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RECIP.NAME RECIPIENT AFFILIATION
PARRISH,J.V. Washington Public Power Supply System

SUBJECT: Informs that per telcon of 980416 between T Mckernon & M Westergren,arrangements were made for administration of licensing exam at Washington Nuclear Project 2 during week of 981102.Provide exam outlines by 980705.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

April 28, 1998

Mr. J. V. Parrish (Mail Drop 1023)
Chief Executive Officer
Washington Public Power Supply System
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: NRC INSPECTION 50-397/98-301

Dear Mr. Parrish:

In a telephone conversation on April 16, 1998, between Messrs. Tom McKernon, Chief Examiner, and Mark Westergren, Lead Instructor, arrangements were made for the administration of licensing examinations at Washington Nuclear Project 2, during the week of November 2, 1998.

As agreed during the telephone conversation, your staff will prepare the examinations based on the guidelines in Revision 8, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." The NRC regional office will discuss with your staff any changes that might be necessary before the examinations are administered.

To meet the above schedule, it will be necessary for your staff to furnish the examination outlines by July 5, 1998. The written examinations, operating tests, and the supporting reference materials identified in Attachment 2 of ES-201 will be due by September 3, 1998. Any delay in receiving the required examination and reference materials, or the submittal of inadequate or incomplete materials, may cause the examinations to be rescheduled. A validation of the final examination material will be conducted at your facility the week of October 12, 1998.

In order to conduct the requested written examinations and operating tests, it will be necessary for your staff to provide adequate space and accommodations in accordance with ES-402, and to make the simulation facility available on the dates noted above. In accordance with ES-302, your staff should retain the original simulator performance data (e.g., system pressures, temperatures, and levels) generated during the dynamic operating tests until the examination results are final.

Appendix E of NUREG-1021 contains a number of NRC policies and guidelines that will be in effect while the written examinations and operating tests are being administered.

To permit timely NRC review and evaluation, your staff should submit preliminary reactor operator and senior reactor operator license applications (Office of Management and Budget

(OMB) approval number 3150-0090), medical certifications (OMB approval number 3150-0024), and waiver requests (if any) (OMB approval number 3150-0090) at least 30 days before the first examination date. If the applications are not received at least 30 days before the examination date, a postponement may be necessary. Signed applications certifying that all training has been completed should be submitted at least 14 days before the first examination date.

This letter contains information collections that are subject to the *Paperwork Reduction Act of 1995* (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0101, which expires on April 30, 2000.

The public reporting burden for this collection of information is estimated to average 500 hours per response, including the time for reviewing instructions, gathering and maintaining the data needed, writing the examinations, and completing and reviewing the collection of information. Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0101), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

Thank you for your cooperation in this matter. Mr. Westergren has been advised of the policies and guidelines referenced in this letter. If you have any questions regarding the NRC's examination procedures and guidelines, please contact Mr. Tom McKernon at 817-860-8153 or myself at 817-860-8159.

Sincerely,

John L. Pellet

John L. Pellet, Chief
Operations Branch
Division of Reactor Safety

Docket No.: 50-397

cc:
Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, Washington 98504-3172

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

April 28, 1998

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Chief Executive Officer
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P.O. Box 968
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SUBJECT: NRC INSPECTION 50-397/98-301

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Sincerely,

John Pellet

John L. Pellet, Chief
Operations Branch
Division of Reactor Safety

Docket No.: 50-397

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....9904190226

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-397
License No.:	NPF-21
Report No.:	50-397/99-301
Licensee:	Washington Public Power Supply System
Facility:	Washington Nuclear Project-2
Location:	Richland, Washington
Dates:	March 8 to 11, 1999
Inspectors:	T. O. McKernon, Chief Examiner, Operations Branch H. F. Bundy, Senior Examiner, Operations Branch T. R. Meadows, Senior Examiner, Operations Branch R. Vogt-Lowell, Examiner, HOHB
Approved By:	John L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	Licensee Post Examination Comments and Analysis
Attachment 3:	Written Examinations and Answer Keys

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/99-301

NRC examiners evaluated the competency of 1 reactor operator and 10 senior operator applicants for issuance of operating licenses at the Washington Nuclear Project-2 facility. The licensee developed the initial license examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. NRC examiners reviewed, approved, and administered the examinations. The initial written examinations were administered to all 11 applicants on March 5, 1999, by facility proctors in accordance with instructions provided by the chief examiner. The NRC examiners administered the operating tests on March 8 to 11, 1999.

Operations

- The 11 initial license applicants passed the examination. Operators demonstrated good communications practices, peer checks, and crew briefings. The licensee developed good test material which was adequate for administration as submitted, with only one post-examination change identified (Sections O4.1, O4.2).

- Operators demonstrated weak key parameter monitoring related to reactor building differential pressure during the dynamic scenarios. Two of the four crews examined failed to recognize that reactor building differential pressure went positive and, thus, missed an emergency operating procedure entry condition (Section O4.2).

- The licensee submitted an acceptable examination for administration, requiring only enhancement suggestions. The final as-given examination met the requirements of NUREG-1021 and was good quality (Section O5.1.2).

Report Details

Summary of Plant Status

The plant operated at essentially 100 percent, except for nightly power reductions to support economic dispatch, for the duration of this inspection.

I. Operations

O4 Operator Knowledge and Performance

O4.1 Initial Written Examination

a. Inspection Scope

On March 5, 1999, the licensee proctored the administration of the written examination approved by the NRC to one individual who had applied for a reactor operator license and ten individuals who had applied for senior operator licenses. The licensee graded the written examinations and performed a post-examination question analysis, which was reviewed by the examiners.

b. Observations and Findings

The minimum passing score was 80 percent. All applicants passed the written examination. The reactor operator applicant scored a 89 percent. All senior operator license applicants passed with scores ranging from 80 to 97 percent. The average score overall was 89 percent.

The above grades reflected examination changes recommended by the licensee as a result of post-examination question analysis were incorporated. The examiners reviewed and accepted the recommendations based on the technical merits of each recommendation. As a result of this analysis, one question (Reactor Operator 67, Senior Reactor Operator 71) was changed to accept two answers because the wording in the stem of the question did not provide sufficient information to eliminate the second answer. The submitted comments and analysis are included as part of Attachment 2 to this report. The licensee analysis also identified 16 other questions for which 30 percent or more of the applicants gave an incorrect response. No generic knowledge area weaknesses were identified. The licensee reviewed the missed questions with the applicants to upgrade their knowledge in the appropriate areas.

c. Conclusions

All reactor operator and senior operator license applicants passed the written examination.

O4.2 Initial Operating Test

a. Inspection Scope

The examiners administered the various portions of the operating test to the 11 applicants on March 8-11, 1999. Each applicant participated in 2 or 3 dynamic simulator scenarios. Each applicant also received a walkthrough test, which consisted of either 10 or 5 system tasks, depending on application type, with 2 followup questions for each task. The applicants also received an operating test administrative portion consisting of tasks or questions related to 5 subjects in 4 administrative areas.

b. Observations and Findings

All applicants passed all sections of the operating test. The applicants performed well, with good communication practices, peer checks, and crew briefs. For example, during the dynamic scenarios, the applicants consistently used three-way communications, peer checking while operating control board components under normal plant conditions, and kept the crews well informed through structured briefings. However, some applicants exhibited slow control board awareness or failed to monitor key plant parameters. For example, during the dynamic scenarios, some applicants failed to recognize or monitor reactor building differential pressure. The examiners also observed the validation crew, during the preparatory week, failure to monitor this key parameter. The examiners noted that the only indication of reactor building differential pressure was on two recorders on a back panel. When reactor building differential pressure became positive, the operators were required to enter Emergency Operating Procedure PPM 5.3.1,"Secondary Containment Control." While the temporary loss-of-negative reactor building pressure did not adversely affect plant conditions, it did exhibit weakness on the part of some board operators in control board awareness.

The examiners characterized this finding as a generic weakness because the parameter was an entry condition into the emergency operating procedures, more than one operating crew failed to recognize the condition, prior inspections have also identified operator key parameter monitoring as an issue, and no annunciator or monitoring device was present on the front panels, as was the case for all other emergency operating procedures entry parameters (e.g., reactor pressure vessel level, drywell pressure and others). Further, reactor building differential pressure was not displayed on the safety parameter display system. While the applicants were knowledgeable of the parameter and those conditions under which it could become positive, the poorly human-engineered locations of the instrument makes it unacceptable for the function of rapidly determining the safety status in a transient or emergency condition. As-specified in NUREG-0696,"Functional Criteria for Emergency Response Facilities," the primary function of the safety parameter display system was to help operating personnel in the control room make quick assessments of plant safety status. The licensee entered this item, (i.e., bringing the reactor building differential pressure indication forward to the front panels), into their corrective action program and initiated Problem Event Report-299-0529 on March 15, 1999. The licensee had also implemented corrective actions related to operator key parameter monitoring and forwarded correspondence describing their corrective actions in Letter G02-99-053 dated March 16, 1999.

c. Conclusions

All applicants passed all sections of the operating test. Overall, the license applicants demonstrated good performance and use of good communication practices and peer checks. The examiners identified a generic weakness related to the monitoring of reactor building differential pressure.

O5 Operator Training and Qualification

O5.1 Initial Licensing Examination Development

The licensee developed the initial licensing examination in accordance with guidance provided in NUREG-1021.

O5.1.1 Examination Outline

a. Inspection Scope

The licensee submitted the initial examination outline on November 8, 1998. The examiners reviewed the submittal against the requirements of NUREG-1021.

b. Observations

The chief examiner provided enhancement suggestions related to examination integrity and responsiveness to NUREG-1021 requirements, which were incorporated by the licensee into the job performance measure and dynamic scenario outlines. For example, the job performance measures did not include a job task to change system state, such as raising level or lowering pressure, and the submitted scenario outlines contained some items that were later determined to overlap with the certification examination. These items were replaced. Additional enhancement suggestions were made to the scenarios in order to add balance to the malfunctions so that one particular applicant did not receive all the malfunctions.

c. Conclusions

The licensee submitted initial examination outlines that satisfied the requirements of NUREG-1021. Enhancement suggestions were made to the outlines and the revised outlines were submitted prior to the draft version of the examination submittal.

O5.1.2 Examination Package

a. Inspection Scope

The licensee submitted the initial examination package on January 14, 1999. The chief examiner reviewed the submittal against the requirements of NUREG-1021.

b. Observations and Findings

The licensee submitted 130 draft written examination questions, of which 70 were designated as common questions to both the reactor operator and senior operator examinations. The chief examiner provided comments or questions on 16 of the questions. Additionally, the licensee performed internal audits, which identified other questions for revision. The majority of changes made to the questions were changes to question stems and answers for clarification. Although failure to make the above changes would not have invalidated the written examinations, it would have degraded their discriminatory value. The written examinations were adequate for administration.

The licensee submitted two sets of the administrative test, one for the reactor operator applicant and one for the senior operator applicants. The examiners' review indicated that the administrative and walkthrough portions of the examination were acceptable for administration.

The examiners reviewed the scenarios and found them of good quality. Some enhancement suggestions were made for balancing malfunctions between applicants and to resolve any possible overlap with the certification examination.

As a result of the above comments, the licensee finalized and resubmitted the operating portion of the examination on February 26, 1999. The revised examination materials satisfied NUREG-1021 requirements and were of good quality.

c. Conclusion

The written and operating examinations were acceptable for administration. The licensee made enhancement suggestion changes to the examination. The final as-given examination met the requirements of NUREG-1021 and was considered good quality.

O5.1.3 Licensing Conditions

a. Inspection Scope

The chief examiner reviewed the final applications as submitted by the facility for the license applicants against the requirements of NUREG-1021, Interim Revision 8.

b. Observations and Findings

The chief examiner verified that the facility licensee properly identified the required five significant reactivity manipulations on the applications. The chief examiner also verified that the facility had properly documented these manipulations and that they were significant in accordance with NRC Information Notice 97-67 and Section ES-202, D.1.b(5), of NUREG 1021, Interim Revision 8.

c. Conclusions

The facility's program was adequate to ensure that initial license applicants satisfied the requirements for performance of significant reactivity manipulations

O5.2 Simulation Facility Performance

a. Inspection Scope

The examiners observed simulator performance with regard to fidelity during examination validation and administration.

b. Observations and Findings

The simulation facility supported examination administration well. However, on two occasions during the dynamic scenarios, the simulator abruptly stopped functioning and all annunciators and control board indications were lost. The examiners had the simulator reset at the point just before the malfunction and continued with the scenario to its conclusion without further problems. No simulator modeling problems were identified. Discussions with the simulator technician indicated that the error was typical of an equipment self-check trap in which some data was lost. The licensee had scheduled a hardware and software upgrade of the simulator for spring 1999. The upgraded system will eliminate such errors. A simulator fidelity report was not attached to this report since the malfunction was not a modeling or fidelity problem, the problem did not occur during the validation of the examination, and the simulator's computer hardware and software systems were being replaced immediately following the examination period.

c. Conclusions

The simulator supported examination administration well with two noted exceptions.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on March 11, 1999. The licensee acknowledged the findings presented.

The licensee did not identify as proprietary any information or materials examined during this inspection.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Coleman, Manager Regulatory Affairs
R. Guthrie, Initial License Class Coordinator
J. McDonald, Plant Production Manager
S. Oxenford, Operations Manager
B. Shaeffer, Manager Nuclear Training
G. Smith, Vice President Generation
P. Taylor, Operations Training Superintendent

NRC

J. Spets, Resident Inspector
B. Smalldridge, Resident Inspector Wolf Creek

ATTACHMENT 2

Facility Initial License Examination Comments and Analysis

WNP-2 Initial License Class - NRC Written Exam Analysis
for exam given on March 5, 1999

One RO and ten SRO applicants completed this exam. All applicants passed the exam with a grade of 80% or greater. Our exam analysis revealed 17 questions for which 30% or greater of the students gave an incorrect response. Of these 17, only 1 question was modified (SRO #71, RO #67). This modified question with recommendation and clarification is attached.

For the remaining 16 questions, training will review the questions with the students to upgrade their knowledge in the appropriate areas.

Question # 99066 72.7% Miss Rate	SRO-71 RO-67	A. B. C. D.	Correct Answer 7 1
Recommendation	Accept A and B Review with students.		
Clarifications	B		The stem of the question did not adequately exclude the assumption of cause that some students made when answering this question. The answer is dependent upon the type of failure that has caused the closure of the governor valves. The most probable cause is a trip of the feed pump turbine. It could also be assumed that the governor valves have closed without a turbine trip; for example, they may have been driven closed due to some type of failure in the Digital Feedwater system. If the cause of the governor valve closure is due to a feed pump turbine trip, the RPV level transient to the alarm setpoint (Level 4) will cause an automatic recirc runback to 30 Hz which will allow level restoration before reaching Level 3, the 15 Hz runback setpoint. Recirc will end up at 30 Hz. (Answer B) If the cause of the governor valve closure is assumed to be other than a trip of the feed pump turbine, recirc pumps will not run back at the level alarm setpoint. When level drops to Level 3, the recirc pumps will run back to 15 Hz. (Answer A) Although the assumptions for Answer A are less likely, both scenarios are credible and it is recommended that both Answer A and B be accepted as correct.

The plant is operating at 100% power when a failure causes the governor valves on RFW-DT-1A to go closed.

Assuming RFW-P-1B continues to operate, which of the following is correct for these conditions?

The recirc pumps ....

A. end up at 15 Hz.

B. end up at 30 Hz.

C. are off with Breakers RPT-3A and 3B open.

D. are off with Breakers RPT-3A/3B and 4A/4B open.

cx99066

The plant is operating at 100% power when a failure causes the governor valves on RFW-DT-1A to go closed.

Assuming RFW-P-1B continues to operate, which of the following is correct for these conditions?

The recirc pumps ....

A. end up at 15 Hz.

B. end up at 30 Hz.

C. are off with Breakers RPT-3A and 3B open.

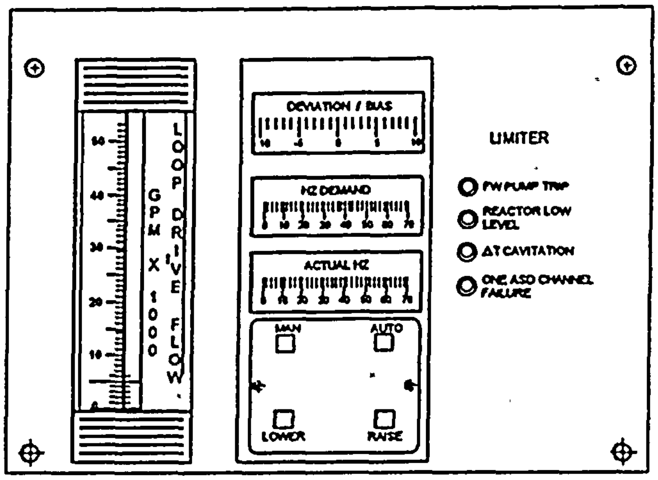
D. are off with Breakers RPT-3A/3B and 4A/4B open.

5. Uncleared alarms are always displayed in the lower left part of all display screens. They may be acknowledged/cleared by touching the screen at the location the alarm is listed. LO-9688-3 Alarms

II. CONTROL THEORY AND INTERLOCKS

A. Individual Loop Controller (located on P602 in the Control Room) LO-9677-2 Deviation/Bias

1. Deviation / Bias meter



When the Individual Loop Controller is in AUTO, this meter indicates the "Bias" applied to the Demand Signal from the Master Controller before it is passed to the ASD MEM units to control pump speed.

When the Individual Loop Controller is in MANUAL, this meter indicates the mismatch or "Deviation" between the Individual and Master Controllers.

2. Hz Demand meter LO-9677-3 Hz Demand

Indicates the value of the frequency demand signal from the Individual Loop Controller to the ASD MEM units for the associated pump.

3. Actual Hz meter LO-9677-4 Actual Hz

Indicates the actual frequency being applied to the Recirc pump motor. The ACTUAL HZ signal is taken from a frequency transducer which monitors the Recirc motor voltage. (Pump Speed = Actual Hz x 30)

4. FW PUMP TRIP Limiter light LO-9677-5 FW Pump Trip

Illuminates when a runback condition exists due to at least one RFPT tripped concurrent with RPV Low Level Alarm - Level 4 (31.5"). Extinguishes when ASD frequency is LE 30 Hz and the runback condition has cleared.

5. REACTOR LOW LEVEL Limiter light LO-9677-6 Rx Low Level

Illuminates when a runback condition exists due to RPV Level 3 (13"). Extinguishes when ASD frequency is LE 15 Hz and the runback condition has cleared.

6. ΔT CAVITATION Limiter light LO-9677-7 ΔT Cavitation

Illuminates when a runback condition exists due to a differential temperature between the RPV Steam Dome and the RRC suction less than 10.7° F for 10 min. Extinguishes when ASD frequency is LE 15 Hz and the runback condition has cleared.

Group 2 NRC Exam Analysis

Exam Bank #	SRO	RO	A	B	C	D	% Miss	Comment
99001	1	16		5	1	X	54.55	
99002	2	2	X				0.00	
99003	3	3			X		0.00	
99004	4	4	X				0.00	
98002	5			X			0.00	
99006	6	28		X			0.00	
99007	7	6		1		X	9.09	
99005	8	7		1		X	9.09	
99008	9	19			X	1	9.09	
99009	10	34	X				0.00	
99010	11	12	2			X	18.18	
99011	12	8	1	X			9.09	
99012	13	24		3	6	X	81.82	
98008	14					X	0.00	
98021	15		X				0.00	
98018	16			X			0.00	
99016	17					X	0.00	
99017	18	11			X		0.00	
99018	19				X	1	10.00	
99019	20			1	X	4	50.00	
99020	21			1	1	X	20.00	
99039	22	13	X				0.00	
99040	23			X			0.00	
99041	24		1			X	10.00	
99042	25		2	1	X	1	40.00	
98024	26					X	0.00	
98041	27				X	1	10.00	
99043	28	15	1	X			9.09	
98040	29					X	0.00	
99044	30	1				X	0.00	
99045	31	29	X				0.00	
99046	32		X			1	10.00	
99047	33	20		1	X		9.09	
99048	34	21	X	1		1	18.18	
99049	35	22				X	0.00	
99050	36	33			X		0.00	
98039	37			X			0.00	
99051	38	30		X	2		18.18	
99052	39				X		0.00	
99053	40	26				X	0.00	
99054	41	27			X		0.00	
99055	42	35	X				0.00	
99056	43	31	1			X	9.09	
99021	44	39	X			1	9.09	
99022	45	40			X	1	9.09	
98049	46					X	0.00	
99024	47	42		X			0.00	
99025	48	43				X	0.00	
[illegible]26	49	44	X		1		9.09	
[illegible]27	50	58				X	0.00	
99028	51	47		X		1	9.09	
99029	52	46	2	X			18.18	
99030	53	48				X	0.00	
99031	54				X		0.00	
99032	55		X		3		30.00	
99033	56			6	X		60.00	
99034	57		X				0.00	
99035	58	59			X		0.00	
99036	59	52		X			0.00	
99037	60	53		6		X	54.55	
99038	61	54	X				0.00	
99057	62	60		X			0.00	
99058	63	56	X				0.00	
99059	64	71	1	1	X		18.18	
99060	65	72	1	X		1	18.18	
99061	66	57				X	0.00	
99062	67		X			4	40.00	
99063	68	38	X		5		45.45	
99064	69		2	1	X		30.00	
99065	70	73			X		0.00	
99066	71	67	X	7	1		72.73	Accept A & B
99067	72			1	X	2	30.00	
99068	73					X	0.00	
99069	74			X			0.00	
99070	75		X				0.00	
99071	76	55		1	X		9.09	
99072	77	74		X	1		9.09	
99073	78	70	X				0.00	
99074	79	75				X	0.00	
99075	80		2	1	X		30.00	
99076	81			2	1	X	30.00	
99077	82	69		X			0.00	
99078	83	65	X	1		1	18.18	
98084	84				X		0.00	
98085	85		2		X	2	40.00	
98086	86				X	2	20.00	
98087	87					X	0.00	
98088	88		X			1	10.00	
98089	89					X	0.00	
98090	90		X		1		10.00	
99079	91			X			0.00	
98091	92				X		0.00	
99080	93		3		1	X	40.00	
99081	94		X	2			20.00	
98093	95		X	1			10.00	
98095	96		X			1	10.00	
98094	97	98	1			X	9.09	
99082	98				X	4	40.00	
98097	99		X				0.00	
98099	100			X			0.00	

Group 2 NRC Exam Analysis

98101		5		X			0.00	
99014		9			X		0.00	
99015		10	X				0.00	
99083		14	X	1			100.00	
99084		17				X	0.00	
98104		18	X				0.00	
98102		23				X	0.00	
99013		25	X				0.00	
99085		32		X			0.00	
99086		36				X	0.00	
98105		37			X		0.00	
99023		41				X	0.00	
98107		45			X		0.00	
98059		49				X	0.00	
98108		50		X			0.00	

98106		51			X		0.00	
99087		61		X			0.00	
99088		62	X		1		100.00	
99089		63			X		0.00	
99090		64				X	0.00	
98109		66				X	0.00	
98112		68				X	0.00	
99091		76		X			0.00	
98114		77	X				0.00	
98113		78		X			0.00	
99092		79	X				0.00	
99093		80			X		0.00	
99094		81				X	0.00	
99095		82		X			0.00	
99096		83				X	0.00	
98118		84			X		0.00	
98117		85	X				0.00	
99097		86			X		0.00	
99098		87				X	0.00	
98119		88				X	0.00	
99099		89				X	0.00	
99100		90		X			0.00	
98121		91		X			0.00	
98122		92		1	X		100.00	
99102		93		1		X	100.00	
98124		94			X		0.00	
99101		95	X				0.00	
98125		96		X			0.00	
98126		97	X				0.00	
98129		99		X			0.00	
98130		100	X				0.00	

ATTACHMENT 3

Initial Written Examinations and Answer Keys

Attachment 2.

Master Written Exams and answer key with annotated changes

SENIOR REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: KEY

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change:

1. a b c ~~d~~
2. ~~a~~ b c d
3. a b ~~c~~ d
4. ~~a~~ b c d
5. a ~~b~~ c d
6. a ~~b~~ c d
7. a b c ~~d~~
8. a b c ~~d~~
9. a b ~~c~~ d
10. ~~a~~ b c d
11. a b c ~~d~~
12. a ~~b~~ c d
13. a b c ~~d~~
14. a b c ~~d~~
15. ~~a~~ b c d
16. a ~~b~~ c d
17. a b c ~~d~~
18. a b ~~c~~ d
19. a b ~~c~~ d
20. a b ~~c~~ d
21. a b c ~~d~~
22. ~~a~~ b c d
23. a ~~b~~ c d
24. a b c ~~d~~
25. a b ~~c~~ d
26. a b c ~~d~~
27. a b ~~c~~ d
28. a ~~b~~ c d
29. a b c ~~d~~
30. a b c ~~d~~
31. ~~a~~ b c d
32. ~~a~~ b c d
33. a b ~~c~~ d
34. ~~a~~ b c d
35. a b c ~~d~~
36. a b ~~c~~ d
37. a ~~b~~ c d
38. a ~~b~~ c d
39. a b ~~c~~ d
40. a b c ~~d~~
41. a b ~~c~~ d
42. ~~a~~ b c d
43. a b c ~~d~~
44. ~~a~~ b c d
45. a b ~~c~~ d
46. a b c ~~d~~
47. a ~~b~~ c d
48. a b c ~~d~~
49. ~~a~~ b c d
50. a b c ~~d~~

SENIOR REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: KEY

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

51. a ~~b~~ c d
52. a ~~b~~ c d
53. a b c ~~d~~
54. a b ~~c~~ d
55. ~~a~~ b c d
56. a b ~~c~~ d
57. ~~a~~ b c d
58. a b ~~c~~ d
59. a ~~b~~ c d
60. a b c ~~d~~
61. ~~a~~ b c d
62. a ~~b~~ c d
63. ~~a~~ b c d
64. a b ~~c~~ d
65. a ~~b~~ c d
66. a b c ~~d~~
67. ~~a~~ b c d
68. ~~a~~ b c d
69. a b ~~c~~ d
70. a b ~~c~~ d
a or b 71. ~~a~~ ~~b~~ c d
72. a b ~~c~~ d
73. a b c ~~d~~
74. a ~~b~~ c d
75. ~~a~~ b c d
76. a b ~~c~~ d
77. a ~~b~~ c d
78. ~~a~~ b c d
79. a b c ~~d~~
80. a b ~~c~~ d
81. a b c ~~d~~
82. a ~~b~~ c d
83. ~~a~~ b c d
84. a b ~~c~~ d
85. a b ~~c~~ d
86. a b ~~c~~ d
87. a b c ~~d~~
88. ~~a~~ b c d
89. a b c ~~d~~
90. ~~a~~ b c d
91. a ~~b~~ c d
92. a b ~~c~~ d
93. a b c ~~d~~
94. ~~a~~ b c d
95. ~~a~~ b c d
96. ~~a~~ b c d
97. a b c ~~d~~
98. a b ~~c~~ d
99. ~~a~~ b c d
100. a ~~b~~ c d

REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: KEY

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

1. a b c ~~d~~
2. ~~a~~ b c d
3. a b ~~c~~ d
4. ~~a~~ b c d
5. a ~~b~~ c d
6. a b c ~~d~~
7. a b c ~~d~~
8. a ~~b~~ c d
9. a b ~~c~~ d
10. ~~a~~ b c d
11. a b ~~c~~ d
12. a b c ~~d~~
13. ~~a~~ b c d
14. ~~a~~ b c d
15. a ~~b~~ c d
16. a b c ~~d~~
17. a b c ~~d~~
18. ~~a~~ b c d
19. a b ~~c~~ d
20. a b ~~c~~ d
21. ~~a~~ b c d
22. a b c ~~d~~
23. a b c ~~d~~
24. a b c ~~d~~
25. ~~a~~ b c d
26. a b c ~~d~~
27. a b ~~c~~ d
28. a ~~b~~ c d
29. ~~a~~ b c d
30. a ~~b~~ c d
31. a b c ~~d~~
32. a ~~b~~ c d
33. a b ~~c~~ d
34. ~~a~~ b c d
35. ~~a~~ b c d
36. a b c ~~d~~
37. a b ~~c~~ d
38. ~~a~~ b c d
39. ~~a~~ b c d
40. a b ~~c~~ d
41. a b c ~~d~~
42. a ~~b~~ c d
43. a b c ~~d~~
44. ~~a~~ b c d
45. a b ~~c~~ d
46. a ~~b~~ c d
47. a ~~b~~ c d
48. a b c ~~d~~
49. a b c ~~d~~
50. a ~~b~~ c d

REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: KEY

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

	51.	a	b	~~c~~	d	76.	a	~~b~~	c	d
	52.	a	~~b~~	c	d	77.	~~a~~	b	c	d
	53.	a	b	c	~~d~~	78.	a	~~b~~	c	d
	54.	~~a~~	b	c	d	79.	~~a~~	b	c	d
	55.	a	b	~~c~~	d	80.	a	b	~~c~~	d
	56.	~~a~~	b	c	d	81.	a	b	c	~~d~~
	57.	a	b	c	~~d~~	82.	a	~~b~~	c	d
	58.	a	b	c	~~d~~	83.	a	b	c	~~d~~
	59.	a	b	~~c~~	d	84.	a	b	~~c~~	d
	60.	a	~~b~~	c	d	85.	~~a~~	b	c	d
	61.	a	~~b~~	c	d	86.	a	b	~~c~~	d
	62.	~~a~~	b	c	d	87.	a	b	c	~~d~~
	63.	a	b	~~c~~	d	88.	a	b	c	~~d~~
	64.	a	b	c	~~d~~	89.	a	b	c	~~d~~
	65.	~~a~~	b	c	d	90.	a	~~b~~	c	d
	66.	a	b	c	~~d~~	91.	a	~~b~~	c	d
a or b	67.	~~a~~	~~b~~	c	d	92.	a	b	~~c~~	d
	68.	a	b	c	~~d~~	93.	a	b	c	~~d~~
	69.	a	~~b~~	c	d	94.	a	b	~~c~~	d
	70.	~~a~~	b	c	d	95.	~~a~~	b	c	d
	71.	a	b	~~c~~	d	96.	a	~~b~~	c	d
	72.	a	~~b~~	c	d	97.	~~a~~	b	c	d
	73.	a	b	~~c~~	d	98.	a	b	c	~~d~~
	74.	a	~~b~~	c	d	99.	a	~~b~~	c	d
	75.	a	b	c	~~d~~	100.	~~a~~	b	c	d

U.S. Nuclear Regulatory Commission
Site Specific
Written Examination

Applicant Information

Name:	Region IV
Date:	WNP-2
SRO Written Exam	GE-BWR 5
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 four 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	________	Points
Applicant's Score	________	Points
Applicant's Grade	________	Percent

WRITTEN EXAMINATION GUIDELINES

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

2. To pass the examination, you must achieve a grade of 80.00 percent or greater. Every question is worth one point.

3. The time limit for completing the examination is four hours.

4. You may bring pens and calculators into the examination room. Use only black ink to ensure legible copies.

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

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If the intent of a question is unclear, ask questions of the NRC examiner or the designated facility instructor only.

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

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. After you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

SENIOR REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: ____________________

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

1. a b c d	26. a b c d
2. a b c d	27. a b c d
3. a b c d	28. a b c d
4. a b c d	29. a b c d
5. a b c d	30. a b c d
6. a b c d	31. a b c d
7. a b c d	32. a b c d
8. a b c d	33. a b c d
9. a b c d	34. a b c d
10. a b c d	35. a b c d
11. a b c d	36. a b c d
12. a b c d	37. a b c d
13. a b c d	38. a b c d
14. a b c d	39. a b c d
15. a b c d	40. a b c d
16. a b c d	41. a b c d
17. a b c d	42. a b c d
18. a b c d	43. a b c d
19. a b c d	44. a b c d
20. a b c d	45. a b c d
21. a b c d	46. a b c d
22. a b c d	47. a b c d
23. a b c d	48. a b c d
24. a b c d	49. a b c d
25. a b c d	50. a b c d

SENIOR REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice)   NAME: ____________________

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

51.	a	b	c	d	76.	a	b	c	d
52.	a	b	c	d	77.	a	b	c	d
53.	a	b	c	d	78.	a	b	c	d
54.	a	b	c	d	79.	a	b	c	d
55.	a	b	c	d	80.	a	b	c	d
56.	a	b	c	d	81.	a	b	c	d
57.	a	b	c	d	82.	a	b	c	d
58.	a	b	c	d	83.	a	b	c	d
59.	a	b	c	d	84.	a	b	c	d
60.	a	b	c	d	85.	a	b	c	d
61.	a	b	c	d	86.	a	b	c	d
62.	a	b	c	d	87.	a	b	c	d
63.	a	b	c	d	88.	a	b	c	d
64.	a	b	c	d	89.	a	b	c	d
65.	a	b	c	d	90.	a	b	c	d
66.	a	b	c	d	91.	a	b	c	d
67.	a	b	c	d	92.	a	b	c	d
68.	a	b	c	d	93.	a	b	c	d
69.	a	b	c	d	94.	a	b	c	d
70.	a	b	c	d	95.	a	b	c	d
71.	a	b	c	d	96.	a	b	c	d
72.	a	b	c	d	97.	a	b	c	d
73.	a	b	c	d	98.	a	b	c	d
74.	a	b	c	d	99.	a	b	c	d
75.	a	b	c	d	100.	a	b	c	d

The plant is operating at 100% power when the CRO receives a 1/2 scram on RPS-A and several alarms on Board C.

Which of the following caused these indications?

A trip of Bkr........

A. MC-7C

B. 7-71

C. MC-7B

D. 7-73

The reactor has just been scrammed.

Which of the following conditions ensures the reactor will remain shutdown under all conditions?

All control rods...

A. fully inserted except one at position 46.

B. fully inserted except one at position 04 and one at position 08.

C. fully inserted except two and 900 gal. of boron solution injected.

D. did NOT fully insert and 1500 gal of boron solution injected.

The reactor is operating at 99% power when an APRM hi flux scram occurs.

Which of the following caused the scram?

A. Trip of both Reactor Recirculation Pumps.

B. Lockout on SM-7

C. GV closure due to a DEH computer failure.

D. Loss of RPS B

The plant was operating at 97% power when the reactor scrammed. After conditions have stabilized the CRO notes both recirc pumps are tripped with CB-3A and 3B open (CB4A and 4B closed).

Which of the following signals caused the scram?

A. Reactor level -61 inches.

B. Reactor pressure 1112 psig.

C. Main Turbine Trip.

D. Drywell pressure 1.75 psig.

The reactor was operating at 100% power when an earthquake occurred, causing several drywell downcomers to shear off above the level of the suppression pool and a large LOCA.

Which of the following is correct?

A. Wetwell pressure will exceed the SRV Tail Pipe Level Limit (SRVTPLL) and exceed code allowable stresses on the tail pipe, supports, quenchers, or quencher supports.

B. The pressure suppression function has been bypassed; which will cause containment pressure to exceed the Pressure Suppression Pressure (PSP) and the Primary Containment Pressure Limit.

C. Drywell pressure will exceed the Drywell Spray Initiation Limit, causing a drywell-wetwell interface failure if drywell sprays are initiated.

D. Wetwell pressure will exceed the Heat Capacity Temperature Limit, causing containment pressure to exceed the Primary Containment Pressure Limit during a reactor blowdown.

ex99006

Why is it desirable to use ADS SRVs when Emergency Depressurizing?

A. They are designed to handle the dynamic load of 7 SRVs open simultaneously.

B. Their use evenly distributes the heat load around the wetwell.

C. Limits the local stress on the suppression pool floor.

D. Equalizes the stress on the drywell floor/downcomer interface.

A rod withdrawal is underway for a startup. IRMs are all selected to range 1, with a reactor period of 115 seconds. The operator selects the third control rod in Rod Worth Minimizer group 4 to move. After the rod has been moved, a reactor period of 24 seconds is noted.

What action is required with these conditions?

A. Stop control rod withdrawal and notify the SNE.

B. Insert control rods until period is GT 60 sec., and wait until power is in the heating range.

C. Manually scram the reactor.

D. Insert control rods until the reactor is subcritical and notify the CRS and SNE.

The plant is operating at 100% power. CRD-V-102, Overpiston area isolation, has been left closed following maintenance of the HCU for control rod 30-31. The plant then scrams.

Which of the following is correct based on these conditions?

A. The outer tube could rupture and cause a small Loss of Coolant Accident.

B. The collet piston could be damaged to the point the control rod will not stay at position 00 following the scram.

C. The drive seal could be degraded to the point of subsequent slow insert and withdraw speeds.

D. The drive could be damaged and cause the rod to become stuck resulting in not all rods inserting following the scram.

The reactor is operating at 99% power when a fire breaks out in the control room that requires an immediate control room evacuation.

Which of the following actions is required prior to leaving the control room?

A. Manually scram the reactor with all four manual scram pushbuttons.
Lock the mode switch in SHUTDOWN.
Transfer RPV level control to RFW-FCV-10A/B
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.

B. Manually scram the reactor by placing the mode switch in SHUTDOWN.
Transfer RPV level control to RFW-FCV-10A/B
Request security at SAS place 5 security doors in the unlock position.
Announce on the PA and radio; the scram, evacuation and an EO to local start DG2

C. Manually scram the reactor by placing the mode switch in SHUTDOWN.
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.
Request security at SAS place 5 security doors in the unlock position.
Announce on the PA and radio; the scram, evacuation and an EO to local start DG2

D. Manually scram the reactor with all four manual scram pushbuttons.
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.
Verify all rods full in.
Ensure all APRMs are downscale.

The reactor is in MODE 5. The following conditions exist:

SRM-A	OOS
SRM-B	2.8 counts per second (S/N 9.3/1)
SRM-C	2.5 counts per second (S/N 10.7/1)
SRM-D	OOS
MODE SWITCH	REFUEL
Reactor Coolant Temperature	108°F
Reactor Head and Internals	Removed

It is desired to start moving irradiated fuel from the vessel for refueling at this time.

Which of the following describes the reason why or why not irradiated fuel movement would be allowed to start at this time?

Irradiated fuel movement .....

A. may not start at this time, there are no operable SRMs.

B. may not start at this time, reactor coolant temperature is greater than 100°F.

C. may start in either the quadrant with the B or C SRM.

D. may start in any quadrant in the reactor vessel.

QUESTION # 11

3-5-99

ex99010

The plant is operating at 100% power. The following conditions exist:

RHR-P-2B	In operation on minimum flow
Drywell pressure	Increasing due to small steam leak
RRC-P-1B	Motor cooler leak
CRD 30-31	Leaking drive water connection

The CRO then notes suppression pool level increasing.

What is the reason for the suppression pool level increase?

A. RHR-P-2B in operation with minimum flow open.

B. Motor Cooler leak on RRC-P-1B.

C. Leaking drive water connection on CRD 30-31.

D. Drywell pressure increase from the small steam leak.

The reactor was operating at 98%power when an error during the performance of a surveillance caused an MSIV isolation and reactor scram. The following conditions exist:

Reactor level	15 inches
Reactor pressure	1087 psig
Drywell pressure	1.63 psig
Suppression pool temp.	86°F

Which of the following Emergency Operating Procedures should be entered:

A. PPM 5.2.1 Primary Containment Control, based on 86°F in the suppression pool.

B. PPM 5.1.1 RPV Control, based on 1087 psig in the reactor.

C. PPM 5.2.1 Primary Containment Control, based on 1.63 psig in the drywell.

D. PPM 5.1.1 RPV Control, based on 15 inches in the reactor.

The EOPs direct that the reactor be scrammed before wetwell (WW) temperature reaches a value which is equivalent to the Boron Injection Initiation Temperature (BIIT).

The BIIT is defined to be the greater temperature which results from either the WW temperature at which Technical Specifications require a reactor scram or the highest WW temperature at which initiation of SLC will result in....

A. injection of the Hot Shutdown Boron Weight before the WW exceeds pump vortex limits.

B. injection of the Cold Shutdown Boron Weight before the WW heats to the PSP limit.

C. injection of the Cold Shutdown Boron Weight before the WW exceeds the PCPL.

D. injection of the Hot Shutdown Boron Weight before the WW heats to the HCTL.

A 100% power isolated ATWS has occurred coincident with a leak in the suppression pool. Suppression pool level is 14' and going down.

Which of the following is a possible result of this condition?

A. A large LOCA would exceed the Heat Capacity Temperature Limit resulting in the failure of the wetwell/drywell interface.

B. SRV discharge would result in exceeding code allowable stresses in the tail pipe, tail pipe supports, quenchers, or quencher supports.

C. A large LOCA would result in exceeding the SRV Tail Pipe Level limit and exceed the Primary Containment Pressure Limit.

D. SRV discharge would bypass the pressure suppression function and cause Primary Containment pressure to exceed PCPL Primary Containment Pressure Limit.

During an ATWS condition with the reactor power greater than 5% or unknown, direction is given to lower reactor water level to -65 inches to -192 inches.

Which of the following describes the reason for reducing reactor level to at least -65 inches?

Feedwater heating in the steam space .....

A. reduces core inlet subcooling and stabilizes reactor power oscillations.

B. reduces core inlet subcooling and reduces core void production.

C. increases core inlet subcooling and increases core void production.

D. increases core inlet subcooling and stabilizes reactor power oscillations.

PPM 5.5.8 Alternate Boron injection directs that RCIC-V-19 (minimum flow) be closed and the supply breaker opened.

Which of the following describes the reason for the above note?

A. prevent SLC from being pumped to the Condensate Storage Tanks.

B. prevent SLC from being pumped to the suppression pool.

C. allows RCIC to develop the discharge pressure needed to discharge to the reactor.

D. allows RCIC to be operated at speeds lower than 2100 rpm without damage.

PPM 5.4.1, Radioactivity Release Control directs that "If turbine building HVAC system is shutdown, THEN restart turbine building HVAC".

Which of the following is the basis for this action?

A. Results in the radioactivity being discharged as a ground level release to limit the dispersion of the radioactivity.

B. Results in positive pressure inside the turbine building to limit the intrusion of radioactivity from the reactor building.

C. Provides for recirculation of the turbine building atmosphere with a reduction in the amount of radioactivity released.

D. Assures that any radioactivity in the turbine building is discharged through an elevated and monitored release point.

A DBA LOCA has occurred from 100% power. The following plant conditions exist:

Drywell Hydrogen	5.8%
Drywell Oxygen	4.2%
Core damage	Estimated at 15%
Drywell pressure	4 psig and down slow

Which of the following is correct for these conditions?

A. Calibrate the H202 analyzers for the drywell.

B. Secure CAC taking suction on the drywell.

C. Initiate CAC with suction from the drywell.

D. Emergency depressurize the reactor with 7 ADS SRVs.

QUESTION # 19 3-5-99

ex99018

The plant is operating at 99% power when a fault in the switch yard causes a main turbine generator trip. Breaker S-3 fails to auto close. TR-B is tagged out due to a failure at the Benton Sub-Station.

Which of the following describes when and how SM-8 is re-energized?

A. 10 seconds following the turbine trip, when breaker 8-DG2closes.

B. 20 seconds following the turbine trip, when breaker 8-DG2 closes.

C. 10 seconds following the turbine trip, when breaker DG2-8 closes.

D. 20 seconds following the turbine trip, when breaker DG2-8 closes.

The reactor was operating at 98% power when a scram occurred. The CRO noted the following conditions:

Reactor pressure	935 psig, down fast
Reactor level	+ 3 inches, down slow
Bypass Valves	100% open
Generator load	572 MWe, down fast
Turbine throttle valves	Closed
Governor valves	Closed

Which of the following is correct concerning subsequent Bypass Valves response?

The Bypass Valves will.......

A. stay open until feedwater returns reactor level to GT +13 inches.

B. stay open until turbine throttle valves close.

C. close when generator load is less than 25%.

D. close after a 20 second time delay following the turbine trip.

The plant is operating at 97% power when the CRO notes reactor level decrease about 3 inches and returns to normal level.

Which of the following is correct concerning these indications?

A. A failed feed flow transmitter downscale caused reactor level to decrease from a steam flow/feed flow mismatch, which FWLC sensed and changed to single element control.

B. The selected level instrument variable leg fails downscale, causing a steam flow/feed flow mismatch, which FWLC sensed and selected the alternate level transmitter input.

C. The selected level instrument reference leg fails downscale, causing a steam flow/feed flow mismatch, which FWLC sensed and selected the alternate level transmitter input.

D. A failed steam flow transmitter downscale caused reactor level to decrease from a steam flow/feed flow mismatch, which FWLC sensed and changed to single element control.

The first step in PPM 5.1.2, RPV Control - ATWS is to "Inhibit ADS"

Which of the following is the basis for this action?

A. This action prevents unwanted depressurization of the RPV which could result in uncontrolled injection from unborated water causing a power excursion.

B. The ADS system is not analyzed for ATWS conditions, more controlled pressure reductions are performed manually when conditions in the EOPs direct it.

C. This action prevents unwanted depressurization of the RPV which could result in uncontrolled reactivity addition due to addition of voids in the core.

D. The ADS system is not analyzed for ATWS conditions. Auto actuation at the ADS setpoint would be premature for ATWS conditions.

The plant was operating at rated conditions when an ATWS occurred.
Several SRVs lifted and increased Suppression Pool temperature.

The Control Room Supervisor directed the initiation of Standby Liquid Control.

Level in the reactor was lowered to reduce power production.

The crew is now inserting rods by driving and scramming.

Under which one of the following conditions would the termination of Standby Liquid Control be allowed?

A. The Reactor Engineer says that subcriticality can be guaranteed to 200°F

B. All rods are at position 00 except for control rod 34-35 is at position 48.

C. All rods are at position 02 except three in different quadrants at position 04.

D. The Cold Shutdown Boron Weight of SLC has been injected.

The reactor was at 100% power when a scram occurred. All systems operated as designed. After the scram is reset, all RPS Group white lights are on, the CRO notes that the scram discharge volume vents and drains did not open and the scram accumulators did not recharge.

Which of the following initiated the scram?

A. Drywell pressure peaked at 1.9 psig.

B. APRM power peaked at 120 percent.

C. Reactor level dropped to -30 inches.

D. Reactor pressure peaked at 1138 psig.

D

Following a major plant transient, the operating crew observes RPV water level at -175 inches and down slow. At this point, the CRS directs the CRO to emergency depressurize, after determining that ECCS pumps capable of restoring water level following emergency depressurization are running and lined up for injection.

Which of the following describes the CRS' direction?

A. This is the correct decision.
Adequate core cooling exists so long as RPV water level remains above -205 in.

B. This is NOT the correct decision.
Emergency depressurization should have been directed when RPV level was at TAF.

C. This is the correct decision.
Adequate core cooling exists so long as RPV water level remains above -192 in.

D. This is NOT the correct decision.
Emergency depressurization should be directed at -192 in.

QUESTION # 26 3-5-99

ex98024

The plant is in an ATWS condition with the following conditions:

Reactor pressure 827 psig
Reactor level -122 inches
SLC tank level 1107 gal.
Suppression pool temperature 126°F

Which of the following is the correct action?

A. Reduce reactor level to -161 inches to -192 inches.

B. Stop both SLC pumps.

C. Immediately Emergency Depressurize the reactor.

D. Start a cooldown to cold conditions at less than 100°F/hr.

ex98041

The plant was operating at 77% power on the 104% rod line when RRC-P-1B trips. Loop A drive flow is 16,000 gpm.

Which of the following describes the correct action?

A. Trip RRC-P-1A.

B. Reduce flow to the 15 Hz value on RRC-P-1A.

C. Manually scram the reactor.

D. Drive control rods until less than the 100% rod line.

The plant is operating at 100% power when an unisolable vacuum leak develops that is greater than the capacity of Off Gas. A reactor scram then occurs.

Assuming no operator action, which of the following caused the scram?

A. High Reactor Pressure

B. Main Turbine Trip

C. Low reactor level

D. MSIV Closure

ex98040

The plant is shutdown. The main turbine is on the turning gear and RCC is operating normally. RHR has been lined up for suppression pool cooling and RCIC is running for level control, when S2-1 is de-energized.

Which of the following describes the effect on plant operation?

A. CN-V-65 auto closes and isolates the N2 supply to the containment.

B. TO-P-BOP main turbine bearing oil pump trips and trips the MT turning gear.

C. RCC-V-6 auto closes and isolates the non-drywell RCC loads.

D. RCIC will continue to operate and provide level control.

Late in core life, fuel burnup and control rod density cause the first few inches of control rod travel to add much less negative reactivity during a scram.

Which of the following is designed the help mitigate this effect?

A. Reactor Scram from low EH Fluid Header Pressure

B. Reactor Scram from Turbine Throttle Valve Position

C. RRC Pump Runback from a Reactor level 3

D. RRC Pump Trip from a Main Turbine Trip

The plant is operating at 100% power with reactor level at 53 inches and up fast.

Which of the following describes the expected plant response and the reason for that response?

A. The main turbine trips to prevent damage to the Main Turbine blading

B. MSIV Isolation to prevent pitting of the MSIVs

C. The main turbine trips to prevent seat damage to the Main Turbine Throttle valves

D. The RFP turbines trip to prevent overloading of the RFP Moisture Pre-separator

An EOP entry has been made following a reactor scram and steam leak in the drywell. The first step in the EOPs for controlling Drywell temperature is to maintain temperature with "available drywell cooling".

Which of the following describes the reason for this direction?

A. This action assures that the normal method of temperature control is attempted in advance of more complex actions.

B. This action assumes normal cooling is not functional and to use whatever cooling is "available" under the given plant conditions.

C. Other means to control temperature such as containment spray are not available until a LOCA signal has been received.

D. This direction is given as an initial action since drywell cooling equipment will load shed if conditions degrade, resulting in a LOCA signal.

The plant is operating at 100% power. The following conditions exist:

Unidentified Leakage	4 gpm and stable
RCC-V-48 - surge tank makeup	open
FDR-P-4A sump pump in FDR-R-3	running

Which of the following would result in all of the above indications?

A. Packing leak on RHR-V-4A - RHR-P-2A Suction

B. RCC Heat Exchanger RCC-HX-1A leak (RB 548')

C. Drywell Cooler CRA-FC-2A leak

D. Packing leak from RRC-V-67B - RRC-P-2B discharge

The plant is operating at 100% power with EDR-P-5A running more than normal. Steam is coming out of EDR-R-5.

Which of the following problems is the cause of these conditions?

A. A scram valve seat leak.

B. A suction line leak on RRC-P-1A.

C. A Containment Nitrogen leak on the supply to MS-RV-4A.

D. A packing leak on RWCU-V-4, outboard suction isolation.

The plant is operating at 100% reactor power with maintenance underway that causes an inadvertent SLC system initiation.

What is the expected plant response?

A. RHR-V-60A and 60B, RHR heat exchanger sample isolation valves close.

B. RCC-V-5, 104, 40, and 21, RCC Supply and Return Drywell isolations close

C. All Primary Containment Recirculation Fans trip.

D. RWCU-V-4, Reactor Water Cleanup outboard containment isolation closes.

The plant has been in MODE 4 for two hours following a 9 month run at 100% power. RRC-P-1A is in operation with reactor level at 36 inches. RHR-P-2B is in operation in Shutdown Cooling, when it is inadvertently tripped by the CRO.

Which of the following describes the required actions?

A. Start RRC-P-1B and operate both RRC pumps at 15hz

B. Immediately raise RPV level to 60 - 80" to provide natural circulation.

C. Re-establish shutdown cooling as soon as possible and maintain level at 36 inches.

D. Perform Alternate Shutdown cooling using a HPCS or LPCS pump.

D

The plant is operating at 96% power when the operating CRD pump trips.

Which of the following describes a reason a CRD pump should be restored as soon as possible?

A. A loss of drive water pressure causes excessive rod speed during scrams.

B. A loss of drive water pressure causes rod drive temperatures to increase.

C. CRD provides cooling water for RRC Pump motor cooling.

D. CRD provides cooling water for RWCU motor cooling.

ex99051

A steam leak has developed inside primary containment, causing drywell pressure and temperature to increase. Both parameters are within the DSIL Curve and are trending up. Drywell temperature is 275°F. The CRS has given the direction to spray the drywell.

Which of the following describes the reason for spraying the drywell at this time?

A. Ensure that 95% of the non-condensable gases are transferred to the Wetwell.

B. Prevent exceeding the ADS design temperature.

C. Prevent possible damage to the safety relief valve (SRV) operators.

D. Reduce drywell temperature before the structural design temperature limits are reached.

cx99052

The plant is operating at rated power when several alarms are received indicating a RWCU leak. CRO2 reports the following temperatures from the LD monitoring panels:

RWCU Pump 1A Room,	LD-TE-3A	325°F
RWCU Pump 1B Room,	LD-TE-3C	180°F
RWCU Pipe Routing Area,	LD-TE-24C	192°F

Which of the following describes the correct action(s)?

A. Shutdown the reactor per PPM 3.2.1 and close the MSIVs.

B. Scram the reactor per PPM 5.1.1

C. Scram the reactor per PPM 5.1.1 and emergency depressurize the RPV.

D. Shutdown the reactor per PPM 3.2.1

The plant has scrammed following a LOCA with fuel damage. RHR-P-2A, which has been reported as having a large packing leak, was started and lined up for RPV injection. Until this RHR pump began injecting, the other ECCS/injection systems were unable to restore RPV level. RPV level is now slowly recovering.

The REACTOR BLDG RAD HIGH annunciator illuminated shortly after RHR-P-2A was started. ARM-RIS-9, RHR A Pump Room indicates GT 10,000 mr/hr. Two additional Reactor Building ARMs are alarming, but indicate LT 500 mr/hr.

Which of the following describes the correct response to this high radiation condition?

A. Enter PPM 5.3.1 Sec. Containment Control and 5.1.3 Emergency RPV Depressurization
   Continue injecting with RHR-P-2A and Emergency Depressurize the RPV

B. Enter PPM 5.3.1 Secondary Containment Control
   Stop injecting with RHR-P-2A

C. Enter PPM 5.3.1 Sec. Containment Control and 5.1.3 Emergency RPV Depressurization
   Stop injecting with RHR-P-2A and Emergency Depressurize the RPV

D. Enter PPM 5.3.1 Secondary Containment Control
   Continue injecting with RHR-P-2A

Which of the following describes the reason Standby Gas Treatment auto starts on the hi radiation signal from Reactor Building Ventilation?

Standby Gas Treatment.....

A. recirculates and filters reactor building atmosphere during accident conditions to allow personnel entry.

B. maintains reactor building pressure of at least +.25 inches of WC during all conditions.

C. limits the release of radioactive material within the guidelines of 10CFR100 during accident conditions.

D. reduces airborne activity during normal operation by exhausting at least one complete volume of reactor building atmosphere per day.

The plant is in MODE 5 with refueling underway. ROA-FN-1A and REA-FN-1A are out of service for maintenance, when REA-FN-1B trips due to an overload.

Which of the following T.S. actions is required?

A. Immediately suspend movement of irradiated fuel in the secondary containment, suspend CORE ALTERATIONS, and initiate action to suspend operations with the potential to drain the vessel.

B. Restore Secondary Containment to operable within 4 hours.

C. Restore Secondary Containment to operable within 12 hours

D. Immediately suspend movement of irradiated fuel in the secondary containment, suspend CORE ALTERATIONS, and initiate action to start Standby Gas Treatment.

The plant is operating at 100% power. There is a fire in RPS Rm. 1. All immediate actions have been taken. HVAC has been secured to prevent the spread of smoke to other areas of the RW Building. Temperature in RPS RM 1 is 102°F and going up.

Which of the following is the correct action based on these conditions?

A. Insert a 1/2 scram for RPS A.

B. Evacuate the main control room.

C. Dispatch guard to cable chase, 467 RW to monitor for spread of fire.

D. Restart normal HVAC or provide portable ventilation.

The plant is operating at 100% power when the LPCS PUMP DISCHARGE PRESS HIGH/LOW annunciator is received, caused by a failure of LPCS-P-2, LPCS Water Leg Pump.

Which of the following pumps should be started as soon as possible?

A. RHR-P-2A

B. RHR-P-2B

C. RHR-P-2C

D. LPCS-P-1

The plant is in MODE 4, with RHR-P-2B in operation in Shutdown Cooling. A manual valving error results in a reactor level reduction that is terminated when level reaches -143 inches. All ECCS pumps and all three diesel generators start. RHR-B does not inject.

Which of the following describes the reason for these conditions?

A. RHR-V-42B, Injection Valve, does not auto open on an initiation signal when in Shutdown Cooling.

B. RHR-V-24B, Full Flow Test Valve, auto opens causing RHR flow to bypass back to the Suppression Pool.

C. RHR-V-4B, Suppression Pool Suction Valve, does not auto open on an initiation signal resulting in a loss of suction flow path.

D. RHR-V-6B, Shutdown Cooling Suction Valve, auto closes causing RHR-P-2B to trip.

A LOCA with a Loss of Offsite Power is in progress with the following conditions:

Reactor level has been below -129 inches for 2 minutes
SM-8 is out of service due to a lockout
RHR-P-2A is running with a sheared shaft
LPCS-P-1 trips due to an electrical problem

Which of the following describes plant response? Assume ADS has NOT been inhibited.

The ADS SRVs ......

A. remain open as long as RHR-P-2A is running.

B. close immediately when the LPCS-P-1 breaker opens.

C. will not open without discharge pressure from both ECCS pumps.

D. close when LPCS discharge pressure is less than 145 psig.

QUESTION # 47 3-5-99

ex99024

The plant is operating at 98% power with TR-S out of service for insulator cleaning. The plant scrams due to a complete loss of feedwater.

Assuming no operator action, which of the following is correct for these conditions?

A. Reactor pressure is reduced to less than 600 psig and Condensate Booster Pumps inject.

B. HPCS-P-1 is powered from DG-3 and injects.

C. RCIC does not inject due to the loss of power.

D. All RHR Pumps start following the repower of SM-7 and 8 from DG-1 and 2.

The reactor was operating at 99% power when a leak in the discharge header of the Condensate Booster Pumps caused a low suction pressure trip of the feed pumps. The following conditions exist:

Reactor pressure	980 psig
Reactor level	-56 inches
HPCS flow	1950 gpm
RCIC flow	600 gpm

Which of the following explains the indicated flow from HPCS-P-1?

A. A high suppression pool level has caused a suction switch over and a low flow until the valves have completed the transfer.

B. HPCS-V-1 and HPCS-V-11, Return to CSTs, have failed open, causing a partial flow bypass to the CSTs and an indicated low flow to the reactor.

C. HPCS-V-4, Injection Valve, has failed 50% open and will not pass full flow.

D. HPCS is rated at 1650 gpm at 1130 psig and 6350 at 200 psig reactor pressure.

The plant is in an ATWS condition with suppression pool temperature 105°F and increasing. When SLC is initiated, only one squib valve fires.

Which of the following is correct under these conditions?

A. Both SLC Pumps start and inject at full system flow.

B. Both SLC Pumps start and inject at 1/2 system flow.

C. The associated SLC Pump (with the open valve) starts and injects at 1/2 system flow.

D. Neither SLC Pump starts until both squib valves have fired.

QUESTION # 50 3-5-99

ex99027

The plant is in a 100% power ATWS condition, with SLC injecting. Efforts are underway to insert control rods.

Which of the following describes when the reactor will remain shutdown under all conditions?

A. SLC-TK-1 has LT 100 gallons of Boron solution remaining in the tank.

B. All withdrawn control rods are seperated by 1 cell in all directions.

C. All control rods inserted to at least position 04.

D. Cold Shutdown Boron Weight injected.

cx99028

The reactor is in MODE 4 with a stable count rate of 230 to 250 counts per second on SRM A, B, and C. SRM D is being withdrawn for maintenance. As the detector is withdrawn through the core, the CRO notes a change in reactor period from ∞ (infinity) to -40 seconds. When the detector drive stops, period changes back to ∞ (infinity).

Which of the following describes these indications?

A. Operation of the drive motor causes a false negative period indication, that clears immediately when the drive motor stops.

B. The detector travels from regions of higher flux to lower flux, causing the period meter to indicate a negative period.

C. The detector undergoes a normalization process during the insertion to the normal location in the core, causing a negative indication.

D. The detector travels from regions of lower flux to higher flux, causing the period meter to indicate a negative period.

cx99029

The reactor is in MODE 5 with the shorting links removed, when SRM-A fails upscale.

Which of the following describes the resulting action?

A. 1/2 scram RPS-A only

B. Full scram.

C. Rod block only

D. 1/2 scram RPS-B only

The plant is operating at 99% power when several alarms are received on P601, P603, and BD C. The CRO notes that APRMs A, C, and E have both the UPSCALE/INOP and DOWNSCALE indicating lights illuminated at P603.

Which of the following describes the reason for the above indications?

Loss of.....

A. MC-8A

B. B1-1

C. B1-2

D. MC-7A

The plant is operating at 98% power when the CRO notes Wide Range level indication on MS-LR/PR-623B about 17 inches lower than the level (+36 inches) indicated on RFW-LI-606A Narrow Range level indication.

Which of the following explains the above indications?

A. There is a reference leg leak on the transmitter for MS-LR/PR-623B.

B. Wide range level instrumentation indicates level inside the core shroud, which is lower than the water level in the downcomer area.

C. Jet pump flow causes Wide range level instrumentation to indicate low at full core flow and power conditions.

D. Drywell temperature at power causes a false low level indication on the Wide Range.

QUESTION # 55 3-5-99

ex99032

The plant is operating at 97% power with a Level 8 trip in on Channel A.
The reference leg for RFW-LS-624B (Channel B level 8) then fails.

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

A. Trip of both Feedpumps, Main turbine, and reactor scram from the turbine trip.

B. Trip of the B Feedpump only and a RRC Pump runback when level 4 is reached.

C. Feedwater level control auto shifts and maintains the reactor at normal level.

D. Trip of both Feedpumps and a reactor scram from loss of feedwater.

QUESTION # 56 3-5-99

ex99033

D

The plant is operating at 75% power with HPCS out of service for replacement of the motor for HPCS-P-2, the water leg pump. HPCS has been out of service for the past 4 days. RCIC-V-8 spuriously goes closed and the breaker trips with the valve in the closed position.

Which of the following is correct for these conditions?

A. Restore RCIC to operable within 14 days.

B. Be in MODE 3 in 12 hours and MODE 4 in 36 hours.

C. Be in MODE 3 in 12 hours and reduce reactor pressure to less than 150 psig in 36 hours.

D. Restore HPCS to operable within 14 days.

The plant was operating at 100% power when a loss of feedwater occurred. RCIC and HPCS have auto started. RCIC then trips and cannot be reset from the control room.

Which of the following caused the RCIC trip?

A. Mechanical Overspeed

B. High Backpressure

C. Low Oil Pressure

D. High Reactor Level

WNP-2 SRO WRITTEN EXAMINATION

QUESTION # 58 3-5-99

ex99035

invalid

The plant is operating at 98% power, when an ~~inadvertent~~ low reactor water level signal starts all Division 1 ECCS equipment.

Which of the following is correct for these conditions?

ADS will......

A. actuate as soon as the 105 second timer times out.

B. not actuate because the -50 inch initiation signal is not sealed in.

C. not actuate because the +13 inch low level confirmatory signal is not sealed in.

D. actuate as soon as the low level initiation signal is received.

RPV level is decreasing following a LOCA and scram from rated conditions. A lockout has occurred on SM-7 and RHR-P-2B is out of service due to a broken pump shaft. RHR-P-2C has auto started on a high drywell pressure signal.

The CRO observes that the ADS timers have initiated and inhibits ADS as directed by the CRS. However, the ADS DIV 2 INHIBIT switch contacts have failed, resulting in no Division 2 inhibit signal being sent to the ADS logic.

Assuming no other operator actions, which of the following describes how ADS will respond to these conditions?

A. ADS will not initiate, because of the lockout on SM-7.

B. When the ADS timer times out, all 7 ADS valves will open.

C. ADS will not initiate, because the ADS DIV 1 INHIBIT switch is in INHIBIT.

D. When the ADS timer times out, 3 ADS valves will open.

The plant is operating at 99% power when MS-RV-5C opens and will not close causing the suppression pool temperature to increase.

Which of the following is the reason for entering PPM 5.1.1 from PPM 5.2.1, prior to exceeding 110°F in the suppression pool?

A. This action is required to maintain temperature within Technical Specification Limits.

B. Assures that the reactor is scrammed to reduce heat input under conditions where heat addition is greater than heat removal capability.

C. This action ensures directions will be give for RPV level control using systems that may also be needed to provide suppression pool cooling.

D. Assures that the reactor is scrammed and shutdown prior to the requirement for boron injection.

The plant was operating at 99% power when a reactor scram and subsequent MSIV (GRP 1) isolation occurred.

Which of the following caused the scram and isolation?

A. Reactor level -53 inches.

B. Drywell pressure 1.76 psig.

C. Loss of RPS-B

D. Loss of RPS-A

WNP-2 SRO WRITTEN EXAMINATION

QUESTION # 62 3-5-99

ex99057

Reactor power is 5% and being increased in preparation for changing to MODE 1. RFW-FCV-10A/B are controlling in auto. RFP-A is operating in manual speed control. The CRO then notes reactor level has decreased to 20" (down from 36") and continues to slowly decrease as reactor power is being increased.

Which of the following describes the reason for this condition?

A. The selected reactor level instrument fails low.

B. A broken air supply line to RFW-FCV-10A/B.

C. An SRV has opened and is stuck open.

D. The B MSL flow transmitter fails high.

ex99058

The plant was operating at 100% power when a loss of Feedwater caused a reactor scram. Several minutes later the CRO notes SGT-FN-1A2 (lag) is running and SGT-FN-1A1 (lead) is off.

Which of the following caused this lineup?

A. SGT-V-3A2, SGT-FN-1A1 Inlet was de-energized and closed

B. SGT gas stream 82% humidity

C. SGT-V-4A1, SGT-FN-1A1 discharge to reactor building failed closed

D. SGT-EHC-1A1 temperature of 224°F

cx99059

The plant is operating at 100% power when a fault at the ASHE Substation causes all high voltage lines into and out of the substation from ASHE to de-energize.

Which of the following is the first to re-power SM-7?

A. TR-S

B. DG-1

C. TR-B

D. DG-2

The plant is operating at 100% power when a loss of MC-7A occurs.

What effect will this have on the operation of IN-1?

A. The inverter is powered from the normal DC power supply, S1-2.

B. The inverter is powered from the normal DC power supply, S2-1.

C. The Static Switch auto transfers to the backup AC supply, MC-7B.

D. The Static Switch auto transfers to the backup AC supply, MC-8A.

ex99061

You are preparing to close the output breaker for DG-1 for the monthly operability test. As the breaker is closed, the generator load increases rapidly and the DG trips off.

Which of the following describes the reason for this condition?

A. The UNIT/PARALLEL switch is in PARALLEL.

B. INCOMING voltage is slightly higher than RUNNING voltage.

C. INCOMING voltage is slightly lower than RUNNING voltage.

D. The UNIT/PARALLEL switch is in UNIT.

D

A fire in Panel DP-S1-1A caused a complete loss of this DC Bus.

Which of the following is the effect this loss will have on the ATWS ARI System?

ATWS-ARI DIV 1 valves will...

A. become inoperable, disabling the ATWS-ARI function.

B. open, a reactor scram will NOT occur.

C. open, resulting in a reactor scram

D. become inoperable, ATWS-ARI will still function.

A plant startup is underway, with the reactor subcritical. Control rod 30-31 is selected and the CRO attempts to withdraw the control rod. The following indications are received:

ACTIVITY CONTROL UNITS DISAGREE on the reactor control console
ROD DRIVE CONTROL SYS INOP annunciator
RPIS OR DMM INOP annunciator
ROD OUT BLOCK annunciator

Which of the following is the cause of these indications?

A. Inoperable transponder card.

B. RWM inoperable.

C. RSCS self test failure.

D. Rod Select Matrix pushbutton failure.

cx99064

As CRO1 withdraws control rods during a plant startup, the 4 rod display for the rod being withdrawn changes from position "32" to "blank" instead of position "34".

Which of the following represents the expected response for the RSCS.

A. - No rod blocks
- Position "34" will automatically appear on the Substitute Position display.
- Depressing Select Substitute Pushbutton will restore indication to the 4 rod display.

B. - A withdraw rod block
- Position "34" should by typed into the Substitute Position display.
- Bypassing that rod on back panel P659 is required to permit continued rod withdrawal.

C. - A withdraw rod block
- Position "34" will automatically appear on the Substitute Position display.
- Depressing Select Substitute Pushbutton will permit continued rod withdrawal.

D. - No rod blocks
- Position "34" should by typed into the Substitute Position display.
- Bypassing that rod on back panel P659 will restore indication to the 4 rod display.

The plant is operating at 100% power. The CRO notices control rod 30-31 has an "i" indication on the RWM.

Which of the following caused this indication?

A. A failed reed switch.

B. A failed activity control unit.

C. The control rod is inserted past its withdraw limit in a lower than latched group.

D. The control rod is withdrawn past its withdraw limit in a lower than latched group.

QUESTION # 71 3-5-99

ex99066

The plant is operating at 100% power when a failure causes the governor valves on RFW-DT-1A to go closed.

Assuming RFW-P-1B continues to operate, which of the following is correct for these conditions?

The recirc pumps ....

A. end up at 15 Hz.

B. end up at 30 Hz.

C. are off with Breakers RPT-3A and 3B open.

D. are off with Breakers RPT-3A/3B and 4A/4B open.

A plant startup is in progress with Blowdown flow control valve, RWCU-FCV-33 throttled open and RWCU-P-1A running. The CLEANUP FLTR INLET TEMP HIGH alarm is received on panel P602. CRO1 reports RWCU NRHX outlet temperature is at 142°F and increasing.

The CRS directs CRO1 to confirm the automatic response of RWCU to high temperature and take manual actions for any automatic actions that failed to occur. The RWCU system automatically responds as follows:

RWCU-V-4 closes
RWCU-P-1A trips

Which of the following is correct for these conditions?

A. Close RWCU-V-1

B. Close RWCU-V-1 and RWCU-FCV-33

C. Close RWCU-FCV-33

D. No actions required, the RWCU system responded correctly.

ex99068

D

A plant startup is in progress when the ROD OUT BLOCK alarm on P603 is received. The CRO determines that the RBM is causing the rod block and observes the following on the RBM indicating "pushbuttons" on the P603 apron for both RBM channel B:

ALARM SET HI	- out	UPSC	- illuminated
ALARM SET INT	- illuminated	DNSC	- out
ALARM SET LO	- out	BYPASS	- out
PUSH TO SET UP	- illuminated		

Which of the following represents the action(s) the CRO should take to clear the rod block and the resulting RBM channel B indication?

A. - place the RBM BYPASS "joystick" in the B position
- UPSC - goes out
- BYPASS - illuminates

B. - depress the PUSH TO SET UP pushbutton
- ALARM SET HI - remains out
- ALARM SET INT- remains illuminated
- UPSC - goes out

C. - depress the BYPASS pushbutton
- UPSC - goes out
- BYPASS - illuminates'

D. - depress the PUSH TO SET UP pushbutton
- ALARM SET HI - illuminates
- ALARM SET INT- goes out
- UPSC - goes out

A reactor startup is in progress with Reactor Power at 10 %.

Which of the following conditions would allow the withdrawal of Intermediate Range Neutron Instrumentation (IRMs) and the reason for retracting the detectors?

A. The Mode Switch in Startup. Only those IRM detectors which have associated APRMs reading > 5 % power can be withdrawn. Detectors are retracted to prolong their life, by decreasing the level of neutron flux to which the detector is exposed during reactor operation in the power range.

B. The Mode Switch in Run. All IRM detectors can be withdrawn. Detectors are retracted to prolong their life, by decreasing the level of neutron flux to which the detector is exposed during reactor operation in the power range.

C. The Mode Switch in Startup. All IRM detectors can be withdrawn as long as IRM to APRM overlap is satisfactorily observed on at least two (2) IRMs. Detectors are retracted to prevent localized overheating of the fuel due to the obstructed coolant flow.

D. The Mode Switch in Run. Only those detectors which have had IRM to APRM overlap satisfactorily observed may be withdrawn. Detectors are retracted to prevent localized overheating of the fuel due to the obstructed coolant flow.

QUESTION # 75 3-5-99

ex99070

Which of the following conditions should be announced to the plant staff over the PA system?

A. A radioactive spill in the RW 437' level has just been reported by the laborer supervisor.

B. The CRO is starting the Auxiliary Oil pump for the 'B' RFP as part of a plant shutdown.

C. The CRO is starting the Turbine Seal Oil Backup pump as part of a plant startup.

D. Stopping SW-P-1A after securing from Shutdown cooling during a plant startup.

The plant is operating at 100% power. A valving error has resulted in HV-V-6A (6A feedwater running vent) being closed.

Which of the following is the expected plant response with continued operation in this lineup?

A. Main generator output will increase due to the lack of steam flow to the 6A heater.

B. Main generator output will decrease due the loss of efficiency of the 6A heater.

C. Reactor power will increase from increased core inlet subcooling.

D. Reactor power will decrease from the increased feedwater inlet temperature.

A plant startup is underway. The following conditions exist:

Reactor Power	567 counts
Reactor Level	36 inches on RFW-FCV-10A/B
Main Turbine	Turning gear
RFP Turbines	A on turning gear, B off
Bearing Oil Pump	Out of service due to a motor replacement

Which of the following describes the expected plant response to a loss of Bus S2-1?

A. Rod block from SRMs inoperable.

B. The Main Turbine turning gear trips from loss of oil pressure.

C. RFP-P-MOP/1A RFP 1A main oil pump trip.

D. Loss of level control due to loss of control signal to RFW-FCV-10A/B.

WNP-2 SRO WRITTEN EXAMINATION

QUESTION # 78 3-5-99

ex99073

The plant is operating at 100% power, with the following auto initiations:

SGT maintaining RB pressure
CSP/CEP have isolated
CR/TSC Emerg Filtration auto start/aligned
RB Room Emerg Coolers start
RB Emerg Lighting Quenched
RB EDR/FDR Sump Headers isolate

These conditions were caused by an inadvertent initiating signal.

Which of the following action have to be taken based on these conditions?

A. Reset the "Z" signal
Reset the FAZ interlock at RC-1and 2
Place the control switches for WMA-FN-54A/B to RESET and back to AUTO

B. Reset the "Z" signal
Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for all 4 SGT Fans to STOP and back to AUTO

C. Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for WMA-FN-54A/B to START

D. Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for all 4 SGT Fans to STOP

The plant is operating at 100% power with RCC-P-1C tagged out for pump alignment. RCC-P-1B trips.

Which of the following is affected by this trip?

A. RRC Seal Heat Exchangers

B. Upper Drywell Coolers

C. Drywell sump heat exchanger

D. RWCU Nonregenerative heat exchangers

While operating at rated power, the Narrow Range level indicators indicate RPV level at +36 inches.

Which of the following represents actual levels inside the RPV and the reason for the difference?

A. 36 inches in the downcomer region
43 inches inside the shroud
due to dP created by resistance to steam flow through the steam dryer

B. 29 inches in the downcomer region
36 inches inside the shroud
due to moisture carryover bypassing the dryer under the seal skirt

C. 36 inches in the downcomer region
29 inches inside the shroud
due to dP created by resistance to steam flow through the steam dryer

D. 43inches in the downcomer region
36 inches inside the shroud
due to moisture carryover bypassing the dryer under the seal skirt

cx99076

The plant is operating at rated power with REA-FN-1A and ROA-FN-1A in service. The "B" fans are in standby.

Which of the following represents the expected response following a momentary loss of bus SM-7?

REA-FN-1A and ROA-FN-1A auto trip and...

A. remain off
Rx BLDG dp will be restored to approximately -1.7" WG by auto start of SGT

B. REA-FN-1B and ROA-FN-1B auto start
Rx BLDG dp will be restored to approximately -0.6" WG by RB HVAC

C. REA-FN-1A and ROA-FN-1A auto restart
Rx BLDG dp will be restored to approximately -0.6" WG by RB HVAC

D. remain off
Rx BLDG dp decreases to approximately 0" WG

The plant is operating at 100% power when breaker N1-3 trips.

Assuming no operator action, which of the following describes the plant response to this malfunction?

A. RFW-P-1B trips causing a power decrease due to a RRC runback to 15 Hz.

B. COND-P-1C and 2C trip causing a loss of Cond/Fw and a low RPV level Scram.

C. RFW-P-1A trips causing a power decrease due to a RRC runback to 15 Hz.

D. COND-P-1B and 2B trip causing a loss of Cond/Fw and a low RPV level Scram.

The control room operator is moving control rods when a ROD DRIFT annunciator is received.

Which of the following controls caused this annunciator?

A. Continuous Insert

B. Notch Insert

C. Notch Withdraw

D. Continuous Withdraw

The plant is operating at 98% power, when several crew members become sick and have to go home 4 hours prior to the end of the scheduled shift. The remaining shift complement consists of 1 SRO, 2 RO and 2 EO. Only Operations personnel were affected by the illness.

Which of the following describes the actions required by Tech Specs?

A. The required EO position may be vacant for a period not to exceed 2 hours provided immediate action is taken to replace the EO.

B. With less than the required shift complement, action must be taken within 2 hours to replace the required position or be in MODE 2 within 12 hours and MODE 3 in the following 12 hours.

C. The required SRO position may be vacant for a period not to exceed 2 hours provided immediate action is taken to replace the SRO.

D. With less than the required shift complement, the position must be filled within 2 hours or immediate action must be taken to place the reactor in MODE 3.

The plant was operating at 100% power when a total Loss of Offsite Power occurred. Conditions have degraded to the point it is necessary to backfeed SM-1 from SM-7 and DG-1.

Which of the following is required?

A. The Bus Shunts have to be removed from the load side of Bkr S-1prior to the backfeed.

B. Permission from the Vice President of Nuclear Operations must be obtained prior to the backfeed.

C. A 10CFR50.54(X) deviation from Tech Specs is required prior to the backfeed.

D. DG-1 must be stopped prior to the backfeed and restarted following breaker alignment.

The reactor is in MODE 5 with fuel movement underway. After moving a bundle through the "cattle chute" and into the vessel cavity, it is observed that the "ROD BLOCK INTERLOCK #1" light does not illuminate. The "HOIST LOADED" indicator is illuminated. The control room reports no rod block indication.

Which of the following actions is correct for these conditions?

A. Immediately stop the refuel bridge until the inoperable rod block is corrected.

B. Immediately initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies.

C. The fuel bundle may be moved back to the spent fuel pool, then immediately suspend in-vessel fuel movement.

D. Fuel movement may continue as long as ROD BLOCK INTERLOCK #2 is operable.

The following conditions exist during a startup:

Reactor level	+37 inches
BPVs controlling pressure	805 psig
APRMs	10% - 11% power
Main Turbine	turning gear
IRMs	20-30 on range 10
BPV position	indicate 9% power

You have been directed to place the mode switch in RUN.

Which of the following is correct concerning this direction?

Placing the Mode Switch in RUN.....

A. is prohibited until exceeding 13% power.

B. is prohibited until BPV power interpolation is greater than APRM power.

C. results in a ROD BLOCK from IRM downscale.

D. results in an MSIV isolation and a reactor scram.

The plant is operating in single loop operation at 21% power. RRC-P-1A is running, RRC-P-1B is ready to restart.

Reactor pressure	977 psig
Bottom head drain temp	391°F
Recirc loop B temp	496°F
Recirc loop A temp	512°F
Reactor level	+36 inches

Considering all of the above data, which of the following is correct concerning the start of RRC-P-1B?

A. The pump cannot be started, based on reactor coolant temperature to bottom head drain temperature.

B. The pump cannot be started, based on recirc loop B to operating loop temperature.

C. The pump can be started, based on recirc loop B to bottom head drain temperature.

D. The pump can be started, based on reactor coolant temperature.

The Control Room Operator has been directed to start the second Reactor Feed Pump during a plant startup. Feed Turbine 1B speed has been increased until the discharge pressure is equal to the discharge pressure of 1A. There is no direction to open RFW-V-102B, the B Feedpump Discharge Valve in PPM 2.2.4 Main Condensate and Feedwater System.

Which of the following actions is correct for these conditions?

A. Stop work on the procedure until a Procedure Revision Form has been submitted and the revision has been approved for implementation.

B. Use a One-Time-Only Temporary Change Notice for PPM 2.2.4 to add the step to open RFW-V-102B and remove the OTO TCN once the procedure is completed.

C. Open RFW-V-102A, continue in the procedure, and submit a Procedure Revision Form when the evolution is completed.

D. Use a Temporary Change Notice for PPM 2.2.4 to add the step to open RFW-V-102B, and continue the startup.

During the performance of a Safety Evaluation it was determined that an Unreviewed Safety Question exists.

Which of the following is **NOT** considered an Unreviewed Safety Question?

It is determined that:

A. Emergency actions that depart from T.S. are needed to protect the public health and safety.

B. The possibility of an accident exists that has not been evaluated by the FSAR.

C. The consequence of a malfunction of equipment evaluated by the FSAR is increased.

D. The margin of safety as defined in T.S. is reduced.

D

Which of the following represents an activity in which maintenance personnel are required to notify the control room **immediately prior to** performing?

A. Electricians testing emergency lighting.

B. Opening the door on WMA-FN-53A air handling unit, causing a dP alarm in control room.

C. Disconnecting a defective fire hose in a reactor building stairwell for replacement.

D. FIN team opening the breaker for the standby potable water pump in the service bldg.

QUESTION # 92 3-5-99

ex98091

The plant is operating at 99% power. A troubleshooting plan for RFW-P-1A is being developed that has a high potential to trip the pump.

Which of the following describes who's concurrence is needed prior to implementation?

A. Licensed SRO, Operations Manager and Plant Manager.

B. Operations Manager, Licensed SRO, and Maintenance Manager.

C. Licensed SRO, CRS/Shift Manager, and Operations Manager.

D. CRS/Shift Manager, Technical Manager, and Engineering Manager.

As part of maintenance on transformers in the WNP-2 switchyard, a clearance is required on BPA equipment.

Which of the following describes the responsibilities for the BPA portion of this clearance?

BPA switches will be tagged by..

A. BPA personnel and
clearance issued in accordance with the WNP2 switching and clearance procedure.

B. WNP2 personnel and
clearance issued in accordance with the BPA switching and clearance procedure.

C. WNP2 personnel and
clearance issued in accordance with the WNP2 switching and clearance procedure.

D. BPA personnel and
clearance issued in accordance with the BPA switching and clearance procedure.

While moving a spent fuel bundle from the core during a refueling outage, the fuel bundle accidentally strikes the edge of the cattle chute while moving into the spent fuel pool. The refuel bridge phone talker then reports seeing bubbles streaming to the surface from the bundle.

What actions are required to be performed immediately?

A. All personnel evacuate the refuel floor and go to the primary access point HP area for further assistance.

B. Place the fuel bundle in a safe location in the nearest refuel storage rack then all personnel leave the refuel bridge.

C. All personnel evacuate the refuel floor area and go to the RB 606' HP control point for further assistance.

D. Lower the fuel bundle into the reactor cavity as far as possible to maximize shielding then all personnel leave the refuel bridge.

A job has to be completed that will cause a Supply System mechanic to receive 9 rem TEDE.

Which of the following describes who has the final review and approval of this Planned Special Exposure?

A. The Plant General Manager and the Vice President, Nuclear Operations.

B. The Radiation Protection Manager and the Operations Manager.

C. The Shift Manager and the employee's immediate Supervisor.

D. The Health Physics Supervisor and the Maintenance Manager

QUESTION # 96 3-5-99

ex98095

An accident has occurred causing combustible concentrations of H_2 and O_2 in both the wetwell atmosphere and the drywell, with wetwell level at 32 feet. The CRS directs the CRO to vent the wetwell per PPM 5.5.20.

Which of the following describes the reason for this direction?

A. This flowpath causes the containment atmosphere to be scrubbed in the wetwell prior to being vented, which reduces the radioactivity of the containment gases.

B. Venting the wetwell is desired because the highest concentration of H_2 and O_2 are expected to be found in the wetwell.

C. N_2 is used to purge the containment because it combines with the H_2 and O_2 and removes them through the drywell vent path.

D. This flowpath uses Nitrogen to push hydrogen out of the wetwell and drywell, and to decrease the flammability of the containment atmosphere.

PPM 5.4.1 Radioactivity Release Control requires Emergency Depressurization if the exclusion area boundary release rate approaches or exceeds the General Emergency limit.

Which of the following describes the reason for this requirement?

A. Emergency Depressurization reduces reactor pressure and allows low pressure systems to inject into the reactor, limiting the release to the environment.

B. The pressure reduction realized by Emergency Depressurization slows the rate of fuel damage in the reactor core and reduces the rate of release outside of the containment.

C. The pressure reduction allows the containment vent path (MSIVs) to open and vent the primary system to the condenser, reducing the discharge to the environment.

D. RPV depressurization reduces the driving head and flow of primary systems that are unisolated and discharging outside of containment.

ex99082

The plant has experienced a major transient including a scram, due to a loss of several electrical buses. As a result, the CRS has implemented actions per the EOPs. CRO2 then observes that a lockout has occurred on bus SM-8 and without prompting, takes the immediate actions for the loss of SM-8.

Which of the following describes the CRO's actions?

A. This action is correct because..
immediate actions per the ARPs should always be performed regardless of the EOPs.

B. This action is in-correct because..
actions per the ARPs should not be performed unless specified in the EOPs.

C. This action is correct because..
he correctly diagnosed the event and took action which does not conflict with the EOPs.

D. This action is in-correct because..
while in the EOPs, the CRS should direct any actions not specified in the EOPs.

The plant was operating at 97% power when a LOCA occurred. Reactor level dropped to -110 inches before recovering. The following conditions now exist:

Reactor level	- 47 inches and up slow with condensate
Drywell pressure	1.3 psig and stable
Reactor pressure	14 psig and stable
Wetwell temperature	peaked at 89°F and is down slow with RHR-P-2B running in Suppression Pool Cooling

Which of the following is correct for the above conditions?

A. Permission from the Emergency Director would be required prior to unisolating containment isolations..

B. Exit the EOPs and return to the normal or abnormal operating procedures.

C. Enter PPM 5.2.1 Primary Containment Control.

D. Start and inject with all available ECCS pumps until level is in the normal range.

A terrorist group has just called the control room and threatened to place a bomb in the A SW Pump House in the next hour.

Which of the following describes the immediate actions to be taken?

A. Notify the on duty Security Lieutenant of the bomb threat and call the Richland City Police Department Bomb Squad .

B. Record information on the Bomb Threat Call Checklist on the phone book and notify Security.

C. Evacuate the Protected Area and the Exclusion Area of all unnecessary personnel and notify Security.

D. Notify the SCC officer and the Hanford Fire Department to respond by pushing the button on FCP-1.

U.S. Nuclear Regulatory Commission Site Specific Written Examination	
Applicant Information	
Name:	Region IV
Date:	WNP-2
RO Written Exam	GE-BWR 5
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 four 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	________ Points
Applicant's Score	________ Points
Applicant's Grade	________ Percent

WRITTEN EXAMINATION GUIDELINES

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

2. To pass the examination, you must achieve a grade of 80.00 percent or greater. Every question is worth one point.

3. The time limit for completing the examination is four hours.

4. You may bring pens and calculators into the examination room. Use only black ink to ensure legible copies.

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

6. Mark your answers on the answer sheet provided and do not leave any question blank. Use only the paper provided and do not write on the back side of the pages. If you decide to change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

7. If the intent of a question is unclear, ask questions of the NRC examiner or the designated facility instructor only.

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

9. When you complete the examination, assemble a package including the examination questions, examination aids, answer sheets, and scrap paper and give it to the NRC examiner or proctor. Remember to sign the statement on the examination cover sheet indicating that the work is your own and that you have neither given nor received assistance in completing the examination. The scrap paper will be disposed of immediately after the examination.

10. After you have turned in your examination, leave the examination area as defined by the proctor or NRC examiner. If you are found in this area while the examination is still in progress, your license may be denied or revoked.

11. Do you have any questions?

REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: ____________________

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

1.	a	b	c	d	26.	a	b	c	d
2.	a	b	c	d	27.	a	b	c	d
3.	a	b	c	d	28.	a	b	c	d
4.	a	b	c	d	29.	a	b	c	d
5.	a	b	c	d	30.	a	b	c	d
6.	a	b	c	d	31.	a	b	c	d
7.	a	b	c	d	32.	a	b	c	d
8.	a	b	c	d	33.	a	b	c	d
9.	a	b	c	d	34.	a	b	c	d
10.	a	b	c	d	35.	a	b	c	d
11.	a	b	c	d	36.	a	b	c	d
12.	a	b	c	d	37.	a	b	c	d
13.	a	b	c	d	38.	a	b	c	d
14.	a	b	c	d	39.	a	b	c	d
15.	a	b	c	d	40.	a	b	c	d
16.	a	b	c	d	41.	a	b	c	d
17.	a	b	c	d	42.	a	b	c	d
18.	a	b	c	d	43.	a	b	c	d
19.	a	b	c	d	44.	a	b	c	d
20.	a	b	c	d	45.	a	b	c	d
21.	a	b	c	d	46.	a	b	c	d
22.	a	b	c	d	47.	a	b	c	d
23.	a	b	c	d	48.	a	b	c	d
24.	a	b	c	d	49.	a	b	c	d
25.	a	b	c	d	50.	a	b	c	d

REACTOR OPERATOR - ANSWER SHEET

Multiple Choice (Circle your choice) NAME: ______________________

If you change your original answer, draw a single line through the error, enter the desired answer, and initial the change.

51.	a	b	c	d	76.	a	b	c	d
52.	a	b	c	d	77.	a	b	c	d
53.	a	b	c	d	78.	a	b	c	d
54.	a	b	c	d	79.	a	b	c	d
55.	a	b	c	d	80.	a	b	c	d
56.	a	b	c	d	81.	a	b	c	d
57.	a	b	c	d	82.	a	b	c	d
58.	a	b	c	d	83.	a	b	c	d
59.	a	b	c	d	84.	a	b	c	d
60.	a	b	c	d	85.	a	b	c	d
61.	a	b	c	d	86.	a	b	c	d
62.	a	b	c	d	87.	a	b	c	d
63.	a	b	c	d	88.	a	b	c	d
64.	a	b	c	d	89.	a	b	c	d
65.	a	b	c	d	90.	a	b	c	d
66.	a	b	c	d	91.	a	b	c	d
67.	a	b	c	d	92.	a	b	c	d
68.	a	b	c	d	93.	a	b	c	d
69.	a	b	c	d	94.	a	b	c	d
70.	a	b	c	d	95.	a	b	c	d
71.	a	b	c	d	96.	a	b	c	d
72.	a	b	c	d	97.	a	b	c	d
73.	a	b	c	d	98.	a	b	c	d
74.	a	b	c	d	99.	a	b	c	d
75.	a	b	c	d	100.	a	b	c	d

ex99044

D

Late in core life, fuel burnup and control rod density cause the first few inches of control rod travel to add much less negative reactivity during a scram.

Which of the following is designed the help mitigate this effect?

A. Reactor Scram from low EH Fluid Header Pressure

B. Reactor Scram from Turbine Throttle Valve Position

C. RRC Pump Runback from a Reactor level 3

D. RRC Pump Trip from a Main Turbine Trip.

The reactor has just been scrammed.

Which of the following conditions ensures the reactor will remain shutdown under all conditions?

All control rods...

A. fully inserted except one at position 46.

B. fully inserted except one at position 04 and one at position 08.

C. fully inserted except two and 900 gal. of boron solution injected.

D. did NOT fully insert and 1500 gal of boron solution injected.

The reactor is operating at 99% power when an APRM hi flux scram occurs.

Which of the following caused the scram?

A. Trip of both Reactor Recirculation Pumps.

B. Lockout on SM-7

C. GV closure due to a DEH computer failure.

D. Loss of RPS B

The plant was operating at 97% power when the reactor scrammed. After conditions have stabilized the CRO notes both recirc pumps are tripped with CB-3A and 3B open (CB4A and 4B closed).

Which of the following signals caused the scram?

A. Reactor level -61 inches.

B. Reactor pressure 1112 psig.

C. Main Turbine Trip.

D. Drywell pressure 1.75 psig.

The plant was operating at 99% power. Following a scram, these conditions exist:

Reactor power	0%
Reactor level	+12 inches
Drywell pressure	+1.72 psig
Suppression Pool level	+1.87 inches
Rx Building Pressure	-.12 inches H2O

Which of the following describes the procedures to be referenced first?

A. PPM 3.3.1 Reactor Scram, PPM 4.5.7.1 Main Turbine Trip

B. PPM 5.1.1 RPV Control, PPM 5.2.1 Primary Containment Control

C. PPM 5.1.1 RPV Control, PPM 5.3.1 Secondary Containment Control

D. PPM 4.5.7.1 Main Turbine Trip, PPM 5.2.1 Primary Containment Control

A rod withdrawal is underway for a startup. IRMs are all selected to range 1, with a reactor period of 115 seconds. The operator selects the third control rod in Rod Worth Minimizer group 4 to move. After the rod has been moved, a reactor period of 24 seconds is noted.

What action is required with these conditions?

A. Stop control rod withdrawal and notify the SNE.

B. Insert control rods until period is GT 60 sec., and wait until power is in the heating range.

C. Manually scram the reactor.

D. Insert control rods until the reactor is subcritical and notify the CRS and SNE.

ex99005

The plant is operating at 100% power. CRD-V-102, Overpiston area isolation, has been left closed following maintenance of the HCU for control rod 30-31. The plant then scrams.

Which of the following is correct based on these conditions?

A. The outer tube could rupture and cause a small Loss of Coolant Accident.

B. The collet piston could be damaged to the point the control rod will not stay at position 00 following the scram.

C. The drive seal could be degraded to the point of subsequent slow insert and withdraw speeds.

D. The drive could be damaged and cause the rod to become stuck resulting in not all rods inserting following the scram.

QUESTION # 8 3-5-99

ex99011

The reactor was operating at 98%power when an error during the performance of a surveillance caused an MSIV isolation and reactor scram. The following conditions exist:

Reactor level	15 inches
Reactor pressure	1087 psig
Drywell pressure	1.63 psig
Suppression pool temp.	86°F

Which of the following Emergency Operating Procedures should be entered:

A. PPM 5.2.1 Primary Containment Control, based on 86°F in the suppression pool.

B. PPM 5.1.1 RPV Control, based on 1087 psig in the reactor.

C. PPM 5.2.1 Primary Containment Control, based on 1.63 psig in the drywell.

D. PPM 5.1.1 RPV Control, based on 15 inches in the reactor.

QUESTION # 9 3-5-99

cx99014

The reactor was operating at 99% power when a main turbine trip occurred due to an electrical fault in the Main Generator. Reactor conditions have stabilized and the CRO is ready to transfer FWLC to RFW-FCV-10A/B, when he notes reactor level at 18 inches and stable.

Which of the following describes the requirements for level returning to +36 inches?

A. The Primary Programmable Logic Controller has failed and needs to be reset.

B. FWLC will return level to normal 12 minutes following the scram.

C. FWLC will return level to normal 9 minutes following the scram reset..

D. The Master Level Controller has failed and will have to be placed in manual.

An ATWS has occurred. SLC is injecting.

Which of the following conditions would permit a reactor cooldown to be initiated?

A. 3400 gallons of SLC have been injected.

B. Reactor power: 1% power and lowering.

C. Reactor power: IRM range 6 and lowering.

D. 3200 gallons of SLC have been injected.

A DBA LOCA has occurred from 100% power. The following plant conditions exist:

Drywell Hydrogen	5.8%
Drywell Oxygen	4.2%
Core damage	Estimated at 15%
Drywell pressure	4 psig and down slow

Which of the following is correct for these conditions?

A. Calibrate the H202 analyzers for the drywell.

B. Secure CAC taking suction on the drywell.

C. Initiate CAC with suction from the drywell.

D. Emergency depressurize the reactor with 7 ADS SRVs.

The plant is operating at 100% power. The following conditions exist:

RHR-P-2B	In operation on minimum flow
Drywell pressure	Increasing due to small steam leak
RRC-P-1B	Motor cooler leak
CRD 30-31	Leaking drive water connection

The CRO then notes suppression pool level increasing.

What is the reason for the suppression pool level increase?

A. RHR-P-2B in operation with minimum flow open.

B. Motor Cooler leak on RRC-P-1B.

C. Leaking drive water connection on CRD 30-31.

D. Drywell pressure increase from the small steam leak.

The first step in PPM 5.1.2, RPV Control - ATWS is to "Inhibit ADS"

Which of the following is the basis for this action?

A. This action prevents unwanted depressurization of the RPV which could result in uncontrolled injection from unborated water causing a power excursion.

B. The ADS system is not analyzed for ATWS conditions, more controlled pressure reductions are performed manually when conditions in the EOPs direct it.

C. This action prevents unwanted depressurization of the RPV which could result in uncontrolled reactivity addition due to addition of voids in the core.

D. The ADS system is not analyzed for ATWS conditions. Auto actuation at the ADS setpoint would be premature for ATWS conditions.

The plant was operating at 63% power when RRC-P-1B tripped.

Which of the following describes how accurate core flow indication is maintained?

Jet Pump Loop B flow indication is ...

A. excluded from total core flow summation.

B. averaged with Jet Pump Loop A indicated flow.

C. added to Jet Pump Loop A flow.

D. reduced to one-half of Jet Pump Loop A indicated flow.

The plant is operating at 100% power when an unisolable vacuum leak develops that is greater than the capacity of Off Gas. A reactor scram then occurs.

Assuming no operator action, which of the following caused the scram?

A. High Reactor Pressure

B. Main Turbine Trip

C. Low reactor level

D. MSIV Closure

The plant is operating at 100% power when the CRO receives a 1/2 scram on RPS-A and several alarms on Board C.

Which of the following caused these indications?

A trip of Bkr........

A. MC-7C

B. 7-71

C. MC-7B

D. 7-73

A fault has caused the complete loss of DC Bus S1-1.

Which of the effects would this loss have on plant equipment?

Loss of ....

A. DC to IN-2

B. power to Bkr 1-7

C. DC to IN-1

D. power to Bkr 7-1

cx98104

The plant is operating at 99% power with the following conditions:

Rx building pressure	-.21 inches H2O
RFW-P-1A area temperature	187°F
Drywell air temperature	139°F
Wetwell temperature	87°F

Which of the following is correct considering these conditions?

A. Enter PPM 5.2.1 Primary Containment Control because the drywell air temperature exceeds the LCO value.

B. Enter PPM 5.3.1 Secondary Containment Control because the RFW-P-1A area exceeds the maximum safe operating temperature for that area.

C. Enter PPM 5.3.1 Secondary Containment Control because Rx building pressure is less than the alarm setpoint.

D. Enter PPM 5.2.1 Primary Containment Control because wetwell temperature exceeds the high level alarm value.

The reactor is operating at 99% power when a fire breaks out in the control room that requires an immediate control room evacuation.

Which of the following actions is required prior to leaving the control room?

A. Manually scram the reactor with all four manual scram pushbuttons.
Lock the mode switch in SHUTDOWN.
Transfer RPV level control to RFW-FCV-10A/B
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.

B. Manually scram the reactor by placing the mode switch in SHUTDOWN.
Transfer RPV level control to RFW-FCV-10A/B
Request security at SAS place 5 security doors in the unlock position.
Announce on the PA and radio; the scram, evacuation and an EO to local start DG2

C. Manually scram the reactor by placing the mode switch in SHUTDOWN.
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.
Request security at SAS place 5 security doors in the unlock position.
Announce on the PA and radio; the scram, evacuation and an EO to local start DG2

D. Manually scram the reactor with all four manual scram pushbuttons.
Close the MSIVs by arming and depressing the A, B, C, and D pushbuttons.
Verify all rods full in.
Ensure all APRMs are downscale.

The plant is operating at 100% power. The following conditions exist:

Unidentified Leakage	4 gpm and stable
RCC-V-48 - surge tank makeup	open
FDR-P-4A sump pump in FDR-R-3	running

Which of the following would result in all of the above indications?

A. Packing leak on RHR-V-4A - RHR-P-2A Suction

B. RCC Heat Exchanger RCC-HX-1A leak (RB 548')

C. Drywell Cooler CRA-FC-2A leak

D. Packing leak from RRC-V-67B - RRC-P-2B discharge

ex99048

The plant is operating at 100% power with EDR-P-5A running more than normal. Steam is coming out of EDR-R-5.

Which of the following problems is the cause of these conditions?

A. A scram valve seat leak.

B. A suction line leak on RRC-P-1A.

C. A Containment Nitrogen leak on the supply to MS-RV-4A.

D. A packing leak on RWCU-V-4, outboard suction isolation.

The plant is operating at 100% reactor power with maintenance underway that causes an inadvertent SLC system initiation.

What is the expected plant response?

A. RHR-V-60A and 60B, RHR heat exchanger sample isolation valves close.

B. RCC-V-5, 104, 40, and 21, RCC Supply and Return Drywell isolations close

C. All Primary Containment Recirculation Fans trip.

D. RWCU-V-4, Reactor Water Cleanup outboard containment isolation closes.

D

Reactor pressure is 198 psig with a rod withdrawal for startup underway. All systems are normal except, CRD-P-1A is out of service for a motor replacement. CRD-P-1B then trips causing charging header pressure decreases to less than 940 psig and an accumulator alarm on control rod 30-31, which is withdrawn.

Which of the following is correct for these conditions?

The reactor Mode Switch must be placed in Shutdown within one hour because ......

A. the loss of the second CRD Pump will cause an rapid increase in CRD seal temperatures.

B. of the loss of seal purge to the Recirculation Pumps.

C. of the potential indicated level transient from NBI.

D. at low RPV pressure, accumulators are the only motive force for scramming control rods.

ex99012

The EOPs direct that the reactor be scrammed before wetwell (WW) temperature reaches a value which is equivalent to the Boron Injection Initiation Temperature (BIIT).

The BIIT is defined to be the greater temperature which results from either the WW temperature at which Technical Specifications require a reactor scram or the highest WW temperature at which initiation of SLC will result in....

A. injection of the Hot Shutdown Boron Weight before the WW exceeds pump vortex limits.

B. injection of the Cold Shutdown Boron Weight before the WW heats to the PSP limit.

C. injection of the Cold Shutdown Boron Weight before the WW exceeds the PCPL.

D. injection of the Hot Shutdown Boron Weight before the WW heats to the HCTL.

Why is Emergency Depressurization required when Wetwell Level cannot be maintained above 19 feet 2 inches?

Below 19 feet 2 inches.....

A. suppression of steam from the RPV cannot be assured with LOCA blowdown.

B. failure of the drywell floor seal may occur during a LOCA.

C. a LOCA could cause downcomer failure above the level of the suppression pool.

D. initiation of Suppression Pool Sprays may cause a failure of the WW/DW interface.

The plant has scrammed following a LOCA with fuel damage. RHR-P-2A, which has been reported as having a large packing leak, was started and lined up for RPV injection. Until this RHR pump began injecting, the other ECCS/injection systems were unable to restore RPV level. RPV level is now slowly recovering.

The REACTOR BLDG RAD HIGH annunciator illuminated shortly after RHR-P-2A was started. ARM-RIS-9, RHR A Pump Room indicates GT 10,000 mr/hr. Two additional Reactor Building ARMs are alarming, but indicate LT 500 mr/hr.

Which of the following describes the correct response to this high radiation condition?

A. Enter PPM 5.3.1 Sec. Containment Control and 5.1.3 Emergency RPV Depressurization
Continue injecting with RHR-P-2A and Emergency Depressurize the RPV

B. Enter PPM 5.3.1 Secondary Containment Control
Stop injecting with RHR-P-2A

C. Enter PPM 5.3.1 Sec. Containment Control and 5.1.3 Emergency RPV Depressurization
Stop injecting with RHR-P-2A and Emergency Depressurize the RPV

D. Enter PPM 5.3.1 Secondary Containment Control
Continue injecting with RHR-P-2A

Which of the following describes the reason Standby Gas Treatment auto starts on the hi radiation signal from Reactor Building Ventilation?

Standby Gas Treatment.....

A. recirculates and filters reactor building atmosphere during accident conditions to allow personnel entry.

B. maintains reactor building pressure of at least +.25 inches of WC during all conditions.

C. limits the release of radioactive material within the guidelines of 10CFR100 during accident conditions.

D. reduces airborne activity during normal operation by exhausting at least one complete volume of reactor building atmosphere per day.

Why is it desirable to use ADS SRVs when Emergency Depressurizing?

A. They are designed to handle the dynamic load of 7 SRVs open simultaneously.

B. Their use evenly distributes the heat load around the wetwell.

C. Limits the local stress on the suppression pool floor.

D. Equalizes the stress on the drywell floor/downcomer interface.

cx99045

The plant is operating at 100% power with reactor level at 53 inches and up fast.

Which of the following describes the expected plant response and the reason for that response?

A. The main turbine trips to prevent damage to the Main Turbine blading

B. MSIV Isolation to prevent pitting of the MSIVs

C. The main turbine trips to prevent seat damage to the Main Turbine Throttle valves

D. The RFP turbines trip to prevent overloading of the RFP Moisture Pre-separator

ex99051

A steam leak has developed inside primary containment; causing drywell pressure and temperature to increase. Both parameters are within the DSIL Curve and are trending up. Drywell temperature is 275°F. The CRS has given the direction to spray the drywell.

Which of the following describes the reason for spraying the drywell at this time?

A. Ensure that 95% of the non-condensable gases are transferred to the Wetwell.

B. Prevent exceeding the ADS design temperature.

C. Prevent possible damage to the safety relief valve (SRV) operators.

D. Reduce drywell temperature before the structural design temperature limits are reached.

The plant is operating at 100% power. There is a fire in RPS Rm. 1. All immediate actions have been taken. HVAC has been secured to prevent the spread of smoke to other areas of the RW Building. Temperature in RPS RM 1 is 102°F and going up.

Which of the following is the correct action based on these conditions?

A. Insert a 1/2 scram for RPS A.

B. Evacuate the main control room.

C. Dispatch guard to cable chase, 467 RW to monitor for spread of fire.

D. Restart normal HVAC or provide portable ventilation.

Which of the following describes the reason for Emergency Depressurization with a high Suppression Pool Level?

A. A large LOCA would exceed the Heat Capacity Temperature Limit resulting in the failure of the wetwell/drywell interface.

B. SRV discharge would result in exceeding code allowable stresses in the tail pipe, tail pipe supports, quenchers, or quencher supports.

C. A large LOCA would result in exceeding the SRV Tail Pipe Level limit and exceed the Primary Containment Pressure Limit.

D. SRV discharge would cause excessive containment pressure on the drywell floor and exceed the Primary Containment Pressure Limit.

ex99050

The plant has been in MODE 4 for two hours following a 9 month run at 100% power. RRC-P-1A is in operation with reactor level at 36 inches. RHR-P-2B is in operation in Shutdown Cooling, when it is inadvertently tripped by the CRO.

Which of the following describes the required actions?

A. Start RRC-P-1B and operate both RRC pumps at 15hz

B. Immediately raise RPV level to 60 - 80" to provide natural circulation.

C. Re-establish shutdown cooling as soon as possible and maintain level at 36 inches.

D. Perform Alternate Shutdown cooling using a HPCS or LPCS pump.

ex99009

The reactor is in MODE 5. The following conditions exist:

SRM-A	OOS
SRM-B	2.8 counts per second (S/N 9.3/1)
SRM-C	2.5 counts per second (S/N 10.7/1)
SRM-D	OOS
MODE SWITCH	REFUEL
Reactor Coolant Temperature	108°F
Reactor Head and Internals	Removed

It is desired to start moving irradiated fuel from the vessel for refueling at this time.

Which of the following describes the reason why or why not irradiated fuel movement would be allowed to start at this time?

Irradiated fuel movement .....

A. may not start at this time, there are no operable SRMs.

B. may not start at this time, reactor coolant temperature is greater than 100°F.

C. may start in either the quadrant with the B or C SRM.

D. may start in any quadrant in the reactor vessel.

ex99055

The plant is in MODE 5 with refueling underway. ROA-FN-1A and REA-FN-1A are out of service for maintenance, when REA-FN-1B trips due to an overload.

Which of the following T.S. actions is required?

A. Immediately suspend movement of irradiated fuel in the secondary containment, suspend CORE ALTERATIONS, and initiate action to suspend operations with the potential to drain the vessel.

B. Restore Secondary Containment to operable within 4 hours.

C. Restore Secondary Containment to operable within 12 hours

D. Immediately suspend movement of irradiated fuel in the secondary containment, suspend CORE ALTERATIONS, and initiate action to start Standby Gas Treatment.

ex99086

The plant is shutdown with RHR-P-2B in operation for Shutdown Cooling. A crack in the SDC suction line causes a start of FDR-P-2 and a high water level alarm in the B RHR pump room.

In addition to this...

A. Isolation of RHR SDC suction valve RHR-V- 9 takes place at +13 inches to prevent flooding the RHR pump room.

B. Isolation of RHR SDC suction valve RHR-V-8 takes place at -50 inches to prevent flooding the RHR pump room.

C. Isolation of RHR SDC suction valves RHR-V-8 and 9 take place at -50 inches to prevent the further loss of reactor water inventory.

D. Isolation of RHR SDC suction valves RHR-V-8 and 9 take place at +13 inches to prevent the further loss of reactor water inventory.

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ex98105

D

The plant is operating at 98% power with CRD-P-1B tagged out of service for a seal leak. All other systems are operating normally. Breaker 7-1 then trips on overcurrent.

Which of the following is correct for these conditions?

A. Seal purge will be lost to both Recirc Pumps, both pumps have to be tripped.

B. Motor cooling is lost to both Recirc Pumps, operation may continue.

C. Crud may build up in the Recirc pump seals, operation may continue.

D. Seal Cooling is lost to both Recirc Pumps, both pumps have to be tripped.

ex99063

A plant startup is underway, with the reactor subcritical. Control rod 30-31 is selected and the CRO attempts to withdraw the control rod. The following indications are received:

ACTIVITY CONTROL UNITS DISAGREE on the reactor control console
ROD DRIVE CONTROL SYS INOP annunciator
RPIS OR DMM INOP annunciator
ROD OUT BLOCK annunciator

Which of the following is the cause of these indications?

A. Inoperable transponder card.

B. RWM inoperable.

C. RSCS self test failure.

D. Rod Select Matrix pushbutton failure

ex99021

The plant is operating at 100% power when the LPCS PUMP DISCHARGE PRESS HIGH/LOW annunciator is received, caused by a failure of LPCS-P-2, LPCS Water Leg Pump.

Which of the following pumps should be started as soon as possible?

A. RHR-P-2A

B. RHR-P-2B

C. RHR-P-2C

D. LPCS-P-1

D

The plant is in MODE 4, with RHR-P-2B in operation in Shutdown Cooling. A manual valving error results in a reactor level reduction that is terminated when level reaches -143 inches. All ECCS pumps and all three diesel generators start. RHR-B does not inject.

Which of the following describes the reason for these conditions?

A. RHR-V-42B, Injection Valve, does not auto open on an initiation signal when in Shutdown Cooling.

B. RHR-V-24B, Full Flow Test Valve, auto opens causing RHR flow to bypass back to the Suppression Pool.

C. RHR-V-4B, Suppression Pool Suction Valve, does not auto open on an initiation signal resulting in a loss of suction flow path.

D. RHR-V-6B, Shutdown Cooling Suction Valve, auto closes causing RHR-P-2B to trip.

ex99023

The plant is operating at 100% power with LPCS-P-1 in full flow test operation for a surveillance. A valid high drywell pressure signal is received due to a small steam leak in the drywell.

What is the response of the LPCS system?

A. LPCS-V-5 Injection Valve opens
LPCS-V-12 Test return to suppression pool closes

B. LPCS-V-12 Test return to suppression pool remains open
LPCS-V-5 Injection Valve remains closed

C. LPCS-V-12 Test return to suppression pool remains open
LPCS-FCV-11 Minimum Flow remains closed

D. LPCS-V-12 Test return to suppression pool closes
LPCS-FCV-11 Minimum Flow opens

The plant is operating at 98% power with TR-S out of service for insulator cleaning. The plant scrams due to a complete loss of feedwater.

Assuming no operator action, which of the following is correct for these conditions?

A. Reactor pressure is reduced to less than 600 psig and Condensate Booster Pumps inject.

B. HPCS-P-1 is powered from DG-3 and injects.

C. RCIC does not inject due to the loss of power.

D. All RHR Pumps start following the repower of SM-7 and 8 from DG-1 and 2.

ex99025

The reactor was operating at 99% power when a leak in the discharge header of the Condensate Booster Pumps caused a low suction pressure trip of the feed pumps. The following conditions exist:

Reactor pressure	980 psig
Reactor level	-56 inches
HPCS flow	1950 gpm
RCIC flow	600 gpm

Which of the following explains the indicated flow from HPCS-P-1?

A. A high suppression pool level has caused a suction switch over and a low flow until the valves have completed the transfer.

B. HPCS-V-1 and HPCS-V-11, Return to CSTs, have failed open, causing a partial flow bypass to the CSTs and an indicated low flow to the reactor.

C. HPCS-V-4, Injection Valve, has failed 50% open and will not pass full flow.

D. HPCS is rated at 1650 gpm at 1130 psig and 6350 at 200 psig reactor pressure.

cx99026

The plant is in an ATWS condition with suppression pool temperature 105°F and increasing. When SLC is initiated, only one squib valve fires.

Which of the following is correct under these conditions?

A. Both SLC Pumps start and inject at full system flow.

B. Both SLC Pumps start and inject at 1/2 system flow.

C. The associated SLC Pump (with the open valve) starts and injects at 1/2 system flow.

D. Neither SLC Pump starts until both squib valves have fired.

cx98107

Reactor power is in the IRM range during a rod withdrawal for start up, when 24 VDC power feeding IRM-C fails.

Which of the following describes the effect on the plant startup?

IRM-C.....

A. UPSCALE causes a rod block and 1/2 scram from RPS A, rod withdrawal can continue following bypass of IRM-C.

B. UPSCALE causes a rod block, rod withdrawal is stopped until IRM-C is restored to operation.

C. INOP causes a rod block and 1/2 scram from RPS-A, rod withdrawal can continue following bypass of IRM-C.

D. INOP causes a rod block, rod withdrawal is stopped until IRM-C is restored to operation.

The reactor is in MODE 5 with the shorting links removed, when SRM-A fails upscale.

Which of the following describes the resulting action?

A. 1/2 scram RPS-A only

B. Full scram.

C. Rod block only

D. 1/2 scram RPS-B only

The reactor is in MODE 4 with a stable count rate of 230 to 250 counts per second on SRM A, B, and C. SRM D is being withdrawn for maintenance. As the detector is withdrawn through the core, the CRO notes a change in reactor period from ∞ (infinity) to -40 seconds. When the detector drive stops, period changes back to ∞ (infinity).

Which of the following describes these indications?

A. Operation of the drive motor causes a false negative period indication, that clears immediately when the drive motor stops.

B. The detector travels from regions of higher flux to lower flux, causing the period meter to indicate a negative period.

C. The detector undergoes a normalization process during the insertion to the normal location in the core, causing a negative indication.

D. The detector travels from regions of lower flux to higher flux, causing the period meter to indicate a negative period.

The plant is operating at 99% power when several alarms are received on P601, P603, and BD C. The CRO notes that APRMs A, C, and E have both the UPSCALE/INOP and DOWNSCALE indicating lights illuminated at P603.

Which of the following describes the reason for the above indications?

Loss of.....

A. MC-8A

B. B1-1

C. B1-2

D. MC-7A

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ex98059

The plant is operating at 99% power when the reference line for Narrow Range RFW-DPT-4C starts leaking.

Which of the following describes the expected control room indication?

As the leak increases, P603 indication RFW-LI-606C.....

A. stays the same until the reference and variable leg equalize then increases.

B. decreases.

C. increases for 2 to 3 minutes and returns to the original value.

D. increases.

The plant is operating at 99% power when MS-LS-61A (reactor level 2) fails, causing a spurious trip signal.

Which of the following actions is correct for this failure?

A. Full inboard isolation (MSIVs stay open).

B. RC-1 1/2 TRIP annunciation.

C. Full outboard isolation (MSIVs stay open).

D. ADS A LOGIC INITIATED annunciation.

cx98106

The plant was operating at 71% power when a transient occurred that caused Rx level to decrease to -61 inches. HPCS is out of service.

Which of the following is correct concerning these conditions? Assume no operator actions.

A. Reactor level is returned to +18 inches and maintained there by the feedpumps at the level setdown setpoint.

B. RHR initiates, but does not inject until reactor pressure decreases to the injection valve open permissive at 450 psig.

C. RCIC initiates and injects until reactor level increases to +54 inches, RCIC-V-45 and 13 close until reactor level returns to -50 inches, where RCIC injects again.

D. Reactor level continues to decrease until -129 inches when LPCS initiates and injects into the vessel until the operator closes the injection valve.

ex99036

RPV level is decreasing following a LOCA and scram from rated conditions. A lockout has occurred on SM-7 and RHR-P-2B is out of service due to a broken pump shaft. RHR-P-2C has auto started on a high drywell pressure signal.

The CRO observes that the ADS timers have initiated and inhibits ADS as directed by the CRS. However, the ADS DIV 2 INHIBIT switch contacts have failed, resulting in no Division 2 inhibit signal being sent to the ADS logic.

Assuming no other operator actions, which of the following describes how ADS will respond to these conditions?

A. ADS will not initiate, because of the lockout on SM-7.

B. When the ADS timer times out, all 7 ADS valves will open.

C. ADS will not initiate, because the ADS DIV 1 INHIBIT switch is in INHIBIT.

D. When the ADS timer times out, 3 ADS valves will open.

The plant is operating at 99% power when MS-RV-5C opens and will not close causing the suppression pool temperature to increase.

Which of the following is the reason for entering PPM 5.1.1 from PPM 5.2.1, prior to exceeding 110°F in the suppression pool?

A. This action is required to maintain temperature within Technical Specification Limits.

B. Assures that the reactor is scrammed to reduce heat input under conditions where heat addition is greater than heat removal capability.

C. This action ensures directions will be give for RPV level control using systems that may also be needed to provide suppression pool cooling.

D. Assures that the reactor is scrammed and shutdown prior to the requirement for boron injection.

ex99038

The plant was operating at 99% power when a reactor scram and subsequent MSIV (GRP 1) isolation occurred.

Which of the following caused the scram and isolation?

A. Reactor level -53 inches.

B. Drywell pressure 1.76 psig.

C. Loss of RPS-B

D. Loss of RPS-A

The plant is operating at 100% power. A valving error has resulted in HV-V-6A (6A feedwater running vent) being closed.

Which of the following is the expected plant response with continued operation in this lineup?

A. Main generator output will increase due to the lack of steam flow to the 6A heater.

B. Main generator output will decrease due the loss of efficiency of the 6A heater.

C. Reactor power will increase from increased core inlet subcooling.

D. Reactor power will decrease from the increased feedwater inlet temperature.

The plant was operating at 100% power when a loss of Feedwater caused a reactor scram. Several minutes later the CRO notes SGT-FN-1A2 (lag) is running and SGT-FN-1A1 (lead) is off.

Which of the following caused this lineup?

A. SGT-V-3A2, SGT-FN-1A1 Inlet was de-energized and closed

B. SGT gas stream 82% humidity

C. SGT-V-4A1, SGT-FN-1A1 discharge to reactor building failed closed

D. SGT-EHC-1A1 temperature of 224°F

cx99061

You are preparing to close the output breaker for DG-1 for the monthly operability test. As the breaker is closed, the generator load increases rapidly and the DG trips off.

Which of the following describes the reason for this condition?

A. The UNIT/PARALLEL switch is in PARALLEL.

B. INCOMING voltage is slightly higher than RUNNING voltage.

C. INCOMING voltage is slightly lower than RUNNING voltage.

D. The UNIT/PARALLEL switch is in UNIT.

QUESTION # 58 3-5-99

ex99027

D

The plant is in a 100% power ATWS condition, with SLC injecting. Efforts are underway to insert control rods.

Which of the following describes when the reactor will remain shutdown under all conditions?

A. SLC-TK-1 has LT 100 gallons of Boron solution remaining in the tank.

B. All withdrawn control rods are seperated by 1 cell in all directions.

C. All control rods inserted to at least position 04.

D. Cold Shutdown Boron Weight injected.

QUESTION # 59

3-5-99

cx99035

The plant is operating at 98% power, when an ~~inadvertent~~ invalid low reactor water level signal starts all Division 1 ECCS equipment.

Which of the following is correct for these conditions?

ADS will......

A. actuate as soon as the 105 second timer times out.

B. not actuate because the -50 inch initiation signal is not sealed in.

C. not actuate because the +13 inch low level confirmatory signal is not sealed in.

D. actuate as soon as the low level initiation signal is received.

Reactor power is 5% and being increased in preparation for changing to MODE 1. RFW-FCV-10A/B are controlling in auto. RFP-A is operating in manual speed control. The CRO then notes reactor level has decreased to 20" (down from 36") and continues to slowly decrease as reactor power is being increased.

Which of the following describes the reason for this condition?

A. The selected reactor level instrument fails low.

B. A broken air supply line to RFW-FCV-10A/B.

C. An SRV has opened and is stuck open.

D. The B MSL flow transmitter fails high.

The plant is operating at 100% power when a small steam leak causes a high drywell pressure scram.

What is the basis for this scram?

A. Reduces the amount of reactor coolant discharged to the suppression pool and allows available cooling to maintain containment temperature.

B. Minimizes the possibility of fuel damage and reduces the amount of energy added to the containment.

C. Reduces the inventory loss in the reactor vessel.

D. Minimizes the activity discharged to the reactor building.

The plant was operating at 100% power when a complete loss of CAS caused a reactor scram. CN-V-65 is closed. CIA pressure has been 154 psig for the last 2 minutes.

Which of the following is correct concerning the CIA system?

A. The CIA programmers have placed their respective banks of N2 bottles in service.

B. CIA-V-39A/B have closed.

C. The CIA programmers have placed their respective banks of N2 bottles in service and stopped at step 1.

D. CIA-V-39A/B closed and then reopened.

The plant is operating at 38% power with DEH-P-1A out of service for breaker maintenance. DEH-P-1B then trips.

Assuming no operator action, what is the plant response to these conditions?

A. The main oil pump discharge supplies control oil and operation continues as before.

B. The turbine trips and bypass valves open to control pressure with reactor still at power.

C. The turbine trips and the reactor scrams from the loss of DEH pressure.

D. The seal oil backup pump discharge supplies control oil and operation continues as before.

DG-1 has been running for the last 6 hours at 2000 KW supplying SM-7 following a manual start. Normal power has been restored and is available to (but not supplying) the bus and DG-1 is to be shut down.

What action must be taken prior to shutting down DG-1?

A. The diesel must be stopped with the EMERG STOP pushbutton.

B. Load DG-1 to at least 100% for at least 10 minutes prior to shutting the diesel down.

C. The diesel can be stopped with the normal stop switch following a 30 minute idle period.

D. Load DG-1 to GT 50% of rated for at least 30 minutes prior to shutting the diesel down.

The control room operator is moving control rods when a ROD DRIFT annunciator is received.

Which of the following controls caused this annunciator?

A. Continuous Insert

B. Notch Insert

C. Notch Withdraw

D. Continuous Withdraw

QUESTION # 66 3-5-99

ex98109

A rod withdrawal is underway for a startup. The pull sheet directs the reactor operator to withdraw rod 18-15 from position 00 to position 04. As the rod settles into position 04 an RMCS rod block is generated with no annunciation.

Which of the following describes the reason for these indications?

A. The control rod is withdrawn past its RWM withdraw limit.

B. The RBM has applied a withdraw block and the limit can now be set up.

C. The RMCS has applied a withdraw block from an upscale alarm.

D. The control rod is at the RSCS normal withdraw limit for that rod.

The plant is operating at 100% power when a failure causes the governor valves on RFW-DT-1A to go closed.

Assuming RFW-P-1B continues to operate, which of the following is correct for these conditions?

The recirc pumps ....

A. end up at 15 Hz.

B. end up at 30 Hz.

C. are off with Breakers RPT-3A and 3B open.

D. are off with Breakers RPT-3A/3B and 4A/4B open.

A plant startup is underway. The following conditions exist:

APRM A 28%
APRM B 30%
APRM C 29%
APRM D is bypassed.
APRM E 32%
APRM F 31%
RBM A indicates BYPASS
RBM B indicates ALARM SET LOW

Which of the following describes the reasons for these indications?

A. APRM A less than 30% causes RBM A to indicate BYPASS
APRM F (alternate for APRM D) at 31% causes RBM B to indicate ALARM SET LOW.

B. APRM C less than 30% causes RBM A to indicate BYPASS
APRM B at 30% causes RBM B to indicate ALARM SET LOW.

C. APRM D in bypass causes RBM A to indicate BYPASS
APRM E at 32% causes RBM B to indicate ALARM SET LOW.

D. APRM C less than 30% causes RBM A to indicate BYPASS
APRM F (alternate for APRM D) at 31% causes RBM B to indicate ALARM SET LOW.

QUESTION # 69 3-5-99

ex99077

The plant is operating at 100% power when breaker N1-3 trips.

Assuming no operator action, which of the following describes the plant response to this malfunction?

A. RFW-P-1B trips causing a power decrease due to a RRC runback to 15 Hz.

B. COND-P-1C and 2C trip causing a loss of Cond/Fw and a low RPV level Scram.

C. RFW-P-1A trips causing a power decrease due to a RRC runback to 15 Hz.

D. COND-P-1B and 2B trip causing a loss of Cond/Fw and a low RPV level Scram.

The plant is operating at 100% power, with the following auto initiations:

SGT maintaining RB pressure
CSP/CEP have isolated
CR/TSC Emerg Filtration auto start/aligned
RB Room Emerg Coolers start
RB Emerg Lighting Quenched
RB EDR/FDR Sump Headers isolate

These conditions were caused by an inadvertent initiating signal.

Which of the following action have to be taken based on these conditions?

A. Reset the "Z" signal
Reset the FAZ interlock at RC-1and 2
Place the control switches for WMA-FN-54A/B to RESET and back to AUTO

B. Reset the "Z" signal
Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for all 4 SGT Fans to STOP and back to AUTO

C. Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for WMA-FN-54A/B to START

D. Push the Isolation Logic Reset pushbuttons A&B and C&D on P601
Place the control switches for all 4 SGT Fans to STOP

QUESTION # 71 3-5-99

ex99059

The plant is operating at 100% power when a fault at the ASHE Substation causes all high voltage lines into and out of the substation from ASHE to de-energize.

Which of the following is the first to re-power SM-7?

A. TR-S

B. DG-1

C. TR-B

D. DG-2

The plant is operating at 100% power when a loss of MC-7A occurs.

What effect will this have on the operation of IN-1?

A. The inverter is powered from the normal DC power supply, S1-2.

B. The inverter is powered from the normal DC power supply, S2-1.

C. The Static Switch auto transfers to the backup AC supply, MC-7B.

D. The Static Switch auto transfers to the backup AC supply, MC-8A.

QUESTION # 73 3-5-99

ex99065

The plant is operating at 100% power. The CRO notices control rod 30-31 has an "i" indication on the RWM.

Which of the following caused this indication?

A. A failed reed switch.

B. A failed activity control unit.

C. The control rod is inserted past its withdraw limit in a lower than latched group.

D. The control rod is withdrawn past its withdraw limit in a lower than latched group.

A plant startup is underway. The following conditions exist:

Reactor Power	567 counts
Reactor Level	36 inches on RFW-FCV-10A/B
Main Turbine	Turning gear
RFP Turbines	A on turning gear, B off
Bearing Oil Pump	Out of service due to a motor replacement

Which of the following describes the expected plant response to a loss of Bus S2-1?

A. Rod block from SRMs inoperable.

B. The Main Turbine turning gear trips from loss of oil pressure.

C. RFP-P-MOP/1A RFP 1A main oil pump trip.

D. Loss of level control due to loss of control signal to RFW-FCV-10A/B.

The plant is operating at 100% power with RCC-P-1C tagged out for pump alignment. RCC-P-1B trips.

Which of the following is affected by this trip?

A. RRC Seal Heat Exchangers

B. Upper Drywell Coolers

C. Drywell sump heat exchanger

D. RWCU Nonregenerative heat exchangers

The plant is operating at 100% power when a seal fails on RWCU-P-1A causing reactor coolant leakage into the pump room.

Which of the following describes expected RWCU response?

A. only RWCU-V-4 closes

B. RWCU-V-1 and 4 close

C. only RWCU-V-1 closes

D. RWCU-P-1A trips

The plant is in MODE 4 with RHR-P-2A running in shutdown cooling. RHR-P-2B is running in suppression pool cooling, lowering suppression pool level through RHR-V-40 and 49. The following actions take place:

RHR-V-8	closes
RHR-V-9	closes
RHR-V-53A	closes
RHR-P-2A	trips
RHR-V-40	remains open
RHR-V-49	remains open

Which of the following is the reason for these actions?

A. RHR High Area Temperature

B. Drywell pressure 1.72 psig

C. Reactor level +58 inches

D. High flow to radwaste, through RHR-V-40 and 49

A control rod withdrawal for start up is underway. Control rod 34-11 is being withdrawn from position 00 to 48. At position 42, the RO notes an XX indication.

Which of the following describes the reason for this indication?

The reed switch at position.....

A. 40 failed to close.

B. 40 failed to open.

C. 42 failed to open.

D. 42 failed to close.

ex99092

The plant has been shut down for 2 hours following a long run at 100% power, with RCIC in operation for level control and RHR-P-2B in suppression pool cooling mode. All containment parameters have been stable except drywell pressure. Drywell pressure has increased to 1.75 psig.

Which of the following is correct based on these conditions? Assume no other operator actions.

A. RHR-V-48B, HX Bypass, is interlocked open for 10 min.

B. RHR-V-42B, Injection Valve, cannot be closed.

C. RHR-V-24B, Test Return, cannot be opened.

D. RHR-P-2B has to be restarted to return the system to suppression pool cooling.

QUESTION # 80 3-5-99

ex99093

PPM 5.2.1 gives direction to spray the wetwell prior to exceeding 12 psig in the wetwell.

Which of the following describes the failure prevented by this direction?

A. Containment failure from exceeding external pressure limits.

B. Failure of the SRV T Quenchers.

C. Failure of the drywell floor/downcomer junction.

D. Wetwell/Drywell vacuum breaker failure.

The plant was operating at 100% power when a failure caused a full MSIV Isolation. All parameters are now in the normal band and stable.

Why is the operator directed to Perform OSP-CVB/IST-M701, Suppression Chamber-Drywell Vacuum Breaker Operability by PPM 3.3.1 Reactor Scram?

A. Each vacuum breaker must be cycled to ensure it opens and pressure is equalized between the suppression chamber and drywell following SRV actuation.

B. SRV actuation can cause suppression chamber-drywell vacuum breakers to fail full open causing a flowpath for bypassing the pressure suppression function of the suppression pool during a DBA LOCA.

C. Operation of the suppression chamber-drywell vacuum breakers following SRV actuation allows the non-condensables from the drywell to return to the drywell.

D. Each vacuum breaker must be cycled to ensure it opens and returns to its full closed position to perform its safety function within 12 hours following a discharge of steam through the SRVs

The plant is operating at 100% power. A spill in the reactor building has caused Reactor Building Ventilation exhaust plenum radiation to reach 21 mr/hr.

What effect does this have on operation of the following control room ventilation fans and dampers?

1. WMA-AD-51A1/B1 Fan 51A/B Inlet dampers
2. WOA-V-51C/52C Normal Air Intake valves
3. WMA-FN-54A/B CR Emerg Filtration
4. WEA-FN-51 Toilet/Kitchen Exhaust Fan

A. 1 - open
 2 - close
 3 - starts
 4 - starts

B. 1 - close
 2 - close
 3 - starts
 4 - trips

C. 1 - close
 2 - open
 3 - trips
 4 - starts

D. 1 - open
 2 - open
 3 - trips
 4 - trips

QUESTION # 83 3-5-99

ex99096

The plant is operating at 100% power when a leak develops in the instrument air dryer. The leak is larger than the capacity of the CAS System. The Service Air Header pressure low alarm has annunciated at 85 psi.

Which of the following is the next expected auto action to mitigate the effect of the leak?

A. Standby CAS air compressor auto starts.

B. SA compressor fully loaded.

C. CAS-PCV-1, Air Dryer Bypass Valve, opens

D. SA-PCV-2, Service Air Header Isolation Valve, closes.

Following a RB HVAC system outage, the CRO is returning the system to service. All plant AC and DC electrical busses are energized. An EO reports that the close fuses have been installed for Rx Bldg. supply fan, ROA-FN-1A. All REA and ROA fans are off with their control switches in Normal after Stop.

The CRO places the ROA-FN-1A control switch to start. The CRO then notices that the RED and GREEN lamps for this fan are both off but verifies that both light bulbs are good.

Which of the following describes the current RB HVAC configuration?

A. ROA-FN-1A is off. Rx Bldg. dP has remained at 0 inches WG.

B. ROA-FN-1B has auto started. Rx Bldg. dP has increased to + 4 inches WG.

C. ROA-FN-1A is running. Rx Bldg. dP has increased to + 4 inches WG.

D. ROA-FN-1A is running. Rx Bldg. dP has increased to greater than + 7 inches WG.

The plant is in a DBA LOCA condition due to an earthquake. All ECCS systems are available. The seismic activity has caused a "Rams Head" assembly to break off of one jet pump. The jet pump throat and diffuser remain intact.

Which of the following describes the effect on water level of this condition?

A. Water level will reflood to at least 2/3 core height.

B. Reactor level will return to normal level (+13 to 54 in.).

C. Reactor level will return to 2/3 core height inside the core shroud and normal level in the downcomer area.

D. Water level will not reflood the core to any level due to the loss of the "Rams Head" assembly.

D

The plant is in MODE 5 with a full core off load 98% completed. RHR-P-2B is in operation in Fuel Pool Cooling Assist. Breaker 8-3 trips due to an overcurrent.

Which of the following is correct based on these conditions?

A. DG-2 starts and supplies bus SM-8.

B. Breaker B-8 closes and supplies bus SM-8.

C. The spent fuel pool temperature begins to increase.

D. RHR-P-2A will be started in Fuel Pool Cooling Assist Mode.

ex99098

A reactor startup is underway with one SRM out of service and bypassed. Count rate is 478 counts on the operable SRMs. All IRMs are on Range 1. A second SRM fails downscale.

Which of the following describes the required action?

A. Restore at least one SRM to operable within 4 hours.

B. Fully insert all insertable control rods within 1 hour.

C. Place the Mode Switch in SHUTDOWN.

D. Suspend further control rod withdrawal until 3 SRMs are operable.

QUESTION # 88 3-5-99

cx98119

Two CROs are in the control room. A backpanel alarm occurs for the RB HVAC system when CRO1 observes that CRO3 is in the SM office discussing an upcoming evolution.

Which of the following is the correct action in this situation concerning going to the backpanels to investigate the alarm?

A. CRO1 investigates the alarm, while CRO3 observes plant conditions from the SM office.

B. CRO1 investigates the alarm, but takes as little time as possible to return to the front area.

C. CRO3 accompanies CRO1 to peer check his investigation of the alarm.

D. CRO3 investigates the alarm while CRO1 remains at the CRO desk near P603.

A plant startup is underway, and you have been given the direction to start the A Reactor Feed Pump.

What are the requirements for procedure usage for this evolution?

A. The procedure need not be present for this evolution if the operator assigned is familiar with the task.

B. The procedure must be present but steps can be skipped for more efficient operation of the plant if they are noted administratively on the front page of the procedure.

C. The procedure need not be present for this evolution but the steps of the procedure must be performed in order.

D. The procedure must be present and strict adherence to the procedure is required for operating the equipment.

cx99100

The plant is in a LOCA condition that has caused an auto start of the ECCS systems. HPCS did not auto start.

Who is responsible for attempting to manually starting HPCS?

A. The STA if closest to H13-P601.

B. Any licensed operator at the console.

C. Only the CRO assigned to H13-P601.

D. Only the operator assigned by the CRS to verify system response.

cx98121

The CRO is performing a periodic surveillance in the control room using the supplied procedure.

Which of the following is correct?

The CRO assures that this is the current revision by...

A. the requirements of the surveillance program that it be the current revision.

B. verifying the same revision number in the Level 1 controlled copy in the control room.

C. the CRS/SM signature on the surveillance procedure.

D. verifying the current revision through the Passport program.

QUESTION # 92 3-5-99

ex98122

The plant is in an ATWS condition. The CRO has placed ECCS VALVE LOGIC OVERRIDE keylock switch, LPCS-RMS-S21, for injection valve LPCS-V-5 to OVERRIDE. Reactor pressure is currently at 900 psig. The RPV pressure open permissive relay, LPCS-RLY-K8 is de-energized due to reactor pressure currently above its associated pressure switch setpoint.

Which of the following describes the correct response of LPCS-V-5 when the control switch is taken to the OPEN position?

LPCS-V-5 will .....

A. NOT open, because LPCS-RMS-S21 overrides this control switch.

B. OPEN, because LPCS-RMS-S21 bypasses the pressure permissive.

C. NOT open, because the reactor pressure permissive is not met.

D. OPEN, because the reactor pressure permissive is met.

Shortly before withdrawing control rods as part of a reactor startup, the Mode Switch is moved from the SHUTDOWN to the START/HOT STBY position.

This is done to permit rod withdrawal...

A. for plant startup administrative reasons only.

B. by removing the scram signal that exists in the SHUTDOWN position.

C. by adjusting the setpoints for the SRM and IRM rod blocks to a higher value.

D. by removing the rod withdraw block signal that exists in the SHUTDOWN position.

cx98124

During the performance of a quarterly RCIC Valve Operability Test surveillance, the CRO observed that the stroke time of the RCIC valve just tested failed to meet the acceptance criteria by 2 seconds.

Which of the following is the correct action to be taken at this point?

A. Complete the surveillance then inform the CRS/SM of the discrepancy.

B. Re-test that step in the surveillance to confirm the reading.

C. Immediately inform the CRS/SM for evaluation regarding equipment operability.

D. No action required since Tech. Specs. permits a 2 second tolerance.

Which of the following pieces of equipment would require independent verification for return to service?

A. SGT-FN-1A1

B. REA-FN-1B

C. TOA-FN-1A

D. WOA-FN-1B

The plant is in the middle of a refueling outage. No core alterations are currently in progress. At 1800, the CRO is informed that the fuel handlers will be resuming the fuel shuffle beginning at 2030.

Which of the following is required for communications between the Control Room and the Refuel Floor?

A. Continuously observe the fuel moves via the TV monitor and ensure the fuel handlers are briefed to call the control room if any problems occur.

B. Establish direct communications with the refuel bridge between 1930 and 2030 and document it in the CRO log.

C. Establish direct communications with the refuel bridge between 2030 and 2130.

D. Remain in direct communication with the bridge the entire time the refuel bridge is manned and the vessel head is removed with fuel in the vessel.

cx98126

You've been assigned to briefly inspect a valve inside of a valve room which is a posted RADIATION AREA. Health Physics informs you that the radiation levels where you will be working are at the minimum value as defined for a RADIATION AREA posting.

Your personal electronic dosimeter is set to alarm at an accumulated dose of 10 mr. It currently indicates 0 mr. What is the time you could expect to be in the area before it alarms?

A. 2 hours

B. 1/2 hour

C. 1 hour

D. 1 1/2 hours

ex98094

PPM 5.4.1 Radioactivity Release Control requires Emergency Depressurization if the exclusion area boundary release rate approaches or exceeds the General Emergency limit.

Which of the following describes the reason for this requirement?

A. Emergency Depressurization reduces reactor pressure and allows low pressure systems to inject into the reactor, limiting the release to the environment.

B. The pressure reduction realized by Emergency Depressurization slows the rate of fuel damage in the reactor core and reduces the rate of release outside of the containment.

C. The pressure reduction allows the containment vent path (MSIVs) to open and vent the primary system to the condenser, reducing the discharge to the environment.

D. RPV depressurization reduces the driving head and flow of primary systems that are unisolated and discharging outside of containment.

QUESTION # 99 3-5-99

cx98129

During a major plant transient, the CRS directs a CRO to re-align an RHR loop from RPV injection to containment spray. The CRO, under current plant conditions, doesn't understand why this direction is being given and feels that taking these actions will lead to core uncovery.

Which of the following is the correct action to be taken by the CRO?

A. Take the action as directed, the CRS is the SRO in command.

B. Challenge this instruction by pointing out that the core will become uncovered.

C. Take the action as directed, then question it after completing the re-alignment.

D. First review the EOP flowchart and confirm that this is the correct action.

A generator fault has just resulted in a Main Turbine trip and reactor scram from rated conditions. All plant equipment and parameters responded as expected with the exception that 2 control rods failed to indicate fully inserted. The CRO has performed the following:

Place the Mode Switch in SHUTDOWN.
Monitor reactor power, level, and pressure .
Insert all SRMs and IRMs.
Depress all 4 manual scram logic pushbuttons.

Which of the following describes an additional required immediate action?

A. Manually initiate ARI from P603.

B. Verify Recirc Pumps have tripped to 15 hz.

C. Perform a rapid transfer to RFW-FCV-10A/B control of RPV level.

D. Select and drive in the two rods that failed to insert.