2013 DRESDEN NUCLEAR POWER STATION

INITIAL LICENSE EXAMINATION

ADMINISTRATIVE FILES

Facility: DRESDEN Date of Examination: Oct 2013

Developed by: Written - Facility ☐ NRC ☒ // Operating - Facility ☐ NRC ☒

Target Date*	Task Description (Reference)	Chief Examiner's Initials
-180	1. Examination administration date confirmed (C.1.a; C.2.a and b)	RKW 5/3/13
-120	2. NRC examiners and facility contact assigned (C.1.d; C.2.e)	RKW 5/3/13
-120	3. Facility contact briefed on security and other requirements (C.2.c)	RKW 5/3/13
-120	4. Corporate notification letter sent (C.2.d)	RKW 5/28/13
[-90]	[5. Reference material due (C.1.e; C.3.c; Attachment 3)]	RKW 6/10/13
{-75}	6. Integrated examination outline(s) due, including Forms ES-201-2, ES-201-3, ES-301-1, ES-301-2, ES-301-5, ES-D-1's, ES-401-1/2, ES-401-3, and ES-401-4, as applicable (C.1.e and f; C.3.d)	N/A
{-70}	{7. Examination outline(s) reviewed by NRC and feedback provided to facility licensee (C.2.h; C.3.e)}	N/A
{-45}	8. Proposed examinations (including written, walk-through JPMs, and scenarios, as applicable), supporting documentation (including Forms ES-301-3, ES-301-4, ES-301-5, ES-301-6, and ES-401-6, and any Form ES-201-3 updates), and reference materials due (C.1.e, f, g and h; C.3.d)	N/A
-30	9. Preliminary license applications (NRC Form 398's) due (C.1.l; C.2.g; ES-202)	RKW 9/16/13
-14	10. Final license applications due and Form ES-201-4 prepared (C.1.l; C.2.i; ES-202)	RKW 9/30/13
-14	11. Examination approved by NRC supervisor for facility licensee review (C.2.h; C.3.f)	RKW 10/17/13
-14	12. Examinations reviewed with facility licensee (C.1.j; C.2.f and h; C.3.g)	RKW 10/17/13
-7	13. Written examinations and operating tests approved by NRC supervisor (C.2.i; C.3.h)	RKW 10/17/13
-7	14. Final applications reviewed; 1 or 2 (if >10) applications audited to confirm qualifications / eligibility; and examination approval and waiver letters sent (C.2.i; Attachment 5; ES-202, C.2.e; ES-204)	RKW 10/3/13
-7	15. Proctoring/written exam administration guidelines reviewed with facility licensee (C.3.k)	RKW 10/30/13
-7	16. Approved scenarios, job performance measures, and questions distributed to NRC examiners (C.3.i)	RKW 8/26/13

* Target dates are generally based on facility-prepared examinations and are keyed to the examination date identified in the corporate notification letter. They are for planning purposes and may be adjusted on a case-by-case basis in coordination with the facility licensee.
[Applies only] {Does not apply} to examinations prepared by the NRC.

1. **Pre-Examination**

I acknowledge that I have acquired specialized knowledge about the NRC licensing examinations scheduled for the week(s) of 10/21/13 as of the date of my signature. I agree that I will not knowingly divulge any information about these examinations to any persons who have not been authorized by the NRC chief examiner. I understand that I am not to instruct, evaluate, or provide performance feedback to those applicants scheduled to be administered these licensing examinations from this date until completion of examination administration, except as specifically noted below and authorized by the NRC (e.g., acting as a simulator booth operator or communicator is acceptable if the individual does not select the training content or provide direct or indirect feedback). Furthermore, I am aware of the physical security measures and requirements (as documented in the facility licensee's procedures) and understand that violation of the conditions of this agreement may result in cancellation of the examinations and/or an enforcement action against me or the facility licensee. I will immediately report to facility management or the NRC chief examiner any indications or suggestions that examination security may have been compromised.

2. **Post-Examination**

To the best of my knowledge, I did not divulge to any unauthorized persons any information concerning the NRC licensing examinations administered during the week(s) of 10/21/13. From the date that I entered into this security agreement until the completion of examination administration, I did not instruct, evaluate, or provide performance feedback to those applicants who were administered these licensing examinations, except as specifically noted below and authorized by the NRC.

	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	Jeff Nelson	Exam Author		6/11/13		10/31/13	
2.	THOMAS DIETFIELD	FACILITY REPRESENTATIVE		7/17/13		11/7/13	
3.	Michael Purcell	Contractor Exam Prep		7/30/13		1/29/14	
4.	J Rush	OTPS		8/12/13		10/31/13	
5.	CHRIS WAGNER	CLERK / COPYING		8-12-13		10-31-13	
6.	C Kolocka	Exam Author (Support)		8-13-13		10/31/13	
7.	Ryan Sears	Validation		8/13/13		11/24/13	
8.	BRIAN PIGG	VALIDATION		8-13-13		11-22-13	
9.	Paul J Chernone	Validation		8-13-13		11-22-13	
10.	GLEN CASTRO	Validator		8-13-13		11-22-13	
11.	STEVEN MELL	VALIDATOR		8/13/13		11/22/13	
12.	Steven Gulick	Validator		8-26-13		11-1-13	
13.	M. Kerchenfaut	Validator		8/26/13		11/1/13	
14.	KEN GRAY	Validator		8-26-13		10/31/13	
15.	Anton Reyes	Sim SW		8-26-13		11/7/13	

NOTES:

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	William Sitarz	Hardware Technician	[signature]	8/26	[signature]	11/7/13	
2.	Del Thomas	Simulator Supervisor	[signature]	8/26	[signature]	11/7/13	
3.	Christopher Bachman	Validator	[signature]	9/10/13	[signature]	10/31/13	
4.	Daniel Daly	Validator	[signature]	9/10/13	[signature]	10/31/13	
5.	Doug Bastian	Validator	[signature]	9.10.13	[signature]	10/31/13	
6.	Greg Leonatti	Validator	[signature]	9/10/13	[signature]	11/22/13	
7.	Wayne Cope	Sim Op / Communicator	[signature]	9/30/13	[signature]	10/31/13	
8.	Scott Briley	Validator	[signature]	9/30/13	[signature]	11/22/13	
9.	Al Zlomie	Validator	[signature]	9/30/13	[signature]	11.7.13	
10.	Sean Jenson	Validator	[signature]	9/30/13	[signature]	11/22/13	
11.	Dan Miller	Validator	[signature]	9/30/13	[signature]	11/22/13	
12.	Phillip Prater	Validator / Mgmt Supt.	[signature]	10/4/13	[signature]	10/31/13	
13.	Paul DiGiovanna	Trng Director	[signature]	10/4/13	[signature]	10/31/13	
14.	Katharine Netemeyer	Validator	[signature]	10/9/13	[signature]	11/22/13	
15.	Kyle Koller	Validator	[signature]	10/9/13	[signature]	11/22/13	

NOTES:

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	PRINTED NAME	JOB TITLE / RESPONSIBILITY	SIGNATURE (1)	DATE	SIGNATURE (2)	DATE	NOTE
1.	Michael Johnson	Exam Admin	[signature]	10-23-13	[signature]	10-31-13	
2.							
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4.							
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14.							
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NOTES:

Facility: Dresden Station Date of Exam: 11/30/2013 Exam Level: RO ☒ SRO ☒

Item Description	Initials a	Initials b	Initials c
1. Clean answer sheets copied before grading	dm	N/A	RKW
2. Answer key changes and question deletions justified and documented	dm	↓	RKW
3. Applicants' scores checked for addition errors (reviewers spot check > 25% of examinations)	dm	↓	RKW
4. Grading for all borderline cases (80 ±2% overall and 70 or 80, as applicable, ±4% on the SRO-only) reviewed in detail	dm	↓	RKW
5. All other failing examinations checked to ensure that grades are justified	dm	↓	RKW
6. Performance on missed questions checked for training deficiencies and wording problems; evaluate validity of questions missed by half or more of the applicants	dm	↓	RKW

	Printed Name/Signature	Date
a. Grader	D. McNeil/ [signature]	11/27/13
b. Facility Reviewer(*)	n/a	n/a
c. NRC Chief Examiner (*)	R. K. Walton/ [signature]	11/27/13
d. NRC Supervisor (*)	H. Peterson/ [signature]	11/27/13

(*) The facility reviewer's signature is not applicable for examinations graded by the NRC; two independent NRC reviews are required.

Dresden Station

ILT 12-1 NRC Exam

Examinee & Station Comments

Exelon Generation

Dresden Nuclear Power Station
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Morris, IL 60450
815 942 2920 Telephone
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10 CFR 55.41
10 CFR 55.43

November 6, 2013

SVPLTR # 13-0044

Regional Administrator, Region III
U. S. Nuclear Regulatory Commission
2443 Warrenville Road - Suite 210
Lisle, IL 60532-4352

ATTN: Keith Walton

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
Docket Nos. 50-237 and 50-249

Subject: Submittal of Post 2013 Dresden Initial License Written Examination Comments

In accordance with NUREG 1021, ES-402 E.(4) and (5), enclosed are the post examination comments for the 2013 Dresden Initial License Written Examination.

This submittal includes comments made on seven questions from the Initial License Exam candidates and the corresponding facility response for each of these questions.

Should you have any questions concerning this matter, please contact Mr. G. Morrow, Regulatory Assurance Manager at (815) 416-2800.

Respectfully,

for S. Marik

Shane Marik
Site Vice President
Dresden Nuclear Power Station

Enclosure: Post 2013 Dresden Initial License Written Examination Comments

CC (Without enclosure)
Chief, NRC Operator Licensing Branch
NRC Senior Resident Inspector – Dresden Nuclear Power Station
Regional Administrator - NRC Region III

NOV - 6 2013

Question # 009

Unit 2 is in MODE 3 with the following set of conditions:

- One turbine bypass valve is full open
- RPV water temperature at 316°F and slowly lowering
- RPV pressure is 245 psig, slowly lowering.
- 2B Recirc pump is running at minimum speed.
- 2A EHC Pump is out of service.

Then an overcurrent condition occurs on Bus 27.

For these conditions, what is the preferred method of heat removal from the RPV?

a. Open an additional turbine bypass valve fully.

b. Place Isolation Condenser System in service per DOP 1300-03.

c. Initiate HPCI System in pressure control mode per DOP 2300-03.

d. Alternate opening of Electromatic Relief Valve(s) at five minute intervals.

ANSWER: B

CANDIDATE FEEDBACK (DOCKET NO. 055- 33736):

The correct answer to the question was the Isolation Condenser per DOP 1300-03. However, based on the information provided, the Isolation Condenser would not be able to provide adequate heat removal and be able to control reactor pressure. The question stated that 1 Turbine bypass valve was full open. Per DGP 02-01 page 64, one Turbine Bypass valve is worth 112.5 MWth and the Isolation Condenser is worth 74 MWth (as shown below).

TABLE 1
EQUIPMENT HEAT REMOVAL CAPACITIES

SYSTEM/COMPONENT		HEAT REMOVAL CAPABILITY ($K = X\ 10^3$ $M = X\ 10^6$)			
		MWth	LBM/HR	FLUID	EQUIV MN STM FLOW (LBM/HR)
TURBINE BYPASS VALVES (Per Valve)		112.5	435.5 K	STEAM	435.5 K
ISOLATION CONDENSER		74.0	248.5 K	STEAM	286.5 K
HPCI	At 155 psia	26.5	102.5 K	STEAM	102.6 K
	At 1125 psia	37.2	145 K	STEAM	144 K
SHUTDOWN COOLING (Per Loop)		7.9	168.5 M	WATER	30.6 K
RWCU NRHX		10.0	3.25 M	WATER	38.7 K
SRVs	ELECTROMATIC (Per Valve)	139.5	540 K	STEAM	540 K
	TARGET ROCK	160.7	622 K	STEAM	622 K

Based on the information provided in the question, the Isolation Condenser would not be able to control reactor pressure due to the amount of energy being dissipated. With this is in mind, the ADS valves would be the only pressure control source able to dissipate the required amount of heat to maintain the current conditions. Additionally, per DOA 1000-01 step D.6 (page 7) states “THEN use one or more of the following ECCS alternatives as directed by the Unit Supervisor to control reactor water temperature/pressure:” with the ADS valves being the only single option with enough capacity to dissipate the required heat load. Based on the above information, choosing the ADS valves was the only correct answer.

FACILITY RESPONSE:

Based on the initial conditions given in the stem of the question, the Isolation Condenser alone would not be enough to continue the current trend of reducing reactor pressure. The information provide in the question indicates that one Bypass Valve is full open. This approximates to 112 MWth of heat removal, per DOA 1000-3. This DOA also provides the MWth capacity for all other available systems as follows:

- Isolation Condenser- 74 MWth
- HPCI – 37 MWth
- ERV- 140 MWth

Also consider that DOP 1300-03, step F.8 delineates "preferred order of systems to be used for RPV pressure control". This is merely a preferred order ONLY and does not reflect the best choice of system for a given set of plant conditions. The question requires the operator to interpret the condition and chose the appropriate system.

In this event ~ 112 MWth of heat capacity needs to be removed. DOP 1300-03, step F.8 preferred system use criteria would be utilized, in order, per below:

- Isolation Condenser (74 MWth) would be evaluated as a first choice and discounted due to insufficient heat removal capacity.
- HPCI in pressure control mode (37 MWth) would be evaluated as a second choice and discounted due to insufficient heat removal capacity.
- ERVs (140 MWth) would be evaluated as the next choice and determined to have a suitable heat removal capacity for the given plant conditions.

The candidates' conclusion that the thermal capacity of the Isolation Condenser is not enough to make up for the loss of the turbine bypass valve going shut is supported by the above facts. These facts also support that the preferred order of DOP 1300-03 system for RPV pressure control does not account for the necessary heat removal capacity to support the continuation of the cooldown in progress. Alternate opening of Electromatic Relief Valve(s) during the five minute intervals is the only available option with the capacity of removing the heat load that will allow a continued cooldown.

The Facility supports changing the correct answer for question # 9 to (D), "alternate opening of Electromagnetic Relief Valve(s) at five minute intervals".

RECOMMENDATION:

ACCEPT ONLY (D) as the correct answer.

CATEGORY 1

UNIT SHUTDOWN

REQUIREMENTS:

Technical Specifications

INDEPENDENT TECHNICAL REVIEW:

Disciplines:	NPPT	RO	**RE/QNE**	CH	RS	I&C	M&ES
Required:	[X]	[X]	**[X]**	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: **DEOP Coordinator**

PLANT OPERATIONS REVIEW COMMITTEE (PORC):

PORC REQUIRED: [ ] YES [X] NO

APPROVAL AUTHORITY:

Shift Operations Superintendent (SOS), or designee

POST PERFORMANCE REVIEWS:

NONE.

TABLE 1
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SRVs	ELECTROMATIC (Per Valve)	139.5	540 K	STEAM	540 K
	TARGET ROCK	160.7	622 K	STEAM	622 K
MSL DRAINS (To Main Condenser)		14.2	55 K	STEAM	55 K
SJAE (Per Train)		2.6	9.88 K	STEAM	9.88 K
OFF GAS PREHEATER (Per Train)		0.2	654	STEAM	654
OFF GAS BOOSTER AIR EJECT (Per Train)		1.7	6613	STEAM	6613
GLAND SEAL STEAM		3.9	15 K	STEAM	15 K
RADWASTE REBOILER		1.6	6 K	STEAM	6 K

CATEGORY 1

MANUAL OPERATION OF THE ISOLATION CONDENSER

REQUIREMENTS:

Technical Specifications.

INDEPENDENT TECHNICAL REVIEW:

Disciplines Required:	NPPT	RO	RE/QNE	CH	RS	I&C	M&ES
	[X]	[X]	[ ]	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: **DEOP Coordinator**

PLANT OPERATIONS REVIEW COMMITTEE (PORC):

PORC REQUIRED [ ] YES [X] NO

APPROVAL AUTHORITY:

Shift Operations Superintendent (SOS), or designee

POST PERFORMANCE REVIEWS:

NONE.

F. 8. The preferred order of systems to be used for RPV pressure control is as follows:

a. Isolation Condenser with makeup from Clean Demin Water from the Isolation Condenser Makeup Pumps.

b. Isolation Condenser with makeup from Clean Demin Water using Clean Demin Pumps.

c. HPCI in pressure control mode using DOP 2300-03, High Pressure Coolant Injection System Manual Startup and Operation.

d. IF operation of Fire Water Suppression is NOT required, THEN Isolation Condenser with fire water makeup via MO 2(3)-4102, SERV WTR VLV.

e. ADS System relief valves, provided RPV level is maintained > - 50 inches on medium range to prevent ECCS initiation AND ADS automatic blowdown.

f. Isolation Condenser with (contaminated) makeup water via 2(3)-1301-500, U2(3) ISOL CDSR CNTAM DEMIN WTR FILL SV.

9. Isolation Condenser vent radiation indicated on panels 902(3)-55 AND 902(3)-56.

a. IF the Isolation Condenser Radiation Monitors indicate > 3 mrem/hr, THEN the Isolation Condenser should be removed from service to prevent an uncontrolled release of radioactive material to environment.

b. A significant rise in vent radiation level could be an indication of:

- Boil off of contaminated condensate in the Isolation Condenser shell side,

OR

- Isolation Condenser tube leakage.

10. IF plant shutdown is required, AND severe fire condition exists, THEN refer to DSSP 0010-01, Determining Safe Shutdown Paths for Extensive Plant Damage, to determine if the Isolation Condenser is to be used.

CATEGORY 1

RESIDUAL HEAT REMOVAL ALTERNATIVES

REQUIREMENTS:

NONE.

INDEPENDENT TECHNICAL REVIEW:

Disciplines	NPPT	RO	RE/QNE	CH	RS	I&C	M&ES
Required:	[X]	[X]	[ ]	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: **DEOP Coordinator**

PLANT OPERATIONS REVIEW COMMITTEE (PORC):

PORC REQUIRED: [X] YES* [ ] NO

* PORC required for changes to actions impacting jumper/bypass installation or removal

APPROVAL AUTHORITY:

Station Manager (SM), or designee (PORC Required)
Shift Operations Superintendent (SOS), or designee (PORC NOT required)

POST PERFORMANCE REVIEWS:

NONE.

NOTE

1. The Turbine Bypass valves will automatically close when the Main Condenser vacuum drops below seven (7) inHg.

2. IF Steam Jet Air Ejector System is in service, THEN Turbine Bypass valves should be closed WHEN reactor water temperature is at approximately 350°F.

3. IF Mechanical Vacuum Pump is in service, THEN Turbine Bypass valves should be closed WHEN reactor water temperature is approximately 300°F.

4. At 300°F, reactor pressure will NOT be adequate to maintain Turbine Seal Steam System operation.

D. 5. b. Main Steam Turbine Bypass Valves. (Each Main Steam Turbine Bypass Valve will remove up to 112 MWth.)

(1) Maintain a maximum cooldown rate of 100°F in any one hour period using the Bypass Valve opening Jack per DGP 02-01 (Tech Spec Section 3.4.9).

(2) Maintain reactor water level using Feedwater and Condensate Systems (DGP 02-01).

(3) IF Steam Jet Air Ejector System is in service, THEN close the turbine bypass valve(s) WHEN reactor water temperature is approximately 350°F (approximately 125 psig).

(4) IF Steam Jet Air Ejector System is NOT in service, THEN close the turbine bypass valve(s) when reactor water temperature is approximately 300°F.

(5) WHEN reactor water temperature drops to 300°F, THEN open the condenser vacuum breaker using MO 2(3)-4901, TURB VACUUM BKR.

D. 6. IF the above alternatives are NOT sufficient/available to control reactor water temperature/pressure, THEN use one or more of the following ECCS alternatives as directed by the Unit Supervisor to control reactor water temperature/pressure:

a. Isolation Condenser System. (Isolation Condenser System will remove up to 74 MWth.)

(1) Place Isolation Condenser System in service (DOP 1300-03).

b. High Pressure Coolant Injection (HPCI) System. (HPCI will remove up to 37 MWth.)

(1) IF reactor pressure is above 90 psig, THEN initiate HPCI System in pressure control mode (DOP 2300-03).

NOTE

The Suppression Pool water level should be above six (6) feet to ensure exhausted steam from the Electromatic Relief Valves is condensed in the Suppression Pool water.

c. Electromatic Relief Valves. (Each Electromatic Relief Valve will remove approximately 140 MWth.)

(1) Verify Suppression Pool water level is > 6 feet.

(2) Place LPCI in Suppression Pool cooling (DOP 1500-02).

(3) Open one or more Electromatic Relief Valve(s) as necessary to reduce reactor pressure/temperature, while maintaining the cooldown rate below 100°F/hr.

(4) Alternate opening of Electromatic Relief Valve(s) at five (5) minute intervals, in the following sequence to minimize local torus water heating when possible: A, C, E, D, B.

Question # 015

With Unit 3 in Mode 1 operations, the torus conditions are as follows:

- Torus water temperature is 92°F and rising slowly
- Torus narrow range water level is -5.0 inches and lowering slowly

Should a LOCA occur during these conditions, what is the FIRST concern operators would have for primary containment?

a. Incomplete steam condensation.

b. Insufficient scrubbing of iodine from steam discharged during a LOCA.

c. Condensate oscillation and chugging loads.

d. Excessive clearing loads from steam discharges and pool swell could result in damage to the torus and its supports.

Answer A

CANDIDATE FEEDBACK (DOCKET NO. 055- 33736):

The question asks what the first concern would be during a LOCA with Torus Temp being 92 degrees F and rising slowly and Torus level being -5.00 inches are dropping slowly. The answer selected was Inadequate Steam Condensation. Per the Tech Spec Bases for 3.6.2.2, the background section states the following: "If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve quenchers, downcomer lines, or HPCI turbine exhaust line. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified." However, it later states in Actions A.I "With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as the downcomers are covered, HPCI turbine exhaust is covered, and relief valve quenchers are covered." Based on the current conditions provided in the question, the downcomers , HPCI turbine Exhaust, and Relief valve quenchers are still covered (level is only ½ inch outside of the allowable band).

Prior to the LOCA, DEOP 200-01 would have been entered based on the torus level of -5.0 inches and dropping slowly. Without further information (i.e. source of dropping level), it is plausible Torus make up would have been initiated per DOP 1600-02 step G.2 per the first step in DEOP 200-01 based on this entry condition.

When the LOCA occurs, the SRO would eventually enter DEOP 100 and reenter DEOP 200-01 (due to rising drywell pressure and subsequent scram that would occur.) When DEOP 200-01 is re-entered all 5 legs are re-entered concurrently. The LOCA will cause Drywell pressure to rise at some rate (dependant on size of the leak). With rising drywell pressure, the SRO would prioritize entry into the Primary Containment Pressure Leg of DEOP 200-01 since torus level would already be addressed by the initial entry of DEOP 200-01. With an active LOCA, the first step of DEOP 200-01 Primary Containment pressure leg would not be utilized since you have changing conditions within the Drywell and would need activity samples prior to venting in this step. The SRO would direct initiation of Torus Sprays to try and control drywell pressure, however the pressure suppression function still exists in this condition and there should be no bypass flow with Torus level at approximately 14 feet, so Torus Sprays would have little effect on reducing Drywell Pressure. Drywell Pressure would continue to rise. Due to this continued rise the concern would be chugging that would occur at 9 psig. Based on the above information, it is plausible that chugging would the First Concern as well as lack of Steam Condensation as selected for the correct answer.

FACILITY RESPONSE:

Question is acceptable as written for this exam. The assumptions made in the feedback from the candidate would occur sometime after the conditions given in the stem of the question and therefore would not be the FIRST concern as stated in the call of the question.

RECOMMENDATION:

ACCEPT ONLY (A) as the correct answer.

Question # 018

Unit 3 was operating at rated power when a loss of coolant accident occurred that caused a fuel element failure. Coincident to this, containment has failed.

If members of the public downwind were to receive an acute dose of 150 rem, what biological effects are expected to occur?

1. Death (to 50% of the population).
2. Slight decrease in blood cell count.
3. Nausea/vomiting to <50% of population within 3 hours.
4. Loss of hair after 2 weeks.

a. 2 ONLY

b. 2 AND 3 ONLY

c. 2, 3 AND 4 ONLY.

d. 1, 2, 3 AND 4.

Answer: B

CANDIDATE FEEDBACK (DOCKET NO. 055- 61716):

Question #18, Hair loss – 75 REM by the EPA

This table is excerpted from US EPA Website using the following address: "http://www.epa.gov/radiation/understand/health_effects.html"

Exposure (rem)	Health Effect	Time to Onset (without treatment)
5-10	changes in blood chemistry	
50	nausea	hours
55	fatigue	
70	vomiting	
75	hair loss	2-3 weeks
90	diarrhea	
100	hemorrhage	
400	possible death	within 2 months
1,000	destruction of intestinal lining	
	internal bleeding	
	and death	1-2 weeks
2,000	damage to central nervous system	
	loss of consciousness;	minutes
	and death	hours to days

FACILITY RESPONSE:

Question is acceptable as written for this exam. Answer provided is in alignment with Exelon NGET study material and NRC Reg Guide 8.29.

RECOMMENDATION:

ACCEPT ONLY (B) as the correct answer.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.29

(Draft was issued as DG-8012)

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

A. INTRODUCTION

Section 19.12 of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," requires that all individuals who in the course of their employment are likely to receive in a year an occupational dose in excess of 100 mrem (1 mSv) be instructed in the health protection issues associated with exposure to radioactive materials or radiation. Section 20.1206 of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that before a planned special exposure occurs the individuals involved are, among other things, to be informed of the estimated doses and associated risks.

This regulatory guide describes the information that should be provided to workers by licensees about health risks from occupational exposure. This revision conforms to the revision of 10 CFR Part 20 that became effective on June 20, 1991, to be implemented by licensees no later than January 1, 1994. The revision of 10 CFR Part 20 establishes new dose limits based on the effective dose equivalent (EDE), requires the summing of internal and external dose, establishes a requirement that licensees use procedures and engineering controls to the extent practicable to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA), provides for planned special exposures, establishes a dose limit for the embryo/fetus of an occupationally exposed declared pregnant woman, and explicitly states that Part 20 is not to be construed as limiting action that may be necessary to protect health and safety during emergencies.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 19 or 10 CFR Part 20. These regulations provide the regulatory bases for this guide. The information collection requirements in 10 CFR Parts 19 and 20 have been cleared under OMB Clearance Nos. 3150-0044 and 3150-0014, respectively.

B. DISCUSSION

It is important to qualify the material presented in this guide with the following considerations.

The coefficient used in this guide for occupational radiation risk estimates, 4×10^{-4} health effects per rem, is based on data obtained at much higher doses and dose rates than those encountered by workers. The risk coefficient obtained at high doses and dose rates was reduced to account for the reduced effectiveness of lower doses and dose rates in producing the stochastic effects observed in studies of exposed humans.

The assumption of a linear extrapolation from the lowest doses at which effects are observable down to

3. **What is meant by early effects and delayed or late effects?**

EARLY EFFECTS

Early effects, which are also called immediate or prompt effects, are those that occur shortly after a large exposure that is delivered within hours to a few days. They are observable after receiving a very large dose in a short period of time, for example, 300 rads (3 Gy) received within a few minutes to a few days. Early effects are not caused at the levels of radiation exposure allowed under the NRC's occupational limits.

Early effects occur when the radiation dose is large enough to cause extensive biological damage to cells so that large numbers of cells are killed. For early effects to occur, this radiation dose must be received within a short time period. This type of dose is called an acute dose or acute exposure. The same dose received over a long time period would not cause the same effect. Our body's natural biological processes are constantly repairing damaged cells and replacing dead cells; if the cell damage is spread over time, our body is capable of repairing or replacing some of the damaged cells, reducing the observable adverse conditions.

For example, a dose to the whole body of about 300–500 rads (3–5 Gy), more than 60 times the annual occupational dose limit, if received within a short time period (e.g., a few hours) will cause vomiting and diarrhea within a few hours; loss of hair, fever, and weight loss within a few weeks; and about a 50 percent chance of death if medical treatment is not provided. These effects would not occur if the same dose were accumulated gradually over many weeks or months (Refs. 1 and 2). Thus, one of the justifications for establishing annual dose limits is to ensure that occupational dose is spread out in time.

It is important to distinguish between whole body and partial body exposure. A localized dose to a small volume of the body would not produce the same effect as a whole body dose of the same magnitude. For example, if only the hand were exposed, the effect would mainly be limited to the skin and underlying tissue of the hand. An acute dose of 400 to 600 rads (4–6 Gy) to the hand would cause skin reddening; recovery would occur over the following months and no long-term damage would be expected. An acute dose of this magnitude to the whole body could cause death within a short time without medical treatment. Medical treatment would lessen the magnitude of the effects and the chance of death; however, it would not totally eliminate the effects or the chance of death.

DELAYED EFFECTS

Delayed effects may occur years after exposure. These effects are caused indirectly when the radiation changes parts of the cells in the body, which causes the normal function of the cell to change, for example, normal healthy cells turn into cancer cells. The potential for these delayed health effects is one of the main concerns addressed when setting limits on occupational doses.

A delayed effect of special interest is genetic effects. Genetic effects may occur if there is radiation damage to the cells of the gonads (sperm or eggs). These effects may show up as genetic defects in the children of the exposed individual and succeeding generations. However, if any genetic effects (i.e., effects in addition to the normal expected number) have been caused by radiation, the numbers are too small to have been observed in human populations exposed to radiation. For example, the atomic bomb survivors (from Hiroshima and Nagasaki) have not shown any significant radiation-related increases in genetic defects (Ref. 3). Effects have been observed in animal studies conducted at very high levels of exposure and it is known that radiation can cause changes in the genes in cells of the human body. However, it is believed that by maintaining worker exposures below the NRC limits and consistent with ALARA, a margin of safety is provided such that the risk of genetic effects is almost eliminated.

4. **What is the difference between acute and chronic radiation dose?**

Acute radiation dose usually refers to a large dose of radiation received in a short period of time. Chronic dose refers to the sum of small doses received repeatedly over long time periods, for example, 20 mrem (or millirem, which is 1-thousandth of a rem) (0.2 mSv) per week every week for several years. It is assumed for radiation protection purposes that any radiation dose, either acute or chronic, may cause delayed effects. However, only large acute doses cause early effects; chronic doses within the occupational dose limits do not cause early effects. Since the NRC limits do not permit large acute doses, concern with occupational radiation risk is primarily focused on controlling chronic exposure for which possible delayed effects, such as cancer, are of concern.

The difference between acute and chronic radiation exposure can be shown by using exposure to the sun's rays as an example. An intense exposure to the sun can result in painful burning, peeling, and growing of new skin. However, repeated short exposures provide time for the skin to be repaired between exposures. Whether exposure to the sun's rays is long term or spread over short periods, some of the injury may not be repaired and may eventually result in skin cancer.

Cataracts are an interesting case because they can be caused by both acute and chronic radiation. A certain threshold level of dose to the lens of the eye is required before there is any observable visual impairment, and the impairment remains after the exposure is stopped. The threshold for cataract development

Question # 032

Which of the following combinations of ECCS subsystems will ensure adequate core cooling during a DBA LOCA?

a. One LPCI subsystem

b. One Core Spray subsystem

c. One Core Spray subsystem AND the 5 ADS valves

d. One Core Spray Subsystem AND one LPCI subsystem

Answer: D

CANDIDATE FEEDBACK (DOCKET NO. 055- 33736):

The correct answer to the question was one LPCI subsystem and one Core Spray subsystem per UFSAR Section 6.3.3.1.1 (page 6.3-24). Per DEOP 0010, adequate core cooling is defined as "heat removal from the reactor sufficient to prevent rupturing the fuel clad. Three viable mechanisms for establishing adequate core cooling exist - core submergence, spray cooling and steam cooling. Adequate spray cooling is "is provided, assuming a bounding axial power shape, when design spray flow requirements are satisfied and RPV water level is at or above the elevation of the jet pump suctions (Core Spray flow > 4750 gpm AND reactor water level >-191 inches). The covered portion of the core is then cooled by submergence while the uncovered portion is cooled by the spray flow." Additionally, in section 6.3.3.3.2 (page 6.3-60) of the UFSAR "Long term cooling requirements for a large break are met by either: 1) supplying 4500 gpm of core spray flow to the top of the core and maintaining 2/3 core height" Based on the above information, one division of Core spray is an acceptable answer to the question as well.

FACILITY RESPONSE:

Question is acceptable as written for this exam. The statements made in DEOP 0010 do not consider a DBA LOCA when describing viable mechanisms for core cooling. The correct answer is support by Dresden Station FSAR section 6.3

RECOMMENDATION:

ACCEPT ONLY (D) as the correct answer.

6.3.2.2.1 Low Pressure Coolant Injection Subsystem Interface with Other ECCS Subsystems (Historical)

Under current regulatory requirements, the function of the LPCI subsystem is to ensure adequate core cooling across the entire spectrum of line break accidents when operated with other available ECCS subsystems determined from the Appendix K single active failure criterion. Tables 6.3-21b and 6.3-21c list the ECCS equipment available under different postulated single active failure considered in Appendix K ECCS performance evaluation (see Section 6.3.3). This section presents historical information on the interface between the LPCI subsystem and other ECCS subsystems under the original design. The information is historical and is not needed to support the current design basis.

The LPCI subsystem operates in conjunction with the HPCI and core spray subsystems to achieve its core cooling function. During a loss-of-coolant accident, coolant is lost from the core with a corresponding decrease in reactor vessel pressure. The HPCI subsystem operates initially during the high-pressure phase of the accident to supply a small amount of coolant at high pressure.

As the pressure in the reactor vessel decreases, the HPCI subsystem flow ceases and the core spray and LPCI subsystems automatically begin operation to take over the core cooling function. When the pressure in the reactor vessel equals the pressure in the suppression chamber, the LPCI subsystem is capable of delivering maximum capacity. LPCI delivers rated flow with reactor pressure equal to 20 psid (differential pressure between the reactor vessel dome and the drywell). After the core has been flooded to two-thirds height, only one LPCI pump is required to maintain this level.

6.3.2.2.2 Subsystem Characteristics

The LPCI subsystem is required to inject sufficient makeup water to reflood the vessel to the appropriate core height to provide adequate core cooling and is later required to maintain the level at two-thirds core height. The DBA LOCA analyses take credit for operable LPCI pumps as indicated in Table 6.3-21c for SVEA-96 Optima2 fuel. These analyses also require operation of at least one core spray subsystem to ensure adequate core cooling. To assure long-term cooling of the fuel, the minimum requirements of 4500 gpm of core spray flow to the top of the core and two-thirds core height water level or full height water level coverage to the top of active fuel should be satisfied. The pump head characteristic was selected such that sufficient but less than rated flow is provided before the HPCI turbine is tripped by low vessel pressure. This approach provides core cooling over the complete spectrum of breaks up to the design basis break. The specifications for these pumps are shown in Table 6.3-5 and the performance curve for the LPCI subsystem is shown in Figure 6.3-8.

The two LPCI pumps for each LPCI subsystem are located on the basement floor in shielded rooms in each of two corners of the reactor building. Each LPCI pump room has the necessary piping and instrumentation to perform in any LPCI or containment cooling mode of operation (refer to Section 6.2.2 for a description of containment cooling functions by the LPCI subsystem). A crosstie header between the otherwise separate subsystems makes it possible for the LPCI pumps in one room to deliver their flow through the second loop's piping. This crosstie is located in a well-protected basement floor area and has two normally keylocked open, motor-operated valves. The valves may be closed from the control room if loop isolation is necessary. Separation of the piping provides protection against missiles in the vicinity of the reactor in that only one of the two flow paths must be assumed to be incapacitated by missiles. Missile protection shielding is provided by routing piping along the reactor building structure walls as much as possible.

Each of the two LPCI pump rooms contains its own room cooler and associated fan. Cooling water is normally provided by the service water system, with the containment cooling service water system as a backup, as described in Section 9.2.

The LPCI room cooler fan motors are seismically supported from rod hangers in a pendulum fashion from the ceiling of the reactor building at elevation 517'-6".

The LPCI subsystem is protected from plugging (due to the presence of foreign material which may find its way into the suppression chamber) by the use of multiple suction header connections with strainers. An evaluation of the strainers is provided in Section 6.2.2.

The LPCI pump motors are cooled by use of the LPCI pump discharge water which is routed through a 1-inch line to the pump motor oil coolers and returned to the suction side of the pump. The pump discharge water cools the oil that in turn lubricates and cools the motor. Hence, no external power source is required other than that used to power the pump motors.

During post-accident operation one or more of the LPCI pumps may be used for containment spray mode. The containment long-term response analysis for a DBA LOCA assumes that a minimum of 5000 gal/min is provided by LPCI to cool the containment (refer to Section 6.2.1.3.2.2). Additionally, to meet two-thirds core coverage requirements, continued post-LOCA ECCS flow is required to make up shroud leakage as described by Section 6.3.2.2.3.1. Therefore, post-DBA LOCA operation of the ECCS system must address both core coverage and containment spray requirements. The long term post-LOCA analysis employed the containment spray mode in order to yield the lowest containment pressure over the long term. Note that containment spray mode is not the preferred method of long term cooling, but was assumed in the analysis since it yields the lowest containment pressures and is therefore conservative in determining ECCS pump NPSH requirements.

After a period not exceeding 2 hours, the operator can manually stop one LPCI pump and start the two containment cooling service water pumps powered by the same diesel generator (assuming offsite power is not available). Operating procedures have the operator start the two CCSW pumps in less than this period of time. This pump sequencing serves to limit the load imposed on a diesel generator. Utilization of the LPCI subsystem in conjunction with containment cooling service water would achieve the containment cooling capability as specified in Section 6.2.1.3.2.2.

During LPCI subsystem operation, water is normally taken from the suppression pool and is pumped into the core region of the reactor vessel via one of the two recirculation loops. There is also a connection to the contaminated condensate storage tanks to make condensate available as a backup supply. In the event of a recirculation line break, instrumentation is provided to select the undamaged flow path for injection of the required LPCI flow into the reactor vessel. This instrumentation also causes appropriate valves to close which could otherwise result in spillage of the LPCI flow from the shroud region. The sensing circuit for break detection and loop selection is arranged so that failure of a single device or circuit to function on demand will not prevent correct selection of the loop for injection. All components of the loop selection network are operated from dc power sources so that loop selection can take place irrespective of the availability of ac power. LPCI loop selection logic and instrumentation is described in Section 7.3.

Question # 043

Unit 3 was operating at rated power when a transient occurred, resulting in the following conditions:

- RPV water level is -72 inches and trending up.
- The TR-32 Sudden Pressure Relay (SPR) activated.
- The Unit 3 EDG started but its output breaker subsequently tripped on over-current.

The Unit Supervisor has directed the crew to enter and execute DGA-12, PARTIAL OR COMPLETE LOSS OF AC POWER.

The required electrical lineup is to power Bus 33-1 from __(1)__ and Bus 34-1 from __(2)__ .

	(1)	(2)
a.	Bus 23-1;	U3 SBO
b.	U3 SBO;	Bus 24-1
c.	2/3 EDG;	U3 SBO
d.	2/3 EDG;	Bus 24-1

Answer: C

CANDIDATE FEEDBACK (DOCKET NO. 055- 61716):

What caused the EDG output breaker to trip on overcurrent? Was 34-1 overcurrent? You would not power a bus that is overcurrent. No correct answer given.

FACILITY RESPONSE:
With the conditions given in the stem of the question, the Unit 3 EDG has an auto start signal present via Div II Core Spray Logic due to RPV Level below -59 inches. With an auto start signal present, the Unit 3 EDG Auto Start Relay (ASR-3/"HGA") is picked up. This ASR-3 relay, when picked up, has the effect of bypassing all trips of the Unit 3 EDG output breaker, with the exception of Differential Current. The trips that are bypassed are: Over-current, Reverse Power, Ground Fault, Loss of Field and Under Frequency. The stem indicates that a Unit 3 EDG output breaker subsequently tripped on over-current which, by Station Design is bypassed under the conditions given in the question.

If the Unit 3 EDG output breaker did trip on over-current when it should not have (i.e., when an auto start signal was present), then the status of Unit 3 EDG output breaker and Bus 34-1 electrical protection scheme is in an unknown and potentially unreliable condition. With Division 1 of AC power energized via the 2/3 EDG concurrent with RPV level well above Top of Active Fuel and rising, the risk of re-energizing this Bus with no pressing Public Health & Safety concern is unacceptable (H.B. Robinson, SOER 10-2 Event). Therefore, the correct course of action would be to energize Bus 33-1 from the 2/3 EDG and leave Bus 34-1 de-energized until evaluated.

The facility supports no correct answer since the stem of the question presents a situation that is not consistent with the design of the facility and not enough information is given for the candidates to make a reasonable assumption as to the location of the fault and the status of the bus.

RECOMMENDATION:

REMOVE QUESTION due to no correct answer.

CATEGORY 1

PARTIAL OR COMPLETE LOSS OF AC POWER

REQUIREMENTS:

NONE.

INDEPENDENT TECHNICAL REVIEW:

Disciplines	NPPT	RO	RE/QNE	CH	RS	I&C	M&ES
Required:	[X]	[X]	[ ]	[ ]	[ ]	[ ]	[X]

Unit 1 Review Required: [ ] YES [X] NO

Special Reviews: NONE.

PLANT OPERATIONS REVIEW COMMITTEE (PORC):

PORC REQUIRED [ ] YES [X] NO

APPROVAL AUTHORITY:

Shift Operations Superintendent (SOS), or designee.

POST PERFORMANCE REVIEWS:

NONE.

D.
- Review EP-AA-111, Emergency Classification and Protective Action Recommendations.
- IF conditions of an Emergency Action Level are met, THEN declare Emergency Classification Level AND implement required notifications per EP-AA-114, Notifications.
- IF recovering from abnormal conditions (e.g. fire, explosion, etc.), THEN verify systems/components to be restored have NOT been adversely impacted and conditions are acceptable for restoration. (W-23)
- IF necessary to provide an alternate makeup source to the RPV, THEN refer to TSG-3, ,Operational Contingency Guidelines. TSG-3 has multiple attachments to provide makeup to the RPV

LOSS OF SINGLE PHASE FROM OFF SITE POWER SUPPLY:

NOTE

IF a TR 32 open phase condition is detected, THEN it will alarm annunciator 903-8 E-4, RES AUX TR32 LOSS OF PHASE-TROUBLE. WHEN EC 388778 second phase has been implemented, THEN this condition will lock out (trip) TR 32.

1. IF multiple components are tripping on overcurrent AND loss of a phase on the AC power supply is suspected, THEN open:
   - TR 22(32) TO BUS 21(31) GCB.
   - TR 22(32) TO BUS 22(32) GCB.
   - TR 22(32) TO BUS 23(33) GCB.
   - TR 22(32) TO BUS 24(34) GCB.

Question # 079

Unit 3 was operating at full power when an event occurred. Later during the accident, the following plant conditions exist:

- HPCI pump is out of service
- LPCI pumps running but NOT injecting to the RPV.
- "A" & "B" Trains of Core Spray are injecting into the RPV at 4000 gpm each.
- RPV Water level is -45 inches and steady.
- Drywell pressure is 7 psig and rising slowly.
- Drywell temperature is 240°F and rising slowly.
- Torus Bulk Temperature is 200°F and steady.
- Torus Level is 14 feet and steady.
- Torus Bottom Pressure is 10.2 psig and rising slowly.

The SRO is performing steps from DEOP 100, "RPV Control" and DEOP 200-01, "Primary Containment Control." To spray the drywell, AND to prevent ECCS pump cavitation the SRO orders…

a. spray with one LPCI pump with flow <2750 gpm.

b. reduce CS flow to <2750 gpm, then inject with one LPCI pump at 4000 gpm.

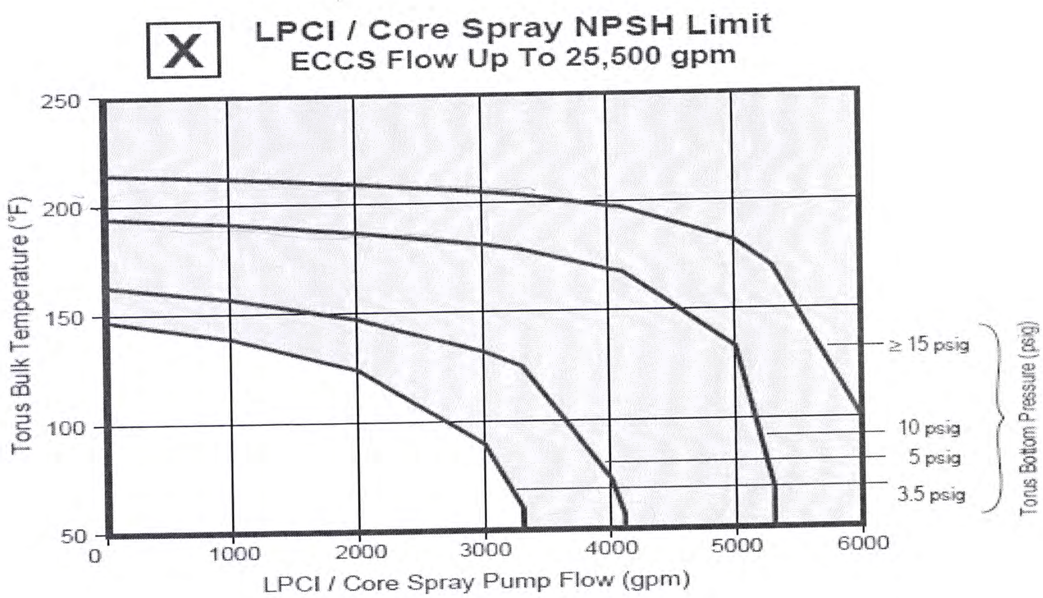
c. the drywell can NOT be sprayed without cavitating existing ECCS pump flow.

d. secure one CS pump before injecting with one LPCI pump, keep LPCI flow <4000 gpm.

Answer: A

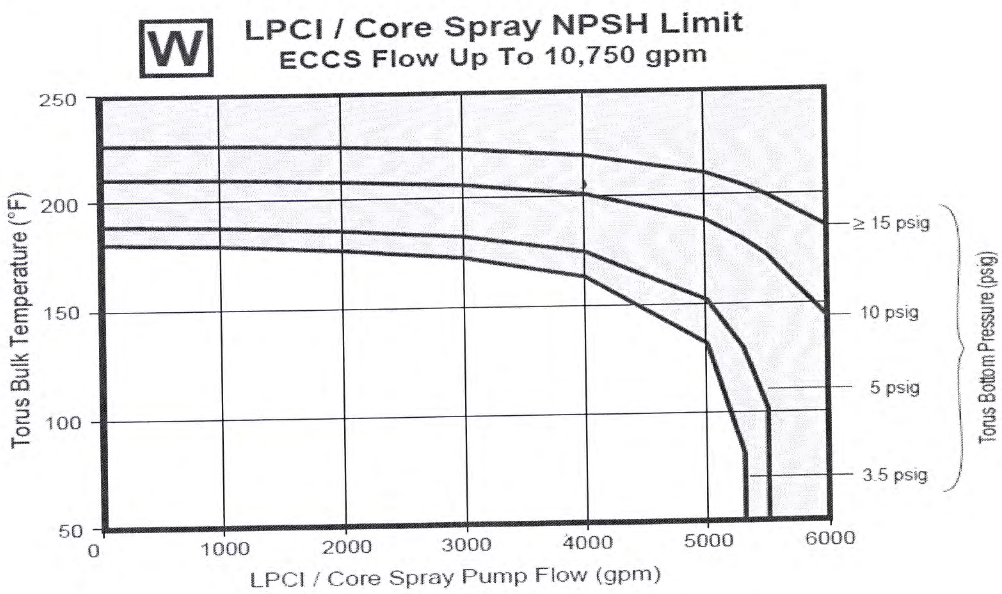
CANDIDATE FEEDBACK (DOCKET NO. 055- 33736):

The correct answer to this question was to throttle LPCI flow to less the 2750 gpm to prevent cavitation. However, based on the information in the question, not being able to avoid cavitation is the correct answer. First, the LPCI Drywell Spray valves (2(3)-1501-27(8)A/B) are not throttable valves and are upstream of the LPCI Injection valves 2(3)-1501-21A/B which are throttable. The only way to throttle LPCI flow in this configuration is to throttle a valve that is not normally used for this purpose (i.e. manually manipulating a LPCI pump discharge valve or the Drywell Spray valves themselves.) There is no procedural guidance to perform this action. The only other way to limit Drywell Spray flow is open other flow paths to divert flow away (i.e. Torus Cooling). Performing this action would increase overall flow to over 10,750 gpm, which causes the use of the X Curve (depicted below).



When the X curve is utilized, there are no pump flows at 200 degrees F Torus Bulk Temperature and 10.2 psig Torus bottom pressure that would prevent cavitation from occurring.

Second, if flow is able to be achieved at 2750 gpm, the SRO would utilize the W curve depicted below.



At initial conditions, 200 degrees F Torus Temperature and 10.2 psig Torus Bottom Pressure, no cavitation would occur the instant you put on drywell sprays, however due to the evaporative cooling that would occur with initiation drywell sprays, pressure would rapidly reduce to below 10 psig. As soon as you drop below 10 psig torus bottom pressure, you would then transfer to the 5 psig line on curve W since interpolation is not allowed. With the 5 psig torus bottom pressure being your limiting curve, and Torus temperature of 200 degrees F, there is no way to prevent cavitations from occurring since there are no flows exists within the curve at this temperature.

FACILITY RESPONSE:

As administered, the correct answer (A) to this question was to spray the DW with LPCI pump flow less than 2750 gpm to prevent cavitation. The candidate is correct that the drywell spray valves utilized to establish these conditions are either full open or full close valves and do not have throttle capability.
In addition, the Training staff utilized the plant reference simulator to determine if different LPCI system lineups and/or pump combinations would facilitate initiation of DW spray to meet the condition of 2750 gpm flow. The best possibility of achieving such conditions would require the DW spray function to be limited to one division of DW Spray, with the LPCI cross tie valves closed, and ONLY one LPCI Pp running in that division. Under these conditions, DW Spray was still in excess of 4000 gpm. This amount of flow, when coupled with the Core Spray injection flow of 8000 gpm, now risks ECCS pump cavitation per DEOP 200-1 Figure W.

Also note that the lineup of a single LPCI pump in the division performing the DW Spray function is a contradiction to the information provided that states “LPCI Pps are running but NOT injecting into the RPV”. This adds further support to the position of the candidate.

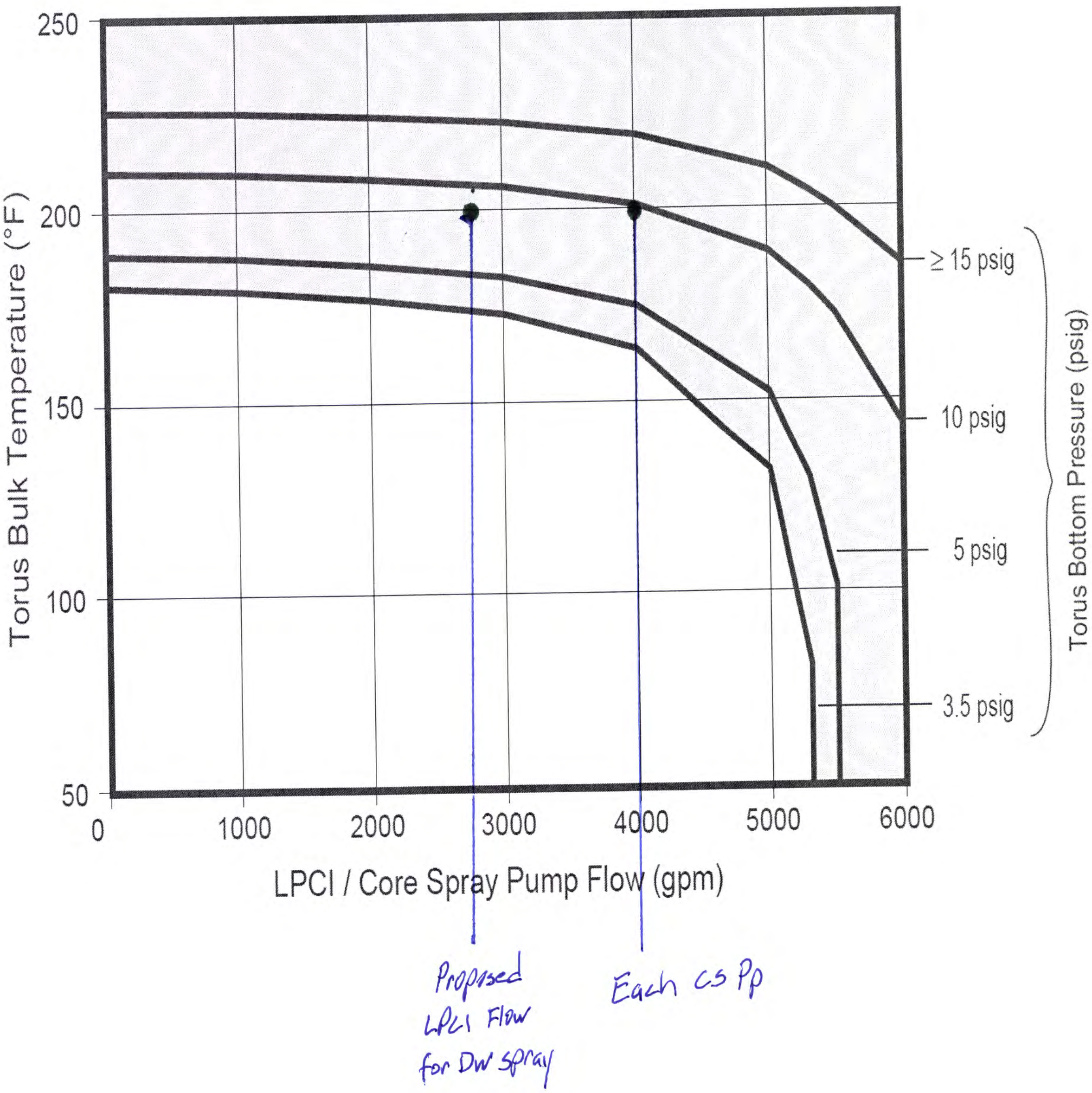
Based on the above findings, no method exists to spray the DW that would maintain a necessary margin to prevent ECCS pump cavitation (reference DEOP 200-1 Figure W). As a result, selection (C) is the only possible correct answer. The station supports the position of the candidate and recommends changing the correct answer for Question # 79 to be (C), “the drywell can NOT be sprayed without cavitating existing ECCS pump flow”.

RECOMMENDATION:

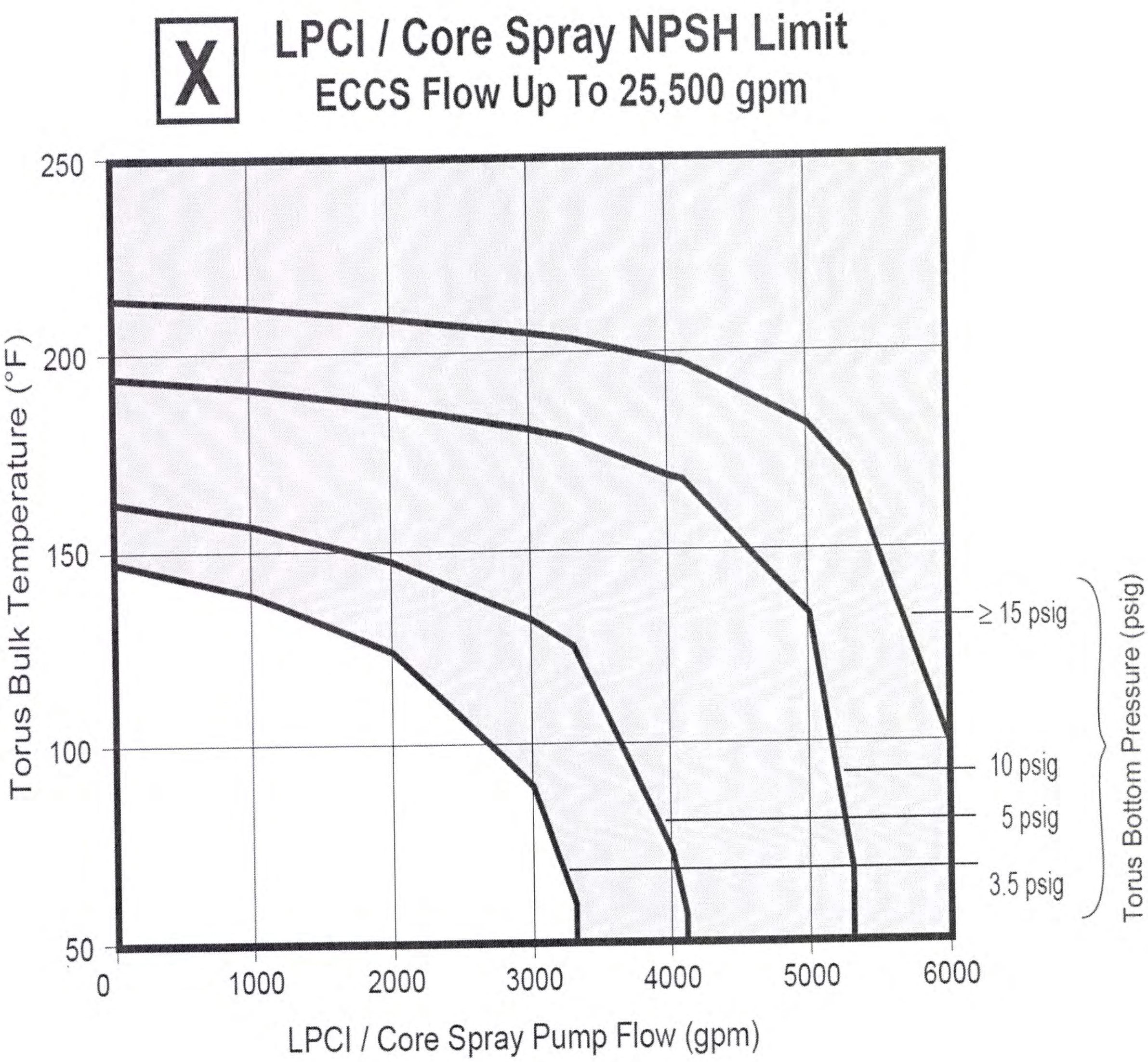
ACCEPT ONLY (C) as the correct answer.

W LPCI / Core Spray NPSH Limit

ECCS Flow Up To 10,750 gpm



The combination of Torus Temp & Torus Bottom Pressure precludes use of this graph.



Question # 098

As the SRO for a shift, you are in the third quarter and have to decide which NSO can substitute for an NSO who had to leave in the middle of your shift. Your present NSO has a "no-solo" license with no other restrictions. From this available list provided, who is eligible for the 2nd NSO position on your crew?

John –	reactivated his license in the 1st quarter, but did NOT stand a watch as an NSO in the 2nd quarter
James –	has reported to the medical staff last week a condition requiring a reduction of dosage of a prescription drug he is currently taking. A license change has been submitted to the NRC for review.
Julia –	has a "no-solo" license and is a declared pregnant worker.

a. Julia ONLY

b. James ONLY

c. James and Julia.

d. John & James.

Answer: C

CANDIDATE FEEDBACK (DOCKET NO. 055- 33736):

The correct answer to the question was that both James and Julia were able to assume the shift. However, choosing just Julia is also a correct answer. On ES-605 page 10, the following is stated: "Physician prescribed changes in medication or dosing for an existing medical condition are not required to be reported to the NRC, unless the examining physician believes the operator's medical condition has become unstable (therefore requiring follow-up medical status reports to the NRC) or that operator requires a no-solo license restriction. Additionally, per OP-AA-105-101 section 4.6, a licensed individual is required to "report the use of prescription or over the counter medications, other than aspirin, aspirin substitute, antibacterial, and birth control to their immediate supervisor and OHS in accordance with SY-AA-102-106." OP-AA-105-101 states further in 4.6.5.1 that "OHS shall **EVALUATE** information provided by the Licensee, and based on the evaluation may place the Licensee's license on "Administrative Hold" pending further evaluation of the condition." Since the change in prescription was reported to the NRC, it is plausible that James has been put on Administrative Hold (defined as an administrative restriction placed upon a NRC Licensed Operator by OHS restricting the licensee from performing licensed duties pending further evaluation of a "health status change".) Further information would be required (i.e. what the medication was, why is it changed, etc) to fully ascertain whether James could assume the shift or he had been placed on Administrative Hold. Based on this information, picking Julia is also a plausible and correct answer.

FACILITY RESPONSE:

The question provides inadequate information concerning the required OHS evaluation for an administrative hold. To select the correct answer (C), the candidates must make an assumption that "James" is not on administrative hold AND that his condition has been evaluated by the Station's Occupation Health Services (OHS) representative.

OP-AA-105-101 (Administrative Process for NRC License and Medical Requirements) and the Dresden OHS representative were utilized for additional insights. OHS indicated that there are reasonable cases where a reduction in dosage of a medication would require a "stabilizing period" to ensure that any effects would not adversely impact the Operators ability to perform shift functions. This insight further supports the candidates' position that the OP-AA-105-101, step 4.6.5.1 requirement to evaluate this condition may already preclude this individual from being able to perform shift functions.

Based on the above findings, there are clear circumstances where both (A) and (C) are correct. Selections (B) and (D) are valid distracters and can never be true due to inclusion of "John" who holds an inactive license. The station supports the position of the candidate in adjusting the answer for Question # 98 to indicate both (A) and (C) as correct choices.

RECOMMENDATION:

ACCEPT BOTH (A or C) as the correct answer.

ADMINISTRATIVE PROCESS FOR NRC LICENSE AND MEDICAL REQUIREMENTS

1. PURPOSE

1.1. This procedure describes the administrative process for United States Nuclear Regulatory Commission (NRC) licenses, including initial license applications, license renewal, Biennial Medical Examinations, and updates to the NRC to report changes in an individual's license status.

2. TERMS AND DEFINITIONS

2.1. **Action Tracking / Action Tracking Item:** Refers to the formal program used by the site to track performance of specific action items and commitments. Examples of formal programs include but are not limited to Passport, PIMS, etc. Within this context, the "owed to" individual is the on-site individual for whom the action or commitment is being completed.

2.2. **"Administrative No Solo"** An administrative restriction placed upon a NRC Licensed Operator by OHS pending further review of a "health status change".

2.3. **"Administrative Hold":** An administrative restriction placed upon a NRC Licensed Operator by OHS restricting the licensee from performing licensed duties pending further evaluation of a "health status change".

2.4. **Annual:** Once per calendar year. For example, an annual test last performed in January 1995, would be due again by December 31, 1996.

2.5. **Applicant:** Person applying for a NRC Reactor Operator, Senior Reactor Operator, or Senior Reactor Operator - Limited license.

2.6. **Biennial Medical Examination:** The medical examination given every 2 years, required by the NRC for all licensed individuals. For purposes of the medical examination, "biennial" is a period of time equal to 730 days and synonymous with the term "two years". Biennial medical examination requirements can extend beyond 730 days if the requirement is met during the anniversary month of the second year. For example, a Biennial Medical Examination last performed on January 10, 1995, would be due again by January 31, 1997. January is seen as the anniversary month, the period of time between the two examinations is longer than 730 days, but the biennial requirement is satisfied. This medical examination is required for **ALL** NRC licensed individuals (active and inactive license status).

2.7. **Certificate of Medical History (CMH):** Health history completed by the applicant. See HR-AA-07-101, Licensed Nuclear Operator Medical Examination - Attachment 2 for a copy of the form.

4.6.3. The immediate Supervisor or the Operations Support Manager shall **INITIATE** Attachment 3 when a change in health condition that affects or has the potential to affect license status occurs.

4.6.4. The License Coordinator shall **CREATE** an Action Tracking item, with sub-assignments as appropriate, owed to the Operations Support Manager to track proper reporting of the status change.

4.6.5. Licensee shall **NOTIFY** OHS of the change in health status prior the next scheduled shift.

1. OHS shall **EVALUATE** information provided by the Licensee, and based on the evaluation may place the Licensee's license on "Administrative Hold" pending further evaluation of the condition.

    A. OHS shall **NOTIFY** the Licensee, and the License Coordinator if an individual's license is placed on "Administrative Hold".

    B. OHS shall **NOTIFY** the Operations Support Manager to remove the individual from license duties.

2. Licensee shall **PROVIDE** follow-up information to OHS as requested, by the date specified by OHS.

4.6.6. Changes in license status must be reported to the NRC within 30 days. The License Coordinator shall **NOTIFY** Regulatory Assurance to develop the letter required for NRC notification.

1. **If** site Regulatory Assurance disagrees with or questions the reportability of medical information provided by OHS, **then REFER** the disagreement / question to the Nuclear OHS Manager and Corporate Licensing for resolution.

4.6.7. **When** the required correspondence has been sent to the NRC providing notification of the status change, **then** the Operations Support Manager shall **CLOSE** the Action Tracking item created above.

4.6.8. Regulatory Assurance shall **ROUTE** a signed copy of the NRC Form 396 to OHS to file in the Licensee's medical file.

4.6.9. The License Coordinator will **NOTIFY** the DTC of any changes in qualifications.

4.7. <u>Reporting Changes in License Status (refer to Attachment 4)</u>

4.7.1. Changes in NRC license status can result from any of the following:

- Permanent Licensee disability due to a physical or mental condition
- Permanent reassignment to a position not requiring a license
- Termination of Licensee