I&C tech must mark the step as N/A (Not Applicable).

______ D. The I&C supervisor must mark the step as N/A (Not Applicable).

Point Value: 1

Question 73.

The plant is at 40% power conducting the quarterly surveillance test on the turbine bypass valves when the following occurs:

- A turbine bypass valve fails to 100% open and cannot be closed.
- 3D91 Turbine Stop/Cont Val Channel Trip Bypassed alarms.

Which ONE of the following contains a complete list of ALL the equipment that must be considered INOPERABLE in response to this event.

______ A. Turbine bypass valves

______ B. RPS Turbine control valve trip function

______ C. Turbine bypass valves and RPS turbine control valve trip function

______ D. Turbine bypass valves, RPS turbine control valve and turbine stop valve trip functions

Point Value: 1

Question 74.

You have donned PCs to hang a tag in a contaminated area where mechanics are disassembling a RWCU valve. The area is also posted as a High Radiation Area with a survey reading of 250 mr/hr in the work area. While attempting to reach the valve requiring a tag, you catch your rubber overshoe on an obstruction and it tears and comes off. You should:

______ A. leave the area and check yourself for contamination.

______ B. stand fast and request one of the mechanics to call Radiation Protection.

______ C. continue hanging the tag to reduce the additional exposure resulting from exiting and re-entry into the area, but attempt to prevent tearing the shoe covers.

______ D. put the overshoe back on attempting to prevent damage to the shoe cover. When leaving the area perform a thorough frisk and inform Radiation Protection of the occurrence.

Point Value: 1

Question 75.

In accordance with 23.406, Standby Gas Treatment System procedure; when performing a containment purge, the Reactor Building HVAC system is the preferred purge path UNLESS:

______ A. SGTS system has both trains operable.

______ B. Effluent release monitoring is required.

______ C. The containment is NOT completely inerted.

______ D. SGTS is needed to comply with the Effluent Release Limit.

Point Value: 1

Question 76.

The plant is operating at 96% power.
The Uninterruptible Power Supply (UPS) battery is isolated for maintenance.
Breaker 72M position 3D is open for repairs on UPS A rectifier.
The fuses for 72R position 2A blow, (UPS B rectifier).
Based on the status of the UPS which ONE of the following actions is taken?

______ A. Place the Mode Switch in Shutdown based on the loss of Feedwater in accordance with 20.107.01, Loss of Feedwater or Feedwater Control procedure.

______ B. Emergency Depressurize the reactor based on the loss of the ADS timer in accordance with the Emergency Operating Procedure, Sht 1, RPV Control.

______ C. Verify UPS buses A & B are energized on ALTERNATE through the Static Transfer Switch in accordance with 20.300.12, Loss of UPS procedure.

______ D. Manually transfer UPS buses A & B to ALTERNATE using the Bypass to Alternate Transfer Switch in accordance with 23.308.01, UPS procedure.

Point Value: 1

Question 77.

You are implementing the Emergency Operating Procedures during an ATWS. The step for confirming the mode switch in shutdown has a small box attached by a dotted line containing the text "EP-101 SA2".
This information listed on the EOP flowchart describes:

______ A. an override required by the EOP flowcharts.

______ B. emergency support procedures which must be implemented at that point.

______ C. direction to branch to the listed procedure if the action taken at that point is successful.

______ D. a condition may exist requiring classification of an event per the site emergency plan.

Point Value: 1

Question 78.

The Rod Worth Minimizer fails during a reactor startup and must be bypassed. Technical Specifications require that a second licensed operator or other technically qualified member of the unit technical staff be present at the reactor control console to verify control rod movement and compliance with the prescribed control rod pattern until:

______ A. The RBM is verified operable.

______ B. Reactor Power reaches 10% power.

______ C. Reactor Power reaches 30% power.

______ D. More than 12 rods have been withdrawn.

Point Value: 1

Question 79.

Which ONE of the following describes the use of level indication if Drywell/Reactor Building Run Temperature is BETWEEN the two values stated in Emergency Operating Procedure Caution #1, Part 3, for the instrument of concern? Assume all temperatures are less than RPV saturation temperature.

______ A. The level instrument may be used for accurate level trending through its full range.

______ B. The level instrtument may be used for trending only above the MINIMUM INDICATED LEVEL.

______ C. The level instrtument may be used for trending only below the MINIMUM INDICATED LEVEL.

______ D. The level indicator may be used only for actual level at the MINIMUM INDICATED LEVEL.

Point Value: 1

Question 80.

The reactor is operating in normal two-loop operation at 85% power and 80% flow. A Main Steam Isolation Valve disc becomes separated from the stem and rapidly shuts. During the transient the following indications are observed:

- APRMs spike to 113.5% and returns to 100%
- Reactor Pressure spikes to 1120 psig and returns to 1000 psig
- Reactor water level lowers to 182" and returns to 216 inches

What Limiting Safety System Setting has been exceeded?

______ A. Reactor water level

______ B. ARPM fixed neutron flux

______ C. Reactor steam dome pressure

______ D. APRM simulated thermal power

Point Value: 1

Question 81.

The General Operating Procedure for reactor shutdown and cooldown directs the operator to raise water level to above 220" prior to entering Shutdown Cooling (SDC).
What is the reason for raising water level above the normal level?

______ A. Ensures adequate NPSH for RHR pumps.

______ B. Satisfies the interlock required for opening RHR SDC valves.

______ C. Provides additional inventory for RHR system warmup in preparation for SDC.

______ D. Provides adequate natural circulation to minimize temperature stratification during SDC.

Point Value: 1

Question 82.

During normal full power operation with no alarms or other unusual indications, two Torus to Drywell vacuum breakers open. Which ONE of the following would cause the vacuum breakers to open?

______ A. Torus pressure is less than the drywell pressure.

______ B. Drywell pressure is less than Reactor Building pressure.

______ C. Torus pressure is higher than the drywell pressure.

______ D. Drywell and torus pressures are higher than Reactor Building pressure.

Point Value: 1

Question 83.

Primary Containment Control EOP Flowchart step TWL-6 directs 'SECURE HPCI (DISREGARD ADEQUATE CORE COOLING)' if suppression pool level cannot be maintained above -68 inches.

Which ONE of the following describes the operation of RCIC and the associated reason when suppression pool level cannot be maintained above -68 inches?

______ A. RCIC will be tripped and isolated at the same time the EOP directs the securing of HPCI to minimize the containment pressure rise.

______ B. RCIC will be secured manually prior to HPCI due to erratic turbine operation as exhaust backpressure increases.

______ C. RCIC operation may continue since the turbine exhaust energy does not contribute excessively to increasing containment pressure.

______ D. RCIC operation may continue only if it is the last operable high pressure injection system available to provide adequate core cooling.

Point Value: 1

Question 84.

A loss of all high-pressure injection systems has resulted in RPV level decreasing to the point where an Emergency RPV Depressurization is required. Which ONE of the following states the reason that a minimum of 4 SRVs must be opened?

______ A. Ensures that the reactor will be depressurized to below ECCS shut off head before RPV level can decrease to below TAF.

______ B. Ensures that at the worst case in core life, the APLHGR thermal limit will not be exceeded and inhibit adequate radiant heat transfer.

______ C. Prevents exceeding 1% plastic strain on the hottest fuel pin in the core allowing fuel cladding failure to release radioactive fission products.

______ D. Ensures that sufficient steam flow will exist to remove decay heat at low enough pressure for the lowest head ECCS pump to make up for steam flow.

Point Value: 1

Question 85.

Emergency Operating Procedure 29.000.01 Sheet 5, Secondary Containment and Rad Release, Step SC-4 requires that "WHEN a primary system is discharging into secondary containment", action is to be taken. Which of the following components would meet the definition of a primary system for the purposes of this step?

______ A. Main steam drain lines

______ B. Core spray suction line from torus

______ C. Standby gas treatment vent piping to drywell

______ D. RBCCW supply to drywell atmospheric coolers

Point Value: 1

Question 86.

The reactor has been operating at 22% power with the steam tunnel coolers OFF. The NASS has directed the CRNSO to start the East Steam Tunnel Cooler. What is the concern with starting the cooler under the present plant conditions?

______ A. The Steam Tunnel Cooler could be temporarily overloaded.

______ B. A Steam Tunnel High Temperature Group 1 Isolation could occur.

______ C. External pressure on the Steam Tunnel may exceed its design limit.

______ D. RBCCW temperature control would be lost.

Point Value: 1

Question 87.

The following initial conditions exist:

Reactor Power	96% and stable.
Control Room Pressure	+ 0.175" $H_2$0 and stable.
Reactor Building Pressure	+0.10" $H_2$0 and stable.
Turbine Building Pressure	0" $H_2$0 and stable.

Which one of the following actions is required?

______ A. Enter 20.413.01, Control Center HVAC System Failure.

______ B. Enter 29.100.01, Secondary Containment and Radioactive Release.

______ C. Enter ARP 8D46, Div I Reactor BLGD Press High/Low. Trip RBHVAC and start SGTS.

______ D. Enter 20.000.11, Loss of Secondary Containment, trip RBHVAC Fans and start SGTS.

Point Value: 1

Question 88.

Drywell to Torus differential pressure is limited during normal operation to prevent:

______ A. damage or failure from "water slugs" if an SRV were to lift.

______ B. bypassing the suppression pool in the event of a steam blowdown from the drywell.

______ C. failure of the primary containment due to high external pressure on the vent pipes.

______ D. damage or failure from "water slugs" in the event of a steam blowdown from the drywell.

Point Value: 1

Question 89.

When would maintenance on the Containment Personnel Airlock doors which requires BOTH doors to be inoperable, be allowed by the Technical Specifications without entering into required actions?

______ A. The primary containment remains inerted.

______ B. An operator is dedicated to close the door.

______ C. The maintenance will last less than 24 hours.

______ D. Reactor Coolant Temperature is 185°F.

Point Value: 1

Question 90.

What is Fermi policy for Unit-2 licensed personnel shift staffing (in addition to any licensed STA) while in the Hot Shutdown condition?

______ A. One SRO license and one RO license

______ B. One SRO license and two RO licenses.

______ C. Two SRO licenses and one RO license.

______ D. Two SRO licenses and two RO licenses.

Point Value: 1

Question 91.

Who is RESPONSIBLE for entering data into the Fermi 2 Unit Log concerning start and completion of allowable time limits for both Technical Specification compliance and requirements of Operability Determinations?

______ A. NSS

______ B. STA

______ C. NASS

______ D. CRNSO

Point Value: 1

Question 92.

The Safety Limits for the Reactor Core are significantly different for low pressure-low flow conditions and high pressure-high flow conditions. Why is the low pressure-low flow limit a specific thermal power whereas the high-pressure-high flow limit is based on MCPR?

______ A. Under low pressure-low flow conditions, fuel cladding cracking due to internal fuel rod pressure and high centerline temperatures occurs more readily requiring power to be held to a fixed low value.

______ B. Since the energy necessary to convert water to steam under low pressure conditions is greater, the centerline temperature required to produce steam must be higher; therefore thermal power must be held to a low fixed value.

______ C. Since subcooling is greater in the lower part of the core under low pressure or low flow conditions, moderation is greater and power to reach the onset of transition boiling (OTB) can be more closely approached.

______ D. The approved correlation used to develop the MCPR is not valid under low pressure-low flow conditions. Therefore, limiting thermal power to a low fixed value significantly below the power necessary to reach OTB is conservative.

Point Value: 1

Question 93.

During refueling operations a Refuel Floor Log is maintained. Who is responsible for maintaining this log in accordance with procedure MOP13, Refueling Operations?

______ A. NASS

______ B. Reactor Engineer

______ C. Refuel Bridge Operator

______ D. Refuel Floor Supervisor

Point Value: 1

Question 94.

You have just left a contaminated area and are required to perform a frisk using a hand held frisker. The meter is reading 340 cpm. You are required to:

______ A. perform a whole body frisk at a rate of .5 inches per minute.

______ B. perform a whole body frisk at a rate of 2 inches per minute.

______ C. request for Radiation Protection assistance or move to another area.

______ D. set the hand held frisker response switch to the FAST position.

Point Value: 1

Question 95.

A safety related component in the Reactor Building has been manipulated to return a system to operable status after maintenance. The component is in an area where there is an airborne radioactivity hazard. The person who would independently verify the lineup would receive exposure of .1 DAC. The NSS may:

______ A. NOT waive the independent verification since the component is safety related.

______ B. NOT waive the independent verification since the exposure is not considered excessive.

______ C. waive the independent verification due to the excessive airborne exposure hazard.

______ D. waive the independent verification provided concurrence is obtained from a another SRO.

Point Value: 1

Question 96.

Plant conditions are as follows:

- A RCIC surveillance is in progress.
- Torus water temperature has risen to 99°F.
- Division 1 RHR is operating in the torus cooling mode.

The crew has not entered the EOPs to address this condition.
Which ONE of the following describes why the crew chose not to enter the EOP?

______ A. Division 2 RHR is available for Torus cooling.

______ B. Division 1 RHR was already in service in the torus cooling mode.

______ C. TWT leg is entered only subsequent to a high drywell pressure signal.

______ D. Temperature is below the testing limit LCO for torus water temperature.

Point Value: 1

Question 97.

Following a LOCA, the following plant conditions exist:

- Reactor Water Level is 122 inches and steady.
- Drywell Pressure is 1.0 psig and steady.
- Reactor Pressue is 910 psig and steady.
- Torus Water level is +4 inches and steady.
- Torus water temperature is 148° F and steady.
- HPCI is maintaining level.
- RCIC is running on minimum flow bypass.

Continued operation with these conditions could cause:

______ A. RCIC Turbine trip due to high exhaust back pressure.

______ B. HPCI Turbine or equipment damage due to lubricating oil failure.

______ C. Drywell Pressure to increase due to incomplete steam condensation.

______ D. Inaccurate Torus Water Level indication due to lack of temperature compensation.

Point Value: 1

Question 98.

A special test of the CCHVAC system is in progress.
- During the test, CCHVAC shifts to the recirc mode.
- The test document describes this actuation as an expected result of the test.
- The NRC resident inspector is in the control room observing the test.

The Nuclear Shift Supervisor has determined that this actuation should NOT be reported to the NRC in accordance with MLS05, Notifications/General Regulatory Reporting Requirements.
Which ONE of the following describes the reason the NSS did NOT report this actuation?

______ A. Actuations which shift the aligment of ventilation systems are not required to be reported.

______ B. The actuation was part of a preplanned evolution documented by the test procedure.

______ C. Actuations are only required to be reported if they are the result of human error.

______ D. The actuation was observed directly by the NRC resident who is obligated to make the report in place of the NSS.

Point Value: 1

Question 99.

Using the attached Emergency Action Level Definitions, SELECT the item below that lists ONLY conditions requiring entry into the Emergency Plan?

______ A. Observation of major damage to spent fuel
Water loss below fuel level in the Fuel Pool
Spent fuel damage resulting in a Refueling Floor area ventilation isolation

______ B. Observation of major damage to spent fuel
Water loss below fuel level in the Fuel Pool
New fuel bundle discovered missing during delivery.

______ C. Observation of major damage to spent fuel
New fuel bundle discovered missing during delivery.
Radiation levels of 300 mrem/hr from an LPRM string that is near the surface of the water

______ D. Water loss below fuel level in the Fuel Pool
Spent fuel damage resulting in a Refueling Floor area ventilation isolation
Radiation levels of 300 mrem/hr from an LPRM string that is near the surface of the water

Point Value: 1

Question 100.

The plant is operating at 90% power when an electrical grid transient occurs. The following plant conditions are noted after the electrical grid transient:

- Reactor vessel level 194 inches and decreasing
- Reactor vessel pressure 1038 psig and increasing
- Drywell pressure 1.2 psig and stable.
- Generator output 770 Mwe, CM tripped open, CF remains closed
- Turbine bypass valves are fully open
- Reactor Power is 107% and stable.
- Core flow 70% and stable.

The following annunciators are in alarm:
- 3D102 APRM simulated thermal power upscale
- 3D101 APRM simulated thermal power upscale trip

No protective functions have automatically occured and all other plant conditions are as they were before the transient.

Which of the following actions is required?

_____ A. Enter EOP 29.100.01 Sheet 1, RPV Control

_____ B. Enter 20.109.01, Turbine Trip

_____ C. Adjust the pressure regulator to reduce reactor pressure

_____ D. Immediately reduce power using Recirculation Flow

Point Value: 1

Emergency Action Level Definitions

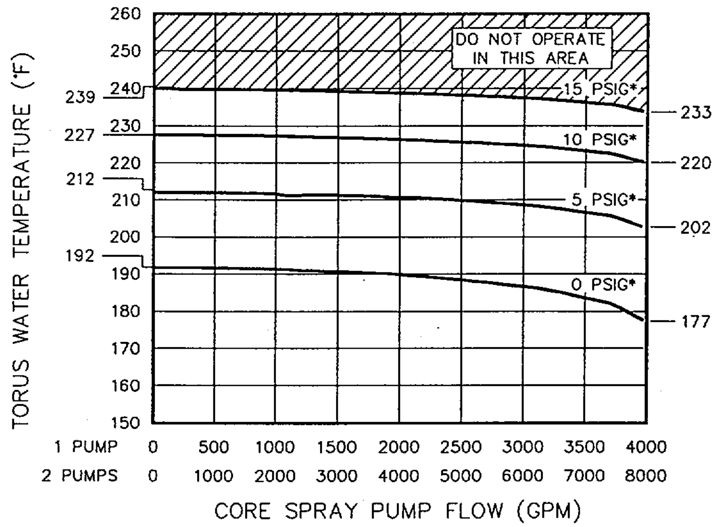
Unusual Event - Events are in process or have occurred that indicate a potential degradation of the level of safety of the plant. No release of radioactive material requiring offsite response or monitoring is expected.

Alert - Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases are expected to be limited to small fractions of the Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels.

Site Area Emergency - Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases are not expected to exceed Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels except at or near the site boundary.

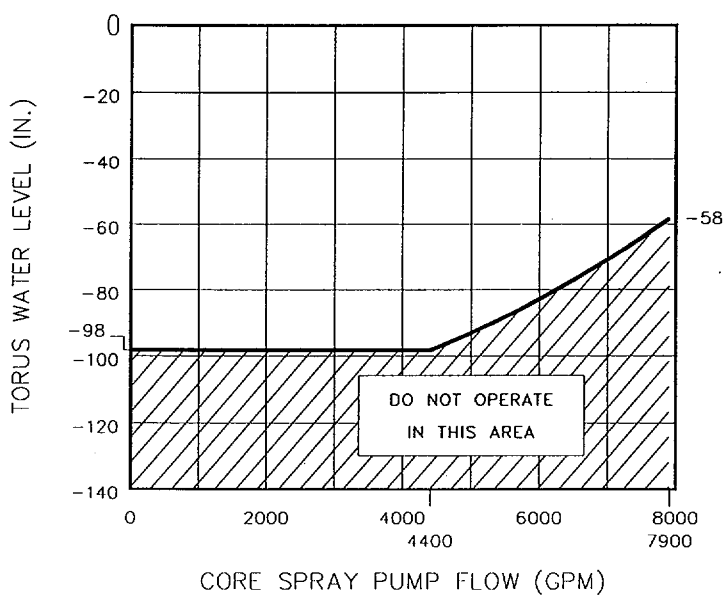
General Emergency - Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed Environmental Protection Agency (EPA) Protection Action Guidelines exposure levels offsite for more than the immediate site area.

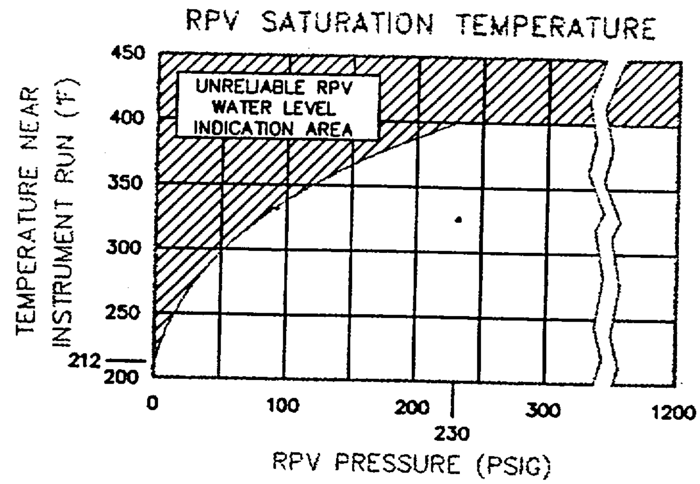
CS NPSH LIMIT



6 < −11 IN.

CS VORTEX LIMIT





RPV SATURATION TEMPERATURE
UNRELIABLE RPV WATER LEVEL INDICATION AREA
TEMPERATURE NEAR INSTRUMENT RUN (°F)
450
400
350
300
250
212
200
0
100
200
230
300
1200
RPV PRESSURE (PSIG)

3. For each of the following instruments, the instrument reads above the minimum indicated level associated with the highest temperature near an instrument reference leg vertical run:

A. Wide range level detectors (10 to 220 IN.)
(B21-N081A,B)
(B21-N091A,C)

HIGHEST REACTOR BUILDING RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
–	80	10
80	150	15
150	250	26
250	350	40

B. Wide range level detectors (10 to 220 IN.)
(B21-N081C,D)
(B21-N091B,D)

HIGHEST REACTOR BUILDING RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
–	167	10
167	250	20
250	350	36

C. Flood up level detector (160 to 560 IN.)
(B21-N027)

HIGHEST DRYWELL RUN TEMPERATURE (°F) BETWEEN LOW	HIGH	MINIMUM INDICATED LEVEL(IN.)
–	150	175
150	250	190
250	350	210
350	450	237
450	550	274

CAUTIONS

1 An RPV water level instrument may be used to determine RPV water level only when all the following conditions are satisfied for that instrument:

1. The temperatures near all the instrument runs are below the RPV saturation temperature.
2. For each of the instruments in the following table, the instrument reads above the minimum indicated level or the temperatures near all the instrument reference leg vertical runs are below the maximum run temperature.

INSTRUMENT	RANGE(IN.)	MAXIMUM REACTOR BUILDING RUN TEMPERATURE (°F)	MINIMUM INDICATED LEVEL(IN.)
Core Level Detector (B21–N085A)	–150 to 50	327	–142
Core Level Detector (B21–N085B)	–150 to 50	309	–134
Narrow Range Level Detectors (B21–N080A,B) (B21–N095A,C) (C32–N004A,C)	160 to 220	103	169
Narrow Range Level Detectors (B21–N080C,D) (B21–N095B,D) (C32–N004B,D)	160 to 220	273	165

PART 1

CODE I.D. NUMBER AT LEFT BY FILLING IN THE APPROPRIATE BOXES. IF A NUMBER IS LARGER THAN 5, FILL IN 5 PLUS THE DIGIT NEEDED TO ADD UP TO THE DESIRED NUMBER.

WRITE I.D. NUMBER HERE

EXAMPLE

PART 2

IMPORTANT

USE NO. 2 PENCIL ONLY

- MAKE *DARK* MARKS
- EXAMPLE: A B C D E
- ERASE **COMPLETELY** TO CHANGE

FEED THIS DIRECTION

NAME SRO ILO NRC Exam Fermi 2

SUBJECT SRO Key

DATE 2/4/00 HOUR

SCANTRON® FORM 2052

TEST RECORD

PART 1	PART 2	TOTAL

(T) (F) KEY

	Key		Key
1	C	51	D
2	D	52	B
3	B	53	C
4	A	54	B
5	A	55	B
6	C	56	A
7	A	57	C
8	C	58	C
9	A	59	A
10	D	60	D
11	C	61	C
12	B	62	D
13	C	63	B
14	B	64	A
15	C	65	B
16	A	66	D
17	B	67	A
18	B	68	B
19	D	69	A
20	C	70	D
21	B	71	A
22	B	72	A
23	C	73	D
24	A	74	A
25	D	75	D
26	C	76	A
27	C	77	D
28	C	78	B
29	D	79	B
30	A	80	C
31	C	81	D
32	A	82	C
33	D	83	C
34	D	84	D
35	C	85	A
36	B	86	B
37	C	87	B
38	A	88	D
39	C	89	D
40	D	90	D
41	C	91	C
42	C	92	D
43	C	93	D
44	D	94	C
45	A	95	B
46	D	96	D
47	B	97	B
48	D	98	B
49	D	99	A
50	B	100	A

Exam Answer Key

Senior Reactor Operator **EC-99-11-05**

9801 ILO

Q#	Ans.	Pts.	Question Number	References
1.	C	1	EQ-OP-315-0114-000-0008-006	23.139 Section 5.3
2.	D	1	EQ-OP-315-0131-000-0005-005	ST-OP-315-0031-001
3.	B	1	EQ-OP-315-0109-000-0006-005	20.106.01, CRD HYDRAULIC SYSTEM FAILURE
4.	A	1	EQ-OP-315-0105-000-0004-008	ST-OP-315-0028-001 ST-OP-315-0027-001
5.	B	1	EQ-OP-315-0158-000-0004-004	ST-OP-315-0058 23.321
6.	C	1	EQ-OP-315-0142-000-0004-006	ST-OP-315-0142-001
7.	A	1	EQ-OP-315-0105-000-0026-008	ST-OP-315-0005-001
8.	C	1	EQ-OP-315-0104-000-0011-006	23.138.01, REACTOR RECIRCULATION SYSTEM
9.	A	1	EQ-OP-315-0127-000-0001-004	ST-OP-315-0027-001
10.	D	1	EQ-OP-802-3004-000-0007-001	ST-OP-802-3004-001
11.	C	1	EQ-OP-802-3006-000-0027-003	29.ESP.02, Alternate Boron Injection, ST-OP-802-3006-001
12.	B	1	EQ-OP-315-0165-000-0019-005	ST-OP- 315-0065-001; 20.300.04, Loss of 4160V ESS Busses

Exam Answer Key

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9801 ILO

Q#	Ans.	Pts.	Question Number	References
13.	C	1	EQ-OP-315-0109-000-0002-002	ST-OP-315-0009-001
14.	B	1	EQ-OP-315-0141-000-0004-006	ST-OP-315-0041-001
15.	C	1	EQ-OP-315-0104-000-0006-006	ST-OP-315-0004-001
16.	A	1	EQ-OP-315-0139-000-0004-013	ST-OP-315-0039-001
17.	B	1	EQ-OP-315-0140-000-0003-001	ST-OP-315-0040-001
18.	B	1	EQ-OP-315-0122-000-0003-010	ST-OP-315-0022-001
19.	D	1	EQ-OP-315-0148-000-0004-008	ST-OP-315-0048
20.	C	1	EQ-OP-315-0127-000-0005-005	I-2155-10
21.	B	1	EQ-OP-315-0141-000-0012-004	ST-OP-315-0041-001
22.	B	1	EQ-OP-315-0108-000-0005-004	ST-OP-315-0008 ST-OP-315-0014
23.	C	1	EQ-OP-315-0141-000-0003-030	ST-OP-315-0141-001
24.	A	1	EQ-OP-315-0150-000-0003-005	ST-OP-315-0050-001, ST-OP-315-0027-001, ST-OP-315-0048-001

Exam Answer Key

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9801 ILO

Q#	Ans.	Pts.	Question Number	References
25.	D	1	EQ-OP-802-1001-000-0008-014	22.000.02, Plant Startup to 25% Power, ST-OP-315-0027-001
26.	C	1	EQ-OP-315-0121-000-0003-008	ST-OP-315-0021-001
27.	C	1	EQ-OP-315-0145-000-0010-006	ST-OP-0045-001
28.	C	1	EQ-OP-315-0128-000-0004-003	ST-OP-315-0028-001
29.	D	1	EQ-OP-315-0146-000-0004-017	ST-OP-315-0046-001
30.	A	1	EQ-OP-315-0146-000-0003-007	ST-OP-315-0046-001
31.	C	1	EQ-OP-802-3002-000-0005-004	29-100-01 SH 6
32.	A	1	EQ-OP-315-0116-000-0008-007	Technical Specifications 3.6.2.1
33.	D	1	EQ-OP-802-2002-000-0002-007	ST-OP-315-0010-001
34.	D	1	EQ-OP-315-0126-000-0002-003	ST-OP-315-0025-001
35.	C	1	EQ-OP-315-0123-000-0005-006	ST-OP-315-0123-001
36.	B	1	EQ-OP-315-0141-000-0012-003	ST-OP-315-0041-001

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Q#	Ans.	Pts.	Question Number	References
37.	C	1	EQ-OP-315-0165-000-0015-003	ST-OP-315-0058-001, ST-OP-315-0065-001
38.	A	1	EQ-OP-315-0143-000-0007-004	ST-OP-315-0043-001, 23,206, Reactor Core Isolation Cooling System
39.	C	1	EQ-OP-315-0157-000-0005-002	ST-OP-315-0057-001
40.	D	1	EQ-OP-315-0144-000-0002-006	ST-OP-315-0044-001, RSD DBD
41.	C	1	EQ-OP-315-0032-000-0108-001	20.125.01, Loss of Condenser Vacuum
42.	C	1	EQ-OP-315-0199-000-0004-004	ST-OP-315-0018-001
43.	C	1	EQ-OP-315-0029-000-0108-001	22.000.02, Plant Startup to 25% Power 23.137
44.	D	1	EQ-OP-315-0110-000-0003-003	ST-OP-315-0010-001
45.	A	1	EQ-OP-802-1001-000-0021-005	Technical Specification Basis 3.4.10
46.	D	1	EQ-OP-315-0048-000-0104-001	ST-OP-0048-001, 10CFR100 100.11.(1)
47.	B	1	EQ-OP-315-0010-000-0105-001	ST-OP-315-0090-001
48.	D	1	EQ-OP-315-0127-000-0004-012	ST-OP-315-0021-001 ST-OP-315-0010-001

Exam Answer Key

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9801 ILO

Q#	Ans.	Pts.	Question Number	References
49.	D	1	EQ-OP-802-2017-000-0007-003	20.000.19, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM
50.	B	1	EQ-OP-802-3002-000-0107-001	ST-OP-802-3002-001
51.	D	1	EQ-OP-802-2017-000-0002-004	ST-OP-315-0099-001
52.	B	1	EQ-OP-802-3002-000-0106-001	23.203, Low Pressure Core Spray EOP CS NPSH Limit Curve EOP CS Vortex Limit Curve
53.	C	1	EQ-OP-802-3003-000-0009-003	ST-OP-802-3003-001
54.	B	1	EQ-OP-802-3003-000-0122-001	20.ESP.03, Alternate Rod Insertion Methods
55.	B	1	EQ-OP-802-3004-000-0011-004	29.100.01, ST-OP-802-3004-001
56.	A	1	EQ-OP-802-3004-000-0110-001	ST-OP-802-3004-001
57.	C	1	EQ-OP-315-0167-000-0003-002	ST-OP-315-0067-001
58.	C	1	EQ-OP-315-0171-000-0004-007	ST-OP-315-0071-001
59.	A	1	EQ-OP-802-2011-000-0004-001	20.205.01, LOSS OF SHUTDOWN COOLING
60.	D	1	EQ-OP-315-0139-000-0003-023	ST-OP-315-0039-001

Exam Answer Key

Q#	Ans.	Pts.	Question Number	References
61.	C	1	EQ-OP-315-0150-000-0003-006	ST-OP-315-0050
62.	D	1	EQ-OP-315-0169-000-0004-008	ST-OP-315-0069-001
63.	B	1	EQ-OP-802-2016-000-0011-003	20.000.22, PLANT FIRES
64.	A	1	EQ-OP-315-0146-000-0003-008	ST-OP-315-0046-001
65.	B	1	EQ-OP-315-0104-000-0006-007	EQ-OP-315-0004-001
66.	D	1	EQ-OP-315-0155-000-0003-014	ST-OP-315-0055-001
67.	A	1	EQ-OP-802-2007-000-0001-007	ST-OP-802-2007-001
68.	B	1	EQ-OP-315-0173-000-0005-003	ST-OP-315-0073-001
69.	A	1	EQ-OP-315-0190-000-0005-003	Technical Specification Basis 3.9.1-1 ST-OP-315-0190-001
70.	D	1	EQ-OP-315-0190-000-0012-014	Technical Specification 3.9.9
71.	A	1	EQ-OP-804-0001-000-0003-021	Technical Specifications
72.	A	1	EQ-OP-802-4101-000-0014-003	MGA03

Exam Answer Key

Senior Reactor Operator **EC-99-11-05**

9801 ILO

Q#	Ans.	Pts.	Question Number	References
73.	D	1	EQ-OP-315-0128-000-0009-003	Technical Specifications and Technical Specification bases.
74.	A	1	EQ-OP-508-0001-000-0001-001	ST-GN-508-0000-001
75.	D	1	EQ-OP-315-0120-000-0005-004	23.406, PRIMARY CONTAINMENT NITROGEN INERTING AND PURGE SYSTEM
76.	A	1	EQ-OP-802-2013-000-0008-004	ST-OP-315-0062-0001 20.300.12 Loss of UPS
77.	D	1	EQ-OP-832-0001-000-0002-002	EP-101
78.	B	1	EQ-OP-315-0113-000-0010-001	ST-OP-315-0013-001
79.	B	1	EQ-OP-802-3002-000-0004-011	Emergency Operating Procedure General Caution #1, Part 3ST-OP-3002-001
80.	C	1	EQ-OP-315-0127-000-0003-004	ITS 3.3.1.1
81.	D	1	EQ-OP-315-0141-000-0009-007	20.000.04 Reactor shutdown procedure
82.	C	1	EQ-OP-315-0119-000-0005-005	ST-OP-315-0016-001
83.	C	1	EQ-OP-802-3004-000-0011-006	ST-OP-802-3004-001
84.	D	1	EQ-OP-802-3003-000-0024-004	ST-OP-802-3003-001

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85.	A	1	EQ-OP-802-3005-000-0011-004	ST-OP-315-3005-001
86.	B	1	EQ-OP-315-0198-000-0006-002	23.414, STEAM TUNNEL COOLING
87.	B	1	EQ-OP-802-3005-000-0001-003	29.100.01 SH 5
88.	D	1	EQ-OP-315-0116-000-0003-018	ST-OP-315-0016-001
89.	D	1	EQ-OP-315-0116-000-0009-003	St-OP-315-0016 Tech Spec 3.6.1.2
90.	D	1	EQ-OP-804-0001-000-0024-003	MOP03 Policies and Practices
91.	C	1	EQ-OP-802-4101-000-0025-004	MOP06
92.	D	1	EQ-OP-804-0001-000-0007-004	Technical Specifications
93.	D	1	EQ-OP-802-4101-000-0030-005	MOP13 Conduct of Refueling and Core Alterations
94.	C	1	EQ-OP-508-0001-000-0036-001	ST-GN-508-0000-001
95.	B	1	EQ-OP-802-4101-000-0034-003	MOP 02, Independent Verfication
96.	D	1	EQ-OP-802-3004-000-0001-008	ST-OP-3001-001

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97.	B	1	EQ-OP-802-3002-000-0004-012	ST-OP-802-3002-001
98.	B	1	EQ-OP-802-4101-000-0017-011	MLS05, Notifications/General Regulatory Reporting Requirements
99.	A	1	EQ-OP-832-0001-000-0002-001	EP-101, Classification of Emergencies
100.	A	1	EQ-OP-802-3003-000-0006-006	29.100.01 Sheet 1, RPV Control