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September 10, 1997
BVY 97-114
TDL 97-028

Regional Administrator, Region I
ATTN: Glenn Meyer
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pa. 19406-1415

References: (a) License No. DPR-28 (Docket No. 50-271)

Subject: Comments on NRC SRO Written Examination

Attachments:
(1) Question 59 Comments and References
(2) Question 69 Comments and References
(3) Question 74 Comments and References
(4) Question 90 Comments and References

In accordance with NUREG 1021, "Operator Licensing Examination Standards for Power Reactors," ES 402, "Administering Initial Written Examinations," Vermont Yankee personnel have reviewed the SRO Initial Licensed Operator written examination that was administered on September 2, 1997. Attached are four (4) questions with facility comments.

If you have any further questions, please contact Mr. Scott T. Brown, Operations Training Supervisor, in our Brattleboro office at (802) 258-4163.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Francis J. Helin for

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ATTACHMENT 1

Exam Question 59

Comments: This question relates to core flow indications following a recirc pump trip and closure of the isolation valve.

There are two issues relative to this question:

(1) The wording in the question stem, "How is core flow determined?" can be interpreted in two different ways. The question can either be interpreted:

a. "how does the circuitry determine core flow?"

OR

b. "how does the operator determine core flow based upon available indications?"

(2) The lesson plan taught to this class (previous revision) [REF: LOT-00-202 Rev. 14 Section IV.B.4 (page 46 of 59)] incorrectly identified the loop flow indicators and recorders on CRP 9-4 as indicating the sum of jet pump flows associated with the individual recirc loop. The lesson plan was revised to provide correct information (Revision 15 06/03/97); however, this was well after the class had received systems training on the recirculation system (01/07/97).

Irrespective of the classroom training, if the question is interpreted as indicated in (1)a. above, a conclusion could be drawn that "By subtracting Loop "A" flow on CRP 9-4 from Total Core flow on CRP 9-5" [Answer B] is the correct answer.

If the question is interpreted as indicated in (1)b. above, "Directly from Total Core Flow recorder on CRP 9-5" [Answer D] is the correct answer [REF: LOT-00-202 Rev 14. Sections IV.B.4 and 5 (pages 46-47 of 59)].

Recommendation: **Accept both B and D** as correct answers.

QUESTION: 059 (1.00)

The plant is operating at 100% reactor power when the "A" reactor recirc pump trips. The operator closes the "A" recirc pump discharge valve.

HOW is core flow determined?

a. Directly from Loop "B" flow indication on CRP 9-4.

b. By subtracting Loop "A" flow on CRP 9-4 from Total Core flow on CRP 9-5.

c. By adding Loop "A" flow on CRP 9-4 and Total Core flow on CRP 9-5.

d. Directly from Total Core Flow recorder on CRP 9-5.

OUTLINE	NOTES
3. Four jet pumps, one in each quadrant are individually instrumented (1, 6, 11, and 16)	
a. The individual D/P instruments are calibrated for flow prior to pump installation	
b. Allows for calibration of all jet pump flow indicators, using these as a reference	
4. Total core flow is determined as follows with both recirc loops operating:	**TRANSPARENCY 7** **TRANSPARENCY 7a** **TRANSPARENCY 7b**
a. Sums the flow from the five jet pumps in each quadrant	
b. Sums the flow from each quadrant associated with one recirc loop, displays this flow for each recirc loop on CRP 9-4 meters and CRP 9-4 recorder	
c. Sums the flow signal from each recirc loop recorder and displays this flow as total core flow on the 9-5	
5. Total core flow is determined as follows with less than two recirc loops operating:	
a. If the recirc pump discharge valve comes off its open seat <u>or</u> the MG set field breaker is open, the idle loop's flow signal is subtracted from the operating loop's flow signal	Necessary since reverse flow will provide a d/p signal that appears to be forward flow
b. Subtracts core bypass flow to develop a total jet pump flow that represents "real" core flow	Core flow indicator would be high

OUTLINE	NOTES
c. When the field breaker is closed and the discharge valve is fully open, both loops are once again added	CWD 729
d. With both loops out of service, they are summed as if they were operating normally	
V. SPEED CONTROL SYSTEM	
A. Flowpath (General)	**TRANSPARENCY 14**
1. Recirc master controller receives a demand signal for a change in pump speed from the operator	
2. The demand signal is processed via the speed controller and scoop tube positioner to the fluid coupler	
3. The fluid coupler varies the MG set slip effecting a change in generator speed	
4. Recirc pump speed changes, changing core flow and hence reactor power	
5. As MG set speed changes the MG set tach. generator provides a feedback signal to the speed control system. It also provides a speed signal to the voltage regulator to ensure 70 volts/cycle	
B. Component Description	
1. Computation Module	
a. Computes the difference between actual reactor steaming rate and desired steaming rate based on speed load changer setting	**TRANSPARENCY 16** **TRANSPARENCY 15** is a simplified version of 16
b. Input to master controller auto circuit	

ATTACHMENT 2

Exam Question 69

Comments: The question tests the use of the Heat Capacity Temperature Limit and Heat Capacity Level Limit curves by requiring the applicant to determine the actual Heat Capacity Level Limit for a given set of degraded conditions.

As noted on the attached EOP charts [REF: OE 3104], a small change in the values interpreted by the applicant could result in a different and more conservative Heat Capacity Level Limit, and therefore a different answer.

Recommendation: **Accept both C and D** as correct answers.

QUESTION: 069 (1.00)

Given the following plant conditions:

- Reactor pressure 500 psig
- Reactor level 125 inches
- Torus water temperature 182 degrees F.

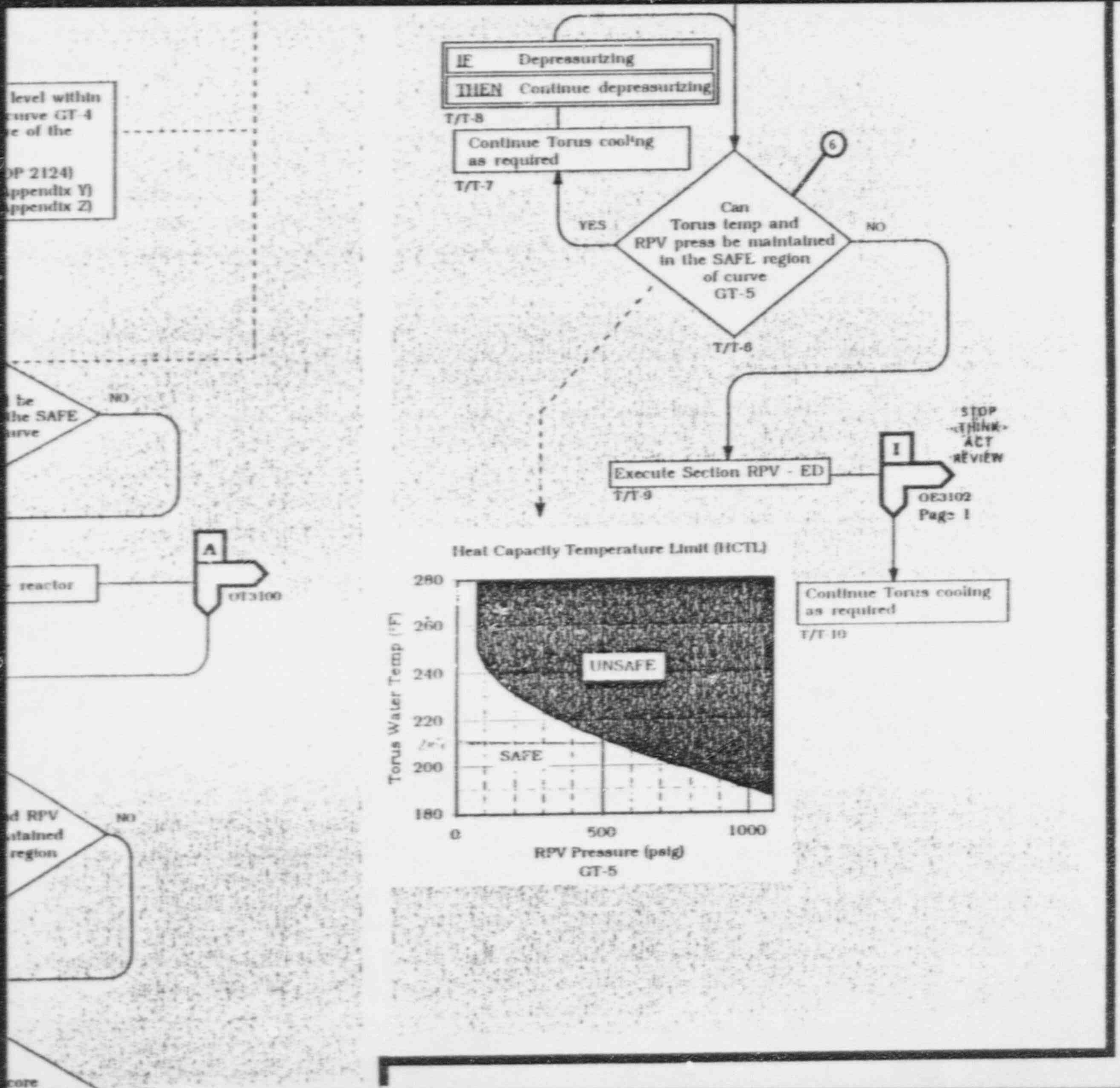
WHICH ONE of the following bands contain the actual Heat Capacity Level Limit?

a. 7.0 to 7.4 feet

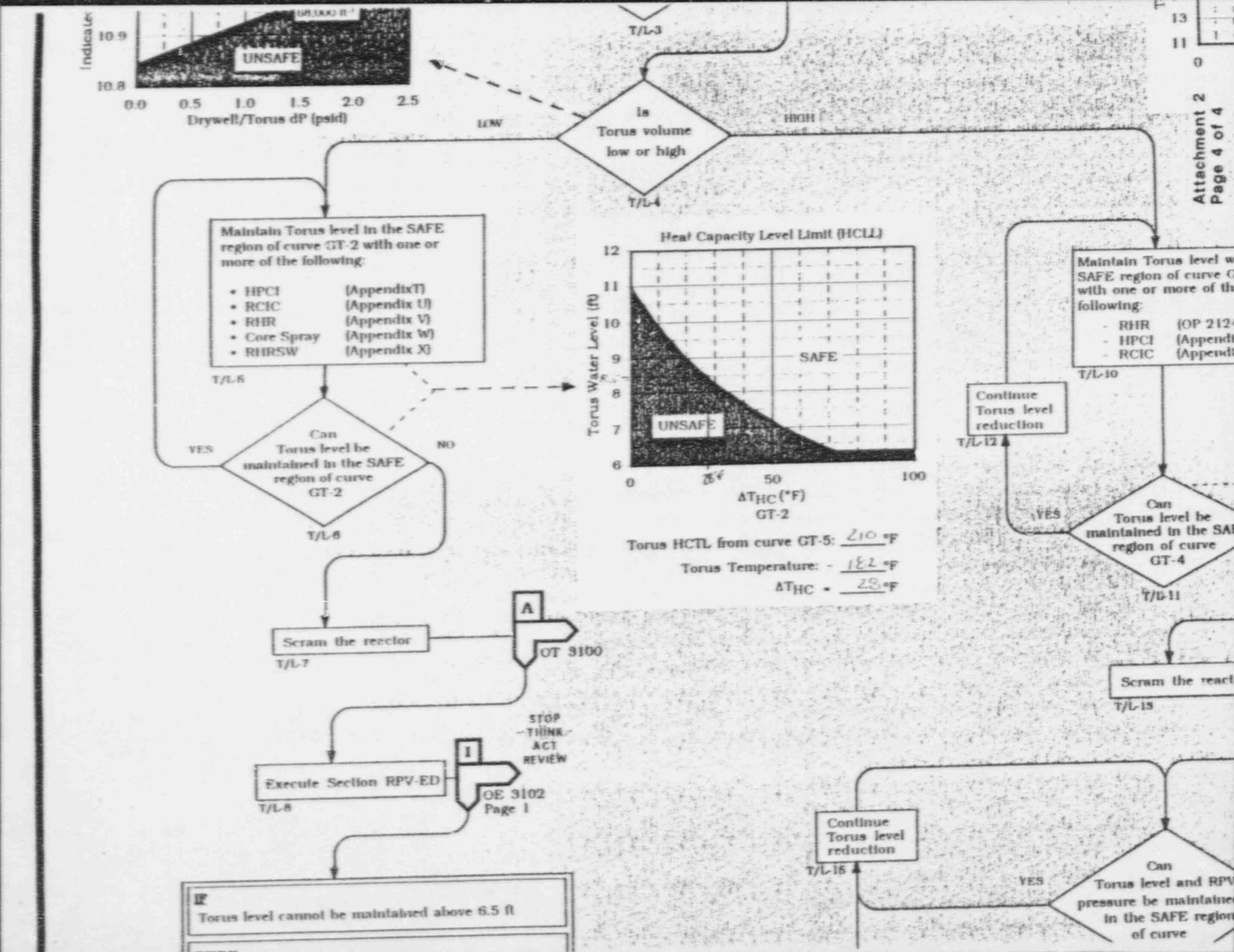
b. 7.5 to 7.9 feet

c. 8.0 to 8.4 feet

d. 8.5 to 8.9 feet



level within
curve GT-4
e of the
OP 2124)
appendix Y)
appendix Z)
be
the SAFE
urve
NO
reactor
A
OT3100
d RPV
ntained
region
NO
core
IF Depressurizing
THEN Continue depressurizing
T/T-8
Continue Torus cooling
as required
T/T-7
YES
Can
Torus temp and
RPV press be maintained
in the SAFE region
of curve
GT-5
NO
T/T-6
6
Execute Section RPV - ED
T/T-9
I
OE3102
Page 1
STOP
THINK
ACT
REVIEW
Continue Torus cooling
as required
T/T-10
Heat Capacity Temperature Limit (HCTL)
Torus Water Temp (°F)
280
260
240
220
200
180
UNSAFE
SAFE
0
500
1000
RPV Pressure (psig)
GT-5


Indicated
10.9
10.8
UNSAFE
0.0 0.5 1.0 1.5 2.0 2.5
Drywell/Torus dP (psid)
T/L-3
Is Torus volume low or high
LOW
HIGH
T/L-4
Maintain Torus level in the SAFE region of curve GT-2 with one or more of the following:
• HPCI (Appendix T)
• RCIC (Appendix U)
• RHR (Appendix V)
• Core Spray (Appendix W)
• RHRSW (Appendix X)
T/L-5
Can Torus level be maintained in the SAFE region of curve GT-2
YES
NO
T/L-6
Scram the reactor
T/L-7
A
OT 3100
STOP -THINK- ACT REVIEW
I
Execute Section RPV-ED
T/L-8
OE 3102 Page 1
IF
Torus level cannot be maintained above 6.5 ft
THEN
Heat Capacity Level Limit (HCLL)
Torus Water Level (ft)
12
11
10
9
8
7
6
SAFE
UNSAFE
0
50
100
ΔT_HC (°F)
GT-2
Torus HCTL from curve GT-5: 210 °F
Torus Temperature: - 182 °F
ΔT_HC = 28 °F
13
11
0
Maintain Torus level wi SAFE region of curve G with one or more of the following:
- RHR (OP 2124
- HPCI (Appendi
- RCIC (Appendi
T/L-10
Continue Torus level reduction
T/L-12
Can Torus level be maintained in the SAF region of curve GT-4
YES
T/L-11
Scram the react
T/L-13
Continue Torus level reduction
T/L-15
Can Torus level and RPV pressure be maintaine in the SAFE region of curve
YES

ATTACHMENT 3

Exam Question 74

Comments: This question tests the applicant's ability to classify an off-site radiation release condition. The question intended that the Radiological Conditions section of the Emergency Plan Classification and Action Level Scheme [REF: AP 3125 Rev. 15 Appendix A (page 1 of 2)] would be used to classify the event.

There are only two ways to get high radiation levels at the fence line a) an event involving the movement of highly radioactive material outside of the reactor building, or b) a transient (accident) within the reactor building. The stem of the question eliminated the first possibility.

Section 14.6 of the FSAR "Analysis of Design Basis Accidents" provides tables for exposures at the fence line for VY's Design Basis Accidents. The tables for the DBA-LOCA (an elevated release) and the DBA-Refueling Accident (a ground release) are attached. For the accidents the tables show the worst case dose received at fence line to be:

DBA-LOCA
2hr exposure from the time of the accident = 0.04mR
24hr exposure from the time of the accident = 2mR

DBA-Refueling Accident
2hr exposure from the time of the accident = 4mR
24hr exposure from the time of the accident = 22mR

The worst case life-time thyroid exposure for either of these accidents is 34mR.

The stated site boundary TEDE in the question is 7 to 75 times higher than the Design Basis Accidents. The stated thyroid CDE at the boundary is 11 to 1400 times higher than that analyzed for the Design Basis Accident.

Based upon the dose levels given in the question stem, the applicant could interpret that an off-site release resulting in the general public receiving such levels of both whole body exposure (due to noble gesses) and thyroid dose (due to Iodine), as indicative of a fuel clad boundary failure, reactor coolant boundary failure, and primary containment barrier failure. As such, the applicant could conclude, based upon the "General" category of the Fission Product Barrier Matrix [REF: AP 3125 Rev 15 Appendix B (page 1 of 1)], that a General Emergency declaration is appropriate.

Recommendation: **Accept both C and D** as correct answers.

QUESTION: 074 (1.00)

Due to a transient an offsite release is in progress. A sample analysis of the discharge as well as a projected offsite dose calculation has been done with the following results.

- Noble gas discharge at the site boundary will result in a dose rate of 2 rem/year total body
- The projected duration of the gaseous release at the site boundary will result in a TEDE of 150 mrem
- The projected duration of the gaseous release at the site boundary will result in a thyroid CDE of 400 mrem

WHAT is the emergency plan classification of this event?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

APPENDIX A

			UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
			OP 3500	OP 3501	OP 3502	OP 3503
1 RADIOLOGICAL CONDITIONS	PARAMETERS	GASEOUS EFFLUENTS (1) (2)	Noble gas discharge to areas at or beyond the site boundary results in a dose rate >500 mRem/yr to the total body or >3000 mRem/yr to the skin. OR I-131, I-133, tritium, radionuclides in particulate form with half lives >8 days result in a dose rate >1500 mRem/yr to any organ. U-1-a	Greater than 10 times the limits for effluents specified for the Unusual Event category. A-1-a		
		LIQUID EFFLUENTS (1) (6a)	Radionuclide concentration (other than noble gases) in liquid released from the site greater than 10CFR20, App. B, Table 2, Col. 2. OR Total activity for dissolved or entrained noble gases >2 E-4 µci/ml. U-1-b	Greater than 10 times the limits for effluents specified for the Unusual Event category. A-1-b		
		AREA & AIRBORNE RADIATION LEVELS		Unexpected area or airborne radiation levels 1000 times normal which could require off site impact assessment. A-1-c		
		SITE BOUNDARY RADIOLOGICAL DOSE			Site boundary dose resulting from an actual or imminent release of gaseous radioactivity for an actual or projected duration such that either of the following limits is exceeded: • TEDE ≥ 100 mRem • thyroid CDE ≥ 500 mRem (3) S-1-d	Site boundary dose resulting from an actual or imminent release of gaseous radioactivity for an actual or projected duration such that either of the following limits is exceeded: • TEDE ≥ 1000 mRem • thyroid CDE ≥ 5000 mRem (3) (4) G-1-a
2 FUEL DAMAGE	PARAMETERS	REACTOR WATER I-131 (1)	Rx water sample analysis exceeds 1.1 µci of I-131 dose equivalent per gram of water for > 2½ hrs. OR Rx water sample analysis reveals an instantaneous spike of 4 µci of I-131 dose equivalent per gram of water. U-2-a	Reactor coolant sample activity >300 µci of I-131 dose equivalent per gram of water. A-2-a		
		AEOG RADIATION LEVELS	Sample analysis determines that the release rate is ≥ 0.16 Ci/sec. (6) U-2-b	Sample analysis determines that the release rate is ≥ 1.6 Ci/sec. A-2-b		
		PRIMARY CONTAINMENT RADIATION LEVELS			Containment radiation monitors reading > 1000 R/hr. S-2-a	
		RPV WATER LEVEL			Inability to maintain reactor water level above -48 inches. S-2-b	
		REFUEL FLOOR RADIATION LEVEL			Refuel floor radiation monitors reading > 1000 mR/hr. S-2-c	
		REACTOR BUILDING VENTILATION RADIATION LEVEL			Reactor building ventilation radiation monitor reading > 140 mR/hr. S-2-d	
		FUEL POOL WATER LEVEL			Spent fuel pool water level below the top of the spent fuel assemblies. S-2-e	
	EVENTS	MAIN STEAM LINE ISOLATION	Main steam line high radiation isolation. U-2-c			
		SPENT FUEL ASSEMBLY ACCIDENT WITH RELEASE TO THE REACTOR BUILDING		Spent fuel assembly accident with release to the reactor building resulting in trip of the reactor building ventilation and start of the Standby Gas Treatment system due to either: a. Refuel floor radiation monitors high radiation trip OR b. Reactor building ventilation radiation monitor high radiation trip. A-2-c		
		LOSS OF 2 OF 3 FISSION				Loss of 2 of 3 fission product barriers with the potential loss of the third

(1) To be determined by sample analysis.

(2) The Stack Gas Rad Mon Alert Alarm should be used to indicate when an analysis must be performed.

(3) Integrated dose to be determined by:
1. Direct field monitoring
OR
2. Projected off site dose calculation.

(4) If limits are exceeded, then protective action recommendations should be made per OP 3511.

(5) The AEOG-15 RAD alarm should be used to indicate when an analysis must be performed.

(6) The AEOG-16-18 RAD alarm should be used to indicate when an analysis must be performed.

(6a) The following parameters should be used to indicate when sample analysis is necessary:
- 3-E-7: RADWASTE EFFLUENT RAD-18
- 3-F-7: DISC. CANAL RAD-18
- 3-F-6: SERVICE WATER EFFLUENT RAD-18

REACTOR COOLANT SYSTEM LEAK RATE			Reactor Coolant System leak rate > 50 gpm inside the drywell (EAL no. A-3-a) OR Unisolable reactor coolant leakage outside the drywell as indicated by high reactor building area temperatures or area radiation levels.			
PRIMARY CONTAINMENT ISOLATION						Failure of two in series PCIS valves to isolate a release pathway OR observed structural failure OR Unisolable reactor coolant leakage outside the drywell as indicated by high reactor building area temperatures or radiation levels.
OTHER				Failure of a primary coolant safety valve to close with reactor coolant Temperature ≥ 212 °F. OR Failure of a primary coolant safety relief valve to close with reactor coolant Temperature ≥ 212 °F.	Explosive hydrogen/oxygen mixture exists within either the drywell or torus air space. (i.e.: ≥ 6 % H_2 and ≥ 5 % O_2 mixtures)	
GENERAL	Any condition which indicates LOSS or POTENTIAL LOSS of the FUEL CLAD barrier.		Any condition which indicates LOSS or POTENTIAL LOSS of the REACTOR COOLANT SYSTEM barrier.		Any condition which indicates LOSS or POTENTIAL LOSS of the PRIMARY CONTAINMENT barrier.	

AP 3125
ATTACHMENT B

TABLE 14.6.8A

Loss-of-Coolant Accident - Radiological Effects
2-Hour Dose

Distance (Miles)	Meteorological Conditions					
	VS-1	MS-1	N-1	N-5	U-1	U-5
Passing Cloud Whole Body Dose (Rem)						
1/11*	2.9E-05	2.9E-05	3.0E-05	5.2E-06	3.6E-05	5.8E-06
1/2	2.3E-05	2.3E-05	2.6E-05	3.7E-06	3.1E-05	4.5E-06
1	1.6E-05	1.6E-05	2.0E-05	2.8E-06	1.6E-05	2.5E-06
5	4.5E-06	4.9E-06	2.8E-06	6.9E-07	1.3E-06	3.0E-07
10	2.2E-06	2.3E-06	7.9E-07	2.7E-07	3.2E-07	1.0E-07
Lifetime Thyroid Dose (Rem)						
1/11*	0.	1.4E-14	2.8E-13	8.7E-19	1.5E-06	9.9E-08
1/2	0.	2.2E-10	8.9E-07	3.7E-08	5.3E-06	1.0E-06
1	7.4E-31	1.6E-08	2.5E-06	3.9E-07	2.3E-06	5.2E-07
5	6.2E-14	5.1E-07	4.4E-07	1.2E-07	1.8E-07	4.7E-08
10	1.7E-10	5.0E-07	1.6E-07	4.5E-08	6.3E-08	1.7E-08

* Site Boundary (283 meters)

	Meteorology	Wind Speed (M/S)
VS-1	Very Stable	1
MS-1	Moderately Stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

NOTE: 2.9E-05 = 2.9×10^{-5}

VYNPS

TABLE 14.6.8B

Loss-of-Coolant Accident - Radiological Effects
24-Hour Dose

Distance (Miles)	VS-1	MS-1	N-1	N-5	U-1	U-5
	Meteorological Conditions					
Passing Cloud Whole Body Dose (Rem)						
1/11*	1.6E-03	1.6E-03	1.6E-03	2.9E-04	2.0E-03	3.2E-04
1/2	1.3E-03	1.3E-03	1.4E-03	2.0E-04	1.7E-03	2.5E-04
1	8.8E-04	8.8E-04	1.1E-03	1.5E-04	8.8E-04	1.4E-04
5	2.5E-04	2.7E-04	1.6E-04	3.8E-05	7.0E-05	1.6E-05
10	1.2E-04	1.3E-04	4.3E-05	1.5E-05	1.8E-05	5.7E-06
Lifetime Thyroid Dose (Rem)						
1/11*	0.	7.3E-13	1.5E-11	4.6E-17	7.8E-05	5.2E-06
1/2	0.	1.2E-08	4.7E-05	2.0E-06	2.8E-04	5.4E-05
1	3.9E-29	8.3E-07	1.3E-04	2.0E-05	1.2E-04	2.7E-05
5	3.2E-12	2.7E-05	2.3E-05	6.3E-06	9.5E-06	2.5E-06
10	9.2E-09	2.6E-05	8.5E-06	2.4E-06	3.3E-06	8.7E-07

* Site Boundary (283 meters)

	Meteorology	Wind Speed (M/S)
VS-1	Very Stable	1
MS-1	Moderately Stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

NOTE: 1.6E-03 = 1.6×10^{-3}

Revision 13

TABLE 14.6.11A

Refueling Accident - Radiological Effects
2-Hour Dose

Distance (Miles)	Meteorological Conditions VS-1	MS-1	N-1	N-5	U-1	U-5
Passing Cloud Whole Body Dose (Rem)						
1/11*	3.1E-03	3.1E-03	3.1E-03	5.5E-04	3.8E-03	6.1E-04
1/2	2.4E-03	2.4E-03	2.7E-03	3.9E-04	3.3E-03	4.7E-04
1	1.7E-03	1.7E-03	2.1E-03	2.9E-04	1.7E-03	2.6E-04
5	4.8E-04	5.2E-04	3.0E-04	7.2E-05	1.3E-04	3.1E-05
10	2.3E-04	2.5E-04	8.3E-05	2.8E-05	3.4E-05	1.1E-05
Lifetime Thyroid Dose (Rem)						
1/11*	0.	1.3E-11	2.6E-10	8.1E-16	1.4E-03	9.2E-05
1/2	0.	2.1E-07	8.4E-04	3.5E-05	4.9E-03	9.7E-04
1	6.9E-28	1.5E-05	2.3E-03	3.6E-04	2.1E-03	4.8E-04
5	5.8E-11	4.7E-04	4.1E-04	1.1E-04	1.7E-04	4.4E-05
10	1.6E-07	4.7E-04	1.5E-04	4.2E-05	5.9E-05	1.5E-05

* Site Boundary (283 meters)

	Meteorology	Wind Speed (M/S)
VS-1	Very Stable	1
MS-1	Moderately Stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

NOTE: 3.1E-03 = 3.1×10^{-3}

TABLE 14.6.11B

Refueling Accident - Radiological Effects
24-Hour Dose

Distance (Miles)	VS-1	MS-1	N-1	N-5	U-1	U-5
	Meteorological Conditions					
Passing Cloud Whole Body Dose (Rem)						
1/11*	1.8E-02	1.8E-02	1.8E-02	3.2E-03	2.2E-02	3.5E-03
1/2	1.4E-02	1.4E-02	1.6E-02	2.2E-03	1.9E-02	2.7E-03
1	9.6E-03	9.6E-03	1.2E-02	1.7E-03	9.6E-03	1.5E-03
5	2.7E-03	3.0E-03	1.7E-03	4.1E-04	7.6E-04	1.8E-04
10	1.3E-03	1.4E-03	4.8E-04	1.6E-04	1.9E-04	6.2E-05
Lifetime Thyroid Dose (Rem)						
1/11*	0.	8.9E-11	1.8E-09	5.6E-15	9.7E-03	6.4E-04
1/2	0.	1.5E-06	5.8E-03	2.4E-04	3.4E-02	6.7E-03
1	4.8E-27	1.9E-04	1.6E-02	2.5E-03	1.5E-02	3.4E-03
5	4.0E-10	3.3E-03	2.9E-03	7.8E-04	1.2E-03	3.0E-04
10	1.1E-06	3.3E-03	1.1E-03	3.0E-04	4.1E-04	1.1E-04

* Site Boundary (283 meters)

	Meteorology	Wind Speed (M/S)
VS-1	Very Stable	1
MS-1	Moderately Stable	1
N-1	Neutral	1
N-5	Neutral	5
U-1	Unstable	1
U-5	Unstable	5

NOTE: 1.8E-02 = 1.8×10^{-2}

Revision 13

ATTACHMENT 4

Exam Question 90

Comments: This question tests the applicant's knowledge of the bases for the reactor protection signals that protect the reactor during a Main Steam Line isolation event. There are two issues relative to this question:

(1) Technical Specifications [REF: TS Bas: ..1.G Amendment 84 (page 17)] states that the Main Steam Line Isolation Valve (MSIV) Closure Scram "anticipates the pressure and flux transients." In addition, the FSAR [REF: FSAR Section 14.5.1.3.1 (page 14.5-4)] specifically refers to the high neutron flux scram as a **backup/indirect** means of shutting down the reactor.

As an anticipatory signal, the MSIV Closure Scram is thus the primary protection for this event. Answers A, C, and D are therefore incorrect.

(2) Technical Specifications [REF: TS Bases 1.2 and 2.2 Amendment 18 (page 19)] states that the indirect scram signal for an MSIV closure is APRM High Flux, with High Pressure as a backup to the APRM High Flux Scram. The High Pressure Scram is therefore one of the backup signals for an MSIV closure event.

Based upon the above listed items, Answer B is the only correct answer.

Recommendation: Change correct answer to **B.**

QUESTION: 090 (1.00)

WHICH ONE of the following describes how RPS is designed to protect against steam line isolation transients at full reactor power?

a. APRM neutron flux is the primary scram signal.
High reactor pressure is the backup scram signal.

b. MSIV closure is the primary scram signal.
High reactor pressure is the backup scram signal.

c. High reactor pressure is the primary scram signal.
APRM neutron flux is the backup scram signal.

d. High reactor pressure is the primary scram signal.
MSIV closure is the backup scram signal.

BASES: 2.1 (Cont'd)

metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists.

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram setpoint at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the available of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

14.5.1.3.1 Closure of All Main Steam Line   ation Valves

The ASME Boiler and Pressure Vessel Code requir   overpressure protection for each vessel designed to meet Code Section III. For the plant, the transient produced by the fast closure (3.0 seconds) of all main steam line isolation valves represents the most severe abnormal operational transient resulting in a nuclear system pressure rise when direct scrams are ignored. The Code overpressure protection analysis hypothetically assumes the failure of the direct isolation valve position scram. The reactor is shutdown by the backup, indirect, high neutron flux scram. This event can be categorized as a core dynamic event for analysis purposes.

Analysis of the event demonstrates that the installed safety valve capacity of 28.35% of rated flow, in conjunction with relief capacity of 49.7% of rated flow, limits the peak Nuclear System pressure at vessel invert to less than 1,375 psig. The margin to the ASME Code limit assures adequate protection against excessive overpressurization of the Nuclear System process barrier even for this hypothetical isolation event. Table 14.5.1 lists the peak values of the key process variables for this transient. Figures 14.5-5 and 14.5-6 graphically show the results produced by this simulated analysis.

14.5.2 Events Resulting in a Reactor Vessel Water Temperature Decrease

Events that result directly in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the vessel or reduce the temperature of water being delivered to the vessel. The events that result in the most severe transients in this category are the following:

1. Loss of a Feedwater Heater
2. Shutdown Cooling (RHRS) Malfunction - Decreasing Temperature
3. Inadvertent Pump Start
4. Loss of Stator Cooling

14.5.2.1 Loss of a Feedwater Heater

A feedwater heater can be lost in at least two ways: (1) if the steam extraction line to the heater is shut, the heat supply to the heater is removed, producing a gradual cooling of the feedwater, and (2) a bypass line is usually provided so that the feedwater flow can be passed around rather than through the heater. In either case, the reactor vessel receives cooler feedwater which produces an increase in core inlet subcooling. Due to the negative void reactivity coefficient, an increase in core power results. The

BASES:

1.2 REACTOR COOLANT SYSTEM

The reactor coolant system is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, and the coolant system piping. The respective design pressures are 1250 psig at 575°F and 1148 psig at 560°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% x 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% x 1148 = 1378 psig).

The safety valves are sized to prevent exceeding the pressure vessel code limit for the worst-case isolation (pressurization) event (MSIV closure) assuming indirect (neutron flux) scram.

2.2 REACTOR COOLANT SYSTEM

The settings on the reactor high pressure scram, reactor coolant system relief and safety valves, have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition to preventing power operation above 1055 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients. (See FSAR Section 14.5 and Supplement 2 to Proposed Change No. 14, November 12, 1973.)